Docket No.: 50-423

AUG 2 1984

Mr. W. G. Counsil
Senior Vice President
Nuclear Engineering and Operations
Northeast Nuclear Energy Company
Post Office Box 270
Hartford, Connecticut 06141-0270

Dear Mr. Counsil:

Subject: Issuance of Safety Evaluation Report - NUREG-1031 - Millstone Nuclear Power Station, Unit No. 3

The U. S. Nuclear Regulatory Commission has issued the Safety Evaluation Report related to operation of the Millstone Nuclear Power Station, Unit No. 3. Two copies of this report are enclosed for your use. Twenty additional copies will be forwarded when they have returned from our printer-contractor.

Sincerely,

/s/

B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing

Enclosure:
SER - NUREG-1031 (2 copies)

cc: See next page

CONCURRENCES:

DL:LB#1	DL:LB#1	DL:LB#1
MRushbrook:es	ELDoolittle	BJYoungblood
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DIST:
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NRC PDR
PRC System
NSIC
Local PDR
LB#1 Rdg
MRushbrook
ELDoolittle
OELD, Attorney
ACRS 16
EJordan
NGrace

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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cc: See next page

MILLSTONE

Mr. W. G. Counsil
Senior Vice President
Nuclear Engineering and Operations
Northeast Nuclear Energy Company
Post Office Box 270
Hartford, Connecticut 06141-0270

cc: Gerald Garfield, Esq.
Day, Berry & Howard
City Place
Hartford, Connecticut 06103-3499

Mr. Maurice R. Scully, Executive
Director
Connecticut Municipal Electric
Energency Cooperative
268 Thomas Road
Groton, Connecticut 06340

Robert W. Bishop, Esq.
Corporate Secretary
Northeast Utilities
Post Office Box 270
Hartford, Connecticut 06141

Mr. T. Rebelowski
Senior Resident Inspector Office
U. S. Nuclear Regulatory Commission
Millstone III
P. O. Box 615
Waterford, Connecticut 06385

Mr. Michael L. Jones, Manager
Project Management Department
Massachusetts Municipal Wholesale
Electric Company
Post Office Box 426
Ludlow, Massachusetts 01056

Mr. Thomas Murley
U. S. Nuclear Regulatory Commission
Region I
631 Park Avenue
King of Prussia, Pennsylvania 19406

Mr. Brian Norris
Public Affairs Office
U.S.N.R.C. - Region I
King of Prussia, Pennsylvania 19406

cc: Attorney General
30 Trinity Street
Hartford, Connecticut 06115

Connecticut Energy Agency
Energy Division
Office of Policy & Management
80 Washington Street
Hartford, Connecticut 06115

First Selectman
Town of Waterford
Hall of Records
1000 Boston Post Road
Waterford, Connecticut 06385

Attorney General
Department of Law
State Capitol
Albany, New York 12224

Director, Technical Development Programs
State of New York Energy Office
Agency Building 2
Empire State Plaza
Albany, New York 12223

Governor's Energy Office
80 Dean Street
Providence, Rhode Island 02903

Director, Criteria and Standards Division
Office of Radiation Programs (ANR-460)
U. S. Environmental Protection Agency
Washington, D. C. 20460

Director, Eastern Environmental
Radiation Facility
U. S. Environmental Protection Agency
P. O. Box 3009
Montgomery, Alabama 36193

EIS Review Coordinator
EPA Region I
JFK Federal Building
Boston, Massachusetts 02203

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NUREG-1031

Safety Evaluation Report
related to the operation of Millstone Nuclear Power Station, Unit No. 3

Docket No. 50-423

Northeast Nuclear Energy Company

U.S. Nuclear Regulatory Commission

Office of Nuclear Reactor Regulation

JULY 1984

NUREG-1031 SER RELATED TO THE OPERATION OF MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3 JULY 1984

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ABSTRACT

The Safety Evaluation Report for the application filed by Northeast Nuclear Energy Company, as applicant and agent for the owners, for a license to operate the Millstone Nuclear Power Station Unit 3 (Docket No. 50-423), has been prepared by the Office of Nuclear Reactor Regulation of the U.S. Nuclear Regulatory Commission. The facility is located in the town of Waterford, New London County, Connecticut, on the north shore of Long Island Sound. Subject to favorable resolution of the items discussed in this report, the NRC staff concludes that the facility can be operated by the applicant without endangering the health and safety of the public.

1 INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 Introduction

This report is a Safety Evaluation Report (SER) on the application for an operating license (OL) for the Millstone Nuclear Power Station Unit 3. The Millstone Point Company and joint applicants filed with the Atomic Energy Commission (AEC) an application docketed on February 10, 1973, for a license to construct and operate the proposed Millstone Nuclear Power Station Unit 3 (Millstone Unit 3 or facility). The site is located in the town of Waterford, New London County, Connecticut, on the north shore of Long Island Sound.

The AEC (now the Nuclear Regulatory Commission (NRC or Commission)) reported the results of its preconstruction review in an SER dated May 16, 1974. Following a public hearing before an Atomic Safety and Licensing Board, Construction Permit No. CPPR-113 was issued on August 9, 1974.

The NRC staff has since reviewed and approved 11 applications to amend CPPR-113 to reflect transfer of fractional share ownership among co-owners. This information is included in the Millstone 3 docket file. The following is a list of owners as approved up to and including Amendment 11 to CPPR-113:

The Connecticut Light & Power Company
Western Massachusetts Electric Company
New England Power Company
The United Illuminating Company
Public Service Company of New Hampshire
Central Vermont Public Service Corporation
Montaup Electric Company
City of Burlington, Vermont Electric Light Department
Chicopee Municipal Lighting Plant
Massachusetts Municipal Wholesale Electric Company
Vermont Electric Cooperative, Inc.
Vermont Electric Generation and Transmission Cooperative, Inc.
Central Maine Power Company
Village of Lyndonville Electric Department
Connecticut Municipal Electric Energy Cooperative
Fitchburg Gas and Electric Light Company

The Northeast Nuclear Energy Company (hereinafter referred to as the applicant) acting as agent and representative for the owners tendered an application for an operating license for Millstone Unit 3 by letter dated October 29, 1982. When NRC staff acceptance review was completed, the Final Safety Analysis Report (FSAR) for Millstone Unit 3 was docketed by a letter dated February 2, 1983.

Before issuing an OL for a nuclear power plant, the NRC staff is required to conduct a review of the effects of the plant on public health and safety. The staff safety review of Millstone Unit 3 has been based on NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Reactors, LWR Edition" (SRP). An audit review of each of the areas listed in the Areas of

Review section of the SRP was performed according to the guidelines provided in the Review Procedures portion of the SRP. Exceptions to this practice are noted in the applicable sections of this report.

This SER summarizes the results of the staff's radiological safety review of Millstone Unit 3 and delineates the scope of the technical details considered in evaluating the radiological safety aspects of its proposed operation. The design of the station was reviewed against the Federal regulations, CP criteria, and the SRP, except where noted otherwise. The SRP covers a variety of site conditions and plant designs. Each section is written to provide the complete procedure and all acceptance criteria for all of the areas of review pertinent to the section. However, for any given application, the staff may select and emphasize particular aspects of each SRP section as appropriate for the application. In some cases, the major portion of the review of a plant feature may be done on a generic basis, with the designer of that feature rather than in the context of reviews of particular applications from utilities. In other cases, a plant feature may be sufficiently similar to that of a previous plant so that a de novo review of the feature is not needed.

During the course of its review, the staff held a number of meetings with representatives of the applicant to discuss the design, construction, and proposed operation of the plant. The staff requested additional information, which the applicant provided in amendments to the FSAR. This information is available to the public for review at the NRC Public Document Room at 1717 H Street, N.W., Washington, D.C. and at the Local Public Document Room at the Waterford Public Library, Rope Ferry Road, Route 156, Waterford, Connecticut 06385.

Following the accident at Three Mile Island Unit 2 (TMI-2), the Commission paused in its licensing activities to assess the impact of the accident. During this pause, the recommendations of several groups established to investigate the lessons learned from the TMI-2 accident became available. All available recommendations were correlated and assimilated ino a "TMI Action Plan," now published as NUREG-0660, entitled "NRC Action Plan Developed as a Result of the TMI-2 Accident." Additional guidance relating to implementation of the Action Plan is in NUREG-0737, "Clarification of TMI Action Plan Requirements," and in Supplement 1 to NUREG-0737. Licensing requirements based on the lessons learned from the TMI-2 accident have been established to provide additional safety margins. These have been incorporated into the design and operation of Millstone Unit 3. Table 1.1 provides a cross-reference relating the TMI items to the sections in this report where they are discussed.

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

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

As part of its review of the application against the NRC regulations, the staff will ask the applicant to certify that Millstone Unit 3 meets the applicable requirements of 10 CFR 20, 50, 51, and 100. Following the applicant's response to this request, the staff will address its findings in this area in a supplement to this Safety Evaluation Report.

In accordance with the provisions of the National Environmental Policy Act (NEPA) of 1969, a Draft Environmental Statement (DES) (NUREG-1064) that sets forth the environmental considerations related to the proposed construction and operation of Millstone Unit 3 was prepared by the staff and was published in July 1984. The Final Environmental Statement (FES) is scheduled to be published in November 1984 and will include a consideration of public comments received on the DES.

The applicant submitted a probabilistic safety study (PSS) on August 1, 1983. The study included an analysis of internally initiated events as well as analyses of the contribution of externally initiated events. The staff is currently evaluating this information and has reflected the results of its evaluation in the DES under "Environmental Impact of Postulated Accidents." Any safety-related findings resulting from the staff's review of the PSS will be reported in a supplement to this report.

The review and evaluation of Millstone Unit 3 for an operating license is only one of many stages at which the staff reviews the design, construction, and operating features of the facility. The facility design was extensively reviewed before the applicant was granted a construction permit for the facility. Construction of the facility has been monitored in accordance with a detailed monitoring and inspection program at the OL stage. The NRC staff has reviewed the final design of the facility to determine that the Commission's regulations have been met. If an operating license is granted, Millstone Unit 3 must be operated in accordance with the terms of the operating license and the Commission's regulations, and the facility will be subject to the staff's continuing inspection program.

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

The NRC Project Manager assigned to the OL application for Millstone Unit 3 is Ms. Elizabeth L. Doolittle. Ms. Doolittle may be contacted by calling (301) 492-4911 or by writing

Ms. Elizabeth L. Doolittle
Division of Licensing
U.S. Nuclear Regulatory Commission
Washington, DC 20555

1.2 General Plant Description

The Millstone Nuclear Power Station consists of three individual nuclear power plant facilities. Millstone Unit 1 uses a single-cycle boiling-water reactor supplied by General Electric Company with a net electrical output of 660 MW; it was licensed to operate in October 1970. Millstone Unit 2 uses a two-loop pressurized-water reactor supplied by Combustion Engineering, Inc., with a net electrical output of 870 MW; it was licensed to operate in September 1975. Millstone Unit 3 will use a four-loop pressurized-water reactor supplied by Westinghouse Electric Corporation with a net calculated electrical output of approximately 1,156 MW; it is now under construction. Stone & Webster Engineering Corporation is the architect-engineer for Millstone Unit 3.

The major structures of the Millstone Unit 3 facility include a containment structure, containment enclosure building, auxiliary building, fuel building, waste disposal building, engineered safety features building, main steam valve building, turbine building, service building, control building, emergency generator enclosure, and circulating and service water pumphouse.

The containment structure houses the nuclear steam supply system (NSSS). The NSSS incorporates a pressurized-water reactor and a four-loop reactor coolant system (RCS). Each loop contains a reactor coolant pump and steam generator, two-loop isolation valves, an isolation bypass valve, and a bypass line. The NSSS also contains an electrically heated pressurizer and auxiliary systems. The NSSS is designed for a power output of 3,565 MWt with a gross electrical output of 1,209 MWe and a net electrical output of approximately 1,156 MWe.

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

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

The reactor is controlled during operation by control rod movement and regulation of boric acid concentration in the reactor coolant. Mechanical rod cluster control assemblies consist of stainless-steel-clad hafnium neutron absorber rods that are inserted in Zircaloy guide tubes located in certain fuel assemblies. The rod cluster control assemblies are attached to stainless steel drive shafts, which will be raised and lowered within the core by individual drive mechanisms mounted on the reactor vessel head.

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

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

The steam generators are vertical shell and U-tube evaporators, which contain Inconel tubes; they are Westinghouse Model F. The steam produced in the steam generators will be used to drive a tandem-compound, six-flow, 1,800-rpm turbine generator and will be condensed in a single-pass, deaerating surface condenser with six inlet and outlet water boxes. These components are housed in the turbine building. Condenser circulating water is drawn from Long Island Sound and supplied to the condenser by six circulating water pumps located in the circulating and service water pumphouse. Circulating water is pumped through the tubes of the condenser to remove heat from, and thus condense, the steam after it has passed through the turbine. The condenser is equipped with titanium condenser tubes, which resist the corrosive action of seawater, aluminum bronze tube sheet, and copper nickel water box cladding. A cathodic protection system is provided in the water boxes to protect against galvanic corrosion.

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

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

Supervision and control of both the NSSS and the steam and power conversion system will be accomplished from the Millstone Unit 3 control room, located in the control building. The control room contains all instrumentation and control equipment required for startup, operation, and shutdown, including normal and accident conditions.

The emergency core cooling system (ECCS) is designed to cool the reactor core and to provide shutdown capability by injecting borated water during certain accident conditions. ECCS components are located in the containment structure, auxiliary building, engineered safety features building and its adjacent yard. The ECCS consists of safety injection accumulators, charging pumps, safety injection pumps, residual heat removal pumps and heat exchangers, containment recirculation pumps and coolers, and the refueling water storage tank along with the associated piping, valves, instrumentation, and other related equipment. The active components of the ECCS are powered from separate safety-related buses, which are energized from offsite power supplies. In addition, emergency diesel generators ensure redundant sources of auxiliary onsite power in the event of a loss of offsite power. The emergency diesel generators are located within separate compartments in the emergency generator enclosure building.

The containment structure housing the NSSS is carbon-steel-lined, reinforced concrete maintained at a subatmospheric pressure of between 9.5 and 11.5 psia. The containment depressurization system consists of the quench spray and the containment recirculation systems. During accident conditions, both systems are used to reduce the containment to subatmospheric pressure. The quench spray system sprays a mixture of borated water and sodium hydroxide to remove iodine and other radionuclides from the containment atmosphere. A separate steel-framed enclosure building with metal siding and a metal roof deck encloses the containment building. The space between the reinforced concrete structure and the enclosure building will confine any leakage that might occur from penetrations and through the reinforced concrete structure walls. This leakage will be filtered and exhausted to the atmosphere by the supplementary leak collection and release system.

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

1.3 Key Features of Plant and Site

The principal features of the design of Millstone Unit 3 are similar to those that have been evaluated and approved previously by the staff for other nuclear power plants now under construction or in operation, especially North Anna Units 1 and 2, Surry Units 1 and 2, Comanche Peak Units 1 and 2, W. B. McGuire Units 1 and 2, Maine Yankee, and Trojan. To the extent feasible and appropriate, the staff has made use of previous evaluations of these plants in conducting the review of Millstone Unit 3. Where this has been done, the appropriate sections of this report identify the other facilities involved. The staff safety evaluations for these facilities have been published and are available for public inspection at the NRC public document room. Table 1.2 compares the principal design features between Millstone Unit 3 and other facilities.

The design of Millstone Nuclear Power Station Unit 3 includes key design features (listed below with the appropriate section of this report referenced) that have been or are being reviewed by the staff. They include

(1) Subatmospheric Containment

The reactor is operated within a reinforced concrete containment structure maintained at a subatmospheric pressure between 9.5 and 11.5 psia. As a result of operating at the lower starting air partial pressure in the containment, the containment can be returned to subatmospheric pressure by the use of containment depressurization systems following a loss-of-coolant accident. Subatmospheric containment results in the termination of outleakage within 60 min after initiation of the accident. (Section 6.2.2)

(2) Supplementary Leak Collection and Release System (SLCRS)

The containment structure is housed within the containment enclosure building, which, along with structures adjacent to the containment, forms the boundary of the SLCRS. The SLCRS maintains a negative pressure in the containment enclosure building and contiguous structures after a design-basis accident (DBA) and collects and filters the leakage before its release to the atmosphere through the Millstone Unit 1 stack. (Section 6.2.3)

(3) Refueling Water Storage Tank (RWST)

The RWST holds approximately 1.2 million gallons of borated water. The larger capacity of the tank provides a longer period for quench spray and a greater volume of cold water available to return the containment to subatmospheric conditions. (Section 6.2.2)

(4) Loop Isolation Valves

There are two double-disk, remotely controlled, motor-operated loop isolation valves in each loop; one is located between the reactor vessel and the steam generator and the other between the reactor vessel and the reactor coolant pump of each of the four loops. The valves permit isolation of the reactor coolant pumps and steam generators, which is advantageous for maintenance activities. (Section 5.1)

(5) Safety-Grade Cold Shutdown

The cold shutdown design enables the nuclear steam supply system to be taken from hot standby to cold shutdown conditions using only safety-grade systems, with or without offsite power, and with the most limiting single failure. The cold shutdown design also enables the reactor coolant system to be taken from hot standby to conditions that will permit initiation of RHRS operation within 36 hours. (Section 5.4.7.1)

1.4 Significant Issues

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

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

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

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

(4) a major disagreement between the staff and applicant

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

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

For Millstone Unit 3, the following is a list of such issues:

(1) Millstone Unit 3 Seismic Capability Beyond Design Basis

Regarding the 1982 New Brunswick earthquake, the staff concludes that the data do not satisfactorily demonstrate the existence of a unique tectonic structure in the New Brunswick epicentral area. However, it is the staff's judgment that the existing design basis is adequate with respect to the impact of the New Brunswick earthquake on the Millstone site. The conclusion is supported by the earthquake recurrence statistics and the valuable insights gained as part of the probabilistic seismic hazard studies. Sufficient uncertainty in the above data exists, so that the staff requires a confirmatory program using available plant-specific information regarding the plant's seismic capability beyond the design basis. (See Section 2.5.2.)

(2) Fire Protection in Cable Spreading Room

The primary fire suppression in the cable spreading room is a total flooding automatic carbon dioxide system. The staff requires the applicant to provide a fixed water-suppression system as a backup to the carbon dioxide system to meet the guidelines of BTP CMEB 9.5-1, Section C.7.C. This is an outstanding item. (See Section 9.5.1.)

(3) Loading Combinations

The applicant has not included the LOCA loads in his evaluation of the faulted condition limits for all ASME Code, Class 1, 2, and 3 balance-of-plant piping and their supports. Furthermore, the applicant has not yet addressed how the guidelines of NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary Systems," have been satisfied. The applicant intends to request an exemption from GDC 4 and the need to consider reactor coolant loop pipe breaks for Millstone Unit 3. The staff requires that current primary loop heavy component support design margins be maintained (i.e., LOCA loads included) even though the "leak-before-break" concept has been proposed. The applicant has clarified in a letter dated July 20, 1984, that LOCA loads are included in the reactor coolant loop heavy component support design. Upon submittal, the staff will review the applicant's request for exemption from GDC 4 in order to determine the extent and acceptability of the exemption. (See Section 3.9.3.1.)

1.5 Outstanding Items

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

1.6 Confirmatory Items

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

1.7 License Condition Items

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

1.8 Unresolved Safety Issues

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

> Unresolved Safety Issues Plan
>
> Section 210. The Commission shall develop a plan for providing for specification and analysis of unresolved safety issues relating to nuclear reactors and shall take such action as may be necessary to implement corrective measures with respect to such issues. Such plan shall be submitted to the Congress on or before January 1, 1978 and progress reports shall be included in the annual report to the Congress thereafter.

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

Since the issuance of NUREG-0410, each annual report has described NRC progress in resolving these issues.

The staff continually evaluates the safety requirements used in its review against new information as it becomes available. In some cases, the staff takes immediate action or interim measures to ensure safety. In most cases, however, the initial staff assessment indicates that immediate licensing actions or changes in licensing criteria are not necessary. In any event, further study may be deemed appropriate to make judgments as to whether existing staff requirements should be modified. These issues being studied are sometimes called generic safety issues because they are related to a particular class or type of nuclear facility. A discussion of these matters and the NRC program for the resolution of these generic issues is provided in Appendix C to this report, which includes references to sections of this report for specific discussions concerning Millstone Unit 3.

1.9 Identification of Agents and Contractors

Northeast Nuclear Energy Company (NNECo) acts as the agent for the applicants and is responsible for the operation, maintenance, and testing of Millstone Unit 3. NNECo uses the technical support services of Northeast Utilities Service Company (NUSCo) in engineering, design, procurement, construction, accounting, planning, and quality assurance activities.

The applicant has retained Stone & Webster Engineering Corporation in Boston, Massachusetts, to perform architectural-engineering and construction management services for Millstone Unit 3. The Westinghouse Electric Corporation designed, manufactured, and delivered to the site the NSSS and initial core for Millstone Unit 3.

The turbine generator is manufactured by General Electric Company.

The applicant utilizes consultants, as required in specialized areas; for example, S. M. Stroller Corporation assists in the areas of nuclear fuel economics, reactor core physics, mechanical design, and plant safety, including evaluation of radioactivity releases to the environment; Nuclear Energy Services, Incorporated, provides inservice inspection consulting services to Millstone Unit 3; Teledyne Engineering Services assists in the area of Class I piping analysis and special piping component analysis; and several consultants in the area of geotechnical services include Weston Geophysical Engineers, Incorporated, and Geotechnical Engineers, Incorporated.

1.10 Summary of Principal Review Matters

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

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

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

(3) The expected response of the facility to various anticipated operating transients and to a broad spectrum of postulated accidents. On the basis of this evaluation, the staff determined that the potential calculated consequences of a few highly unlikely postulated accidents (design-basis accidents) would exceed those of all other accidents considered. The staff

performed conservative analyses of these design-basis accidents to determine that the calculated potential offsite radiation doses that might result - in the very unlikely event of their occurrence - would not exceed the Commission guidelines for site acceptability given in 10 CFR 100.

(4) The applicant's engineering and construction organization, plans for the conduct of plant operations (including the organizational structure and the general qualifications of operating and technical support personnel), the plans for industrial security, and the plans for emergency actions to be taken in the unlikely event of an accident that might affect the general public to determine that the applicant is technically qualified to operate the facility safely.

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

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

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

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

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

Table 1.1 (Continued)

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

Table 1.1 (Continued)

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

Table 1.2 Comparison of principal design features of Millstone Unit 3 and other facilities

Design feature	Millstone 3	Comanche Peak	SNUPPS
Containment type*	S	A	A
Rated thermal power, MWt	3411	3411	3411
Gross electrical output, MWe	1156	1159	1188
Total steam flow, 106 lb/hr	15.05	15.14	15.14
Total core flow rate, lb/hr	140.8	140.3	132.7
Nominal system pressure	2250	2250	2250
Fuel lattice	17 x 17	17 x 17	17 x 17
Number of fuel assemblies	193	193	193
Number of fuel rods per fuel assembly	264	264	264
Number of cluster control assemblies full/part length	61/-	53/-	53/-
Reactor vessel inside diameter, in.	173	173	251
overall reactor vessel height, ft-in.	43-10	43-10	43-8
Reactor vessel design pressure, psig	2485	2485	2485
Reactor vessel minimum cladding thickness, in.	0.125	0.125	0.125
Number of loops	4	4	4

Table 1.2 (Continued)

Design feature	Millstone 3	Comanche Peak	SNUPPS
Number of high-pressure safety injection pumps	3	2	2
Number of intermediate safety injection pumps	2	2	2
Number of low-pressure safety injection pumps	2	2	2
Maximum heat flux, Btu/ft^2/hr	440,300	440,300	440,300
Peak linear power for normal operation kW/ft	12.6	12.6	12.6
Maximum centerline fuel temperature, °F	3435	3435	3430
Minimum DNBR	>1.30	>1.30	>1.30
Total peaking factor	2.32	2.32	2.32

*A = atmospheric; S = subatmospheric.

Table 1.3 Listing of outstanding items

Item		SER section
(1)	Internally generated missiles	3.5.1
(2)	Diesel generators	3.5.2, 8.3.1.11, 9.5.4-9.5.8
(3)	Protection against postulated pipe breaks outside containment	3.6.1
(4)	Loading combinations	3.9.3
(5)	Design and construction of component supports	3.9.3
(6)	Inservice testing of pumps and valves	3.9.6
(7)	Equipment qualification	3.10.1, 3.10.2, 3.11.3
(8)	Flow measurement capability	4.4.4
(9)	Loose parts detection program	4.4.5
(10)	Subcompartment analysis	6.2
(11)	Mass and energy release analysis	6.2
(12)	Volumetric inspection of Class 2 components	6.6
(13)	Power-operated relief valve and block valve, compliance with NUREG-0737	8.3.3.4
(14)	Fire protection	9.5.1
(15)	Functional capability of ac and dc emergency lighting	9.5.3
(16)	Shift technical advisor training program and operating experience for startup	13.1.2, 13.2.2
(17)	Emergency Plan	13.3
(18)	Limitation on overtime	13.5.1
(19)	Q list	17

Table 1.4 Listing of confirmatory items

Item		SER section
(1)	Plant's seismic capability beyond design basis	2.5.2
(2)	Dynamic loading	2.5.4.3.2
(3)	Liquefaction potential	2.5.4.4
(4)	Shoreline slope	2.5.5.1
(5)	Turbine maintenance program	3.5.1.3
(6)	Barrier design procedures	3.5.3
(7)	Inservice examination of all pipe welds in break exclusion area	3.6.2
(8)	Jet impingement effects	3.6.2
(9)	Ultimate capacity of containment	3.8.1
(10)	Design of spent fuel racks	3.8.4
(11)	Program evaluation related to TMI Item II.D.1	3.9.3.2
(12)	Predicted cladding collapse time	4.2.3.2
(13)	Fuel assembly mechanical response	4.2.3.3
(14)	Margins itemized in WCAP-8691	4.4.4.1
(15)	Thermal-hydraulic analyses to support N-1 loop operation	4.4.7
(16)	Control rod drive structural materials	4.5.1
(17)	ASME Code cases for Section III, Class I, components	5.2.1.2
(18)	Yield strength of austenitic stainless steels in reactor coolant pressure boundary	5.2.3
(19)	Onsite demonstration of ultrasonic inspection	5.2.4.3
(20)	Preservice inspection program review and relief requests	5.2.4.3
(21)	Preservice and inservice inspection of steam generators	5.4.2.2
(22)	Containment liner weld channel venting	6.2
(23)	Maximum external differential pressure on containment	6.2

Table 1.4 (Continued)

Item		SER section
(24)	Minimum containment pressure for emergency core cooling system	6.2
(25)	Procedures for actuating hydrogen recombiner	6.2
(26)	Secondary enclosure building	6.2
(27)	Sump flow approach velocity	6.2
(28)	Compliance with GDC 51	6.2.7
(29)	Cable separation in nuclear steam supply system process cabinets	7.2.2.1
(30)	Design modification for automatic reactor trip using shunt coil trip attachment	7.2.2.4
(31)	Reactor coolant pump underspeed trip	7.2.2.6
(32)	Conformance with Branch Technical Position ICSB-26	7.2.2.7
(33)	Test of engineered safeguard P-4 interlock	7.3.3.2
(34)	Steam generator level control and protection	7.3.3.4
(35)	Confirmatory test related to IE Bulletin 80-06	7.3.3.5
(36)	Control building isolation reset	7.3.3.8
(37)	Power lockout feature for motor-operated valves	7.3.3.9
(38)	Failure mode and effects analyses of engineered safety features actuation system	7.3.3.10
(39)	Non-Class 1E control signals to Class 1E control circuits	7.3.3.11
(40)	Sequencer deficiency report	7.3.3.13
(41)	Balance-of-plant instrumentation and control system testing capability	7.3.3.14
(42)	Instrument accuracy related to Positions (4), (5), and (6), NUREG-0737, Item II.F.1	7.5.2.4
(43)	Description and analysis demonstrating compliance with GDC 5	8.2.1.1
(44)	Physical separation of offsite circuits within a common right of way	8.2.2.1

Table 1.4 (Continued)

Item		SER section
(45)	Physical separation of offsite circuits between switchyard and Class 1E system	8.2.2.2
(46)	Generation rejection scheme	8.2.2.5
(47)	Description and analysis demonstrating compliance with GDC 17	8.2.2.6
(48)	Description and analysis demonstrating compliance with GDC 18	8.2.3.1
(49)	Positive statement of compliance with BTP PSB-1	8.3.1.2
(50)	Compliance with Position 1 of BTP PSB-1	8.3.1.3
(51)	Adequacy of station electric distribution system voltage	8.3.1.5
(52)	Routing of power cables in the cable spreading area	8.3.3.3.3
(53)	Battery charger and transformer used as isolation devices	8.3.3.10
(54)	Design criteria of associated circuits from isolation device to load	8.3.3.3.16
(55)	Core damage procedure (II.B.3, Criterion 2)	9.3.2
(56)	Control of concrete dust	9.5.4.1
(57)	Qualification of engine-mounted control panels	9.5.4.1
(58)	7-day fuel oil of storage for each diesel generator	9.5.4.2
(59)	Airborne radioactivity monitoring	10.4.2, 10.4.3
(60)	Process control program for solidification of wet wastes	11.4.2
(61)	Task Action Plan Item II.F.1.1	11.5
(62)	Task Action Plan Item I.C.1 - procedures generation package nuclear steam supply system	13.5.2
(63)	Physical Security Plan	13.6
(64)	Initial test program	14
(65)	Reactor coolant pump trip during loss-of-coolant accident	15

Table 1.4 (Continued)

Item		SER section
(66)	Task Action Plan Item III.D.1.1	15
(67)	Analysis of dropped control rod	15.4.3
(68)	Steam generator tube rupture	15
(69)	No failure in emergency core cooling system (ECCS) is not most limiting case in evaluating ECCS	15
(70)	QA program commitments	17.4

Table 1.5 Listing of license conditions

License condition	SER section
(1) Instrumentation for monitoring postaccident conditions, RG 1.97, Rev. 2, requirements	7.5.2.6
(2) Compliance with NUREG-0612 ("Heavy Load Handling")	9.1.5
(3) Installation of postaccident sampling system	9.3.2
(4) Sediment control during fuel oil storage tank refill	9.5.4.2
(5) Moisture in air start system	9.5.6
(6) Preheating of rocker arm lubrication oil system	9.5.7
(7) Blockage of access hatch in diesel generator exhaust system	9.5.8

Table 1.6 Major licensing, construction, and operation milestones

Milestone	Date
Limited work authorization (LWA) issued	May 1, 1974
Site work commenced	June 5, 1974
Construction permit issued	August 9, 1974
Estimated commercial operation date changed from November 1979 to 1982 (applicant)*	December 1975
Estimated commercial operation date changed from 1982 to May 1, 1986 (applicant)*	October 1977
Safety Analysis Report docketed	February 2, 1983
Safety Evaluation Report issued	July 1984
ACRS full committee meeting	September 1984**
Safety hearings	None
Ready for fuel loading (applicant)	November 1985*

*Announced delays were a result of applicant's inability to raise the necessary capital in the required time frame to maintain the construction schedule.
**Estimated.

2 SITE CHARACTERISTICS

2.1 Geography and Demography

Millstone Nuclear Power Station Unit 3 was reviewed in accordance with SRP Sections 2.1.1, 2.1.2, and 2.1.3 of the July 1981 edition of the "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants" (SRP) (NUREG-0800).

2.1.1 Site Location

Millstone Unit 3 is located in the town of Waterford, New London County, Connecticut, on the north shore of Long Island Sound. The 202-ha (500-acre) site occupies the tip of Millstone Point between Niantic Bay on the west and Jordan Cove on the east and is situated 5.1 km (3.2 mi) west-southwest of New London and 64 km (40 mi) southeast of Hartford. Figure 2.1 shows the general region of the Millstone Unit 3 site.

The Millstone Unit 3 containment structure is located immediately north of Millstone Units 1 and 2 at 41° 18' 41" north latitude and 72° 10' 06" west longitude. The universal transverse mercator coordinates are (zone 18-9) 4,576,977 m north and 737,033 m east.

2.1.2 Exclusion Area Authority and Control

The applicant has defined the exclusion area as equivalent to the area within the site boundary that is identified in Figure 2.2. The exclusion area is owned by two tenants in common: the Connecticut Light and Power Company and Western Massachusetts Electric Company, except for that portion of land designated for the Millstone Unit 3 site. The site, which is entirely within the exclusion area, is owned by a number of participants in ownership. Under contract to the owners, Northeast Nuclear Energy Company (NNECO), the operating company and lead applicant for all three units at the Millstone site, has the controlling authority for the exclusion area.

The site is traversed from east to west by a ConRail/Amtrak railroad right-of-way. The main line tracks are about 0.72 km (0.45 mi) from the Millstone Unit 3 containment structure. Control of this area is provided by written agreement between the applicant and ConRail/Amtrak. A portion of the exclusion area is leased to Waterford for public recreation and is used primarily for soccer and baseball games. A portion of the exclusion area is located off shore. Control of this area is provided by written agreement between the applicant and the U.S. Coast Guard.

By virtue of ownership, as well as arrangements made with the U.S. Coast Guard and the ConRail/Amtrak Company, the staff concludes that the applicant has the authority to determine all activities within the exclusion area, as required by 10 CFR 100.

2.1.3 Population Distribution

The staff has independently estimated the 1982 residential population in the vicinity of the Millstone Unit 3 site. Its population estimates compare favorably with the applicant's data, as shown in Table 2.1.

The applicant has chosen a low population zone (LPZ) radius of 3.8 km (2.4 mi). The LPZ is expected to contain approximately 10,700 persons in 1985 at an average density of 236 persons per square kilometer (591 persons per square mile). By the year 2030, the population is projected to increase to a maximum of 16,000 persons at an average density of 353 persons per square kilometer (884 persons per square mile). Seasonal population variations resulting from an influx of summer residents are minimal. Many of the beaches and recreation facilities in the area are used by residents and do not represent any significant increase in population. Industrial transient populations are discussed in Section 2.2.2 of this report.

The town of Waterford, in which Millstone Unit 3 is located, had a 1980 total population of 17,843 (1980 census). New London is the closest population center to Millstone Unit 3 (i.e., a center with more than 25,000 residents, as defined by 10 CFR 100) with a 1980 population of 28,842. The distance between Millstone Unit 3 and New London is about 5.3 km (3.3 mi), which is beyond the minimum distance requirement of 5.1 km (3.2 mi) as set by 10 CFR 100.

2.1.4 Conclusion

On the basis of (1) the 10 CFR 100 definitions of the exclusion area, the low population zone, and the population center distance; (2) the staff analysis of the onsite meteorological data from which the relative concentration factors (χ/Q) were calculated (see Section 2.3 of this report); and (3) the calculated potential radiological dose consequences of design-basis accidents (see Section 15 of this report), the staff concludes that the exclusion area, low population zone, and population center distance meet the criteria of 10 CFR 100 and are acceptable.

2.2 <u>Nearby Industrial, Transportation, and Military Facilities</u>

Millstone Unit 3 was reviewed in accordance with SRP Sections 2.2.1, 2.2.2, 2.2.3, 3.5.1.5, and 3.5.1.6 (NUREG-0800).

2.2.1 Transportation Routes

The nearest major highway that may be used for frequent transportation of hazardous materials is I-95, which is located 6.4 km (4 mi) from the Millstone site (Figure 2.3). Other principal highways that pass near the site include U.S. Route 1, which is located 4.8 km (3 mi) from the site, and State Highway 156, located 2.4 km (1.5 mi) from the site. The separation distances of transportation routes from the plant and the fact that no hazardous materials are transported along State Highway 156 preclude any significant hazards to the plant from either toxic or explosive materials.

As noted in Section 2.1.2, the ConRail/Amtrak railroad traverses the site from east to west. The smallest distance between the mainline tracks and the

Millstone Unit 3 containment structure is 0.72 km (0.45 mi). The applicant has ascertained the hazardous materials and their shipment frequencies for this rail line based on the basis of data obtained from ConRail for the period January 1978 through June 1979 and January 1982 through December 1982. On the basis of the survey data provided, the applicant concluded that the only hazardous material requiring a hazard assessment is liquefied petroleum (propane) gas. Hence, the applicant has analyzed the direct effects of an instant ignition and explosion, as well as one caused by delayed ignition. The staff has reviewed the applicant's analysis and concludes on the basis of the separation distances, size, frequency, and types of cargo shipped on this railroad, that the risk to this plant is below design requirement.

There are no major pipelines within 8 km (5 mi) of the site. The nearest natural gas distribution line is approximately 4.7 km (2.9 mi) from the site, located along Rope Ferry Road in Waterford. This is a 6-in. plastic pipeline carrying natural gas at 30 psi. There is a possibility of extending the gas distribution line along Rope Ferry Road to the intersection of Great Neck Road. This would bring the pipeline to 3.9 km (2.4 mi) from the Millstone site. The staff has evaluated this pipeline and concluded that the present and proposed locations do not represent a hazard to the plant. The staff will require the applicant to keep the staff advised of any future extensions of this pipeline or introduction of new lines within 8 km (5 mi) of the site.

Ships that pass by the site in the shipping channels of Long Island Sound are of two types: general cargo freighters, which usually are partially unloaded with drafts of 20 to 25 ft, and deep draft tankers with drafts of 35 to 38 ft. Both of these classes of ships must remain at least 3.2 km (2 mi) off shore to avoid running aground on Bartlett Reef. On the average of once a month, a barge carrying 15,000 barrels of sulfuric acid is towed past the site outside of Bartlett Reef. No oil barges pass to the shore side of Bartlett Reef. The staff has concluded that the type of materials shipped and the distances maintained between the carriers and the site do not represent a hazard to the plant.

As shown in Figure 2.4, the air lane nearest to the site is V58, which is approximately 6.4 km (4 mi) northeast of the site. Other air lanes in the vicinity include V16, 9.6 km (6 mi) northwest, and V308, 12.9 km (8 mi) east. The nearest high-altitude jet route, J121-581, passes 14.5 km (9 mi) southeast of the site.

On the basis of these transportation route separation distances from Millstone Unit 3 and the nature of the materials transported, the staff concludes that, with the exception of the rail line, traffic along these transportation routes will not adversely affect the safe operation of Millstone Unit 3.

2.2.2 Nearby Facilities

The area around the Millstone site (see Figure 2.1) contains three major industrial facilities (Dow Chemical Corporation, Pfizer Corporation, and Electric Boat Division of General Dynamics Corporation), two transportation facilities (Groton/New London Airport and New London Airport), and four military installations (U.S. Navy Submarine Base, U.S. Coast Guard Academy, Camp O'Neil, and Stone's Ranch Military Reservation).

The Dow Chemical Corporation of Allen Point, Ledyard, Connecticut, is located on the east bank of the Thames River approximately 16.1 km (10 mi) north-northeast of the site. Dow Chemical is a producer of synthetic compounds and employs approximately 160 persons. Dow Chemical produces inorganic compounds, such as Styron, Styrofoam, and a base product of latex paints. All materials are moved to and from the company by truck and/or railroad.

The Pfizer Corporation of Eastern Point Road, Groton, Connecticut, is located on the east bank of the Thames River, approximately 8 km (5 mi) east-northeast of the site. The Pfizer Corporation is a producer of pharmaceutical and medical supplies, employing approximately 3,000 persons. Pfizer Corporation produces organic compounds and pharmaceutical materials, such as citric acid, antibiotics, synthetic medicines, vitamins, and caffeine. All materials are moved to and from the Pfizer Corporation by truck and/or railroad.

The Electric Boat Division of General Dynamics of Eastern Point Road, Groton, Connecticut, is located 8.9 km (5.5 mi) east-northeast of the site. The Electric Boat Division employs approximately 20,000 persons, and is a producer of submarines and oceanographic equipment for the commercial industry and the U.S. Navy. The nature of products produced at Electric Boat requires that they handle substantial amounts of nuclear materials that are licensed under the Naval Reactors Division. All material is moved by truck, railroad, and/or barge to and from the company with the exception of completed ships that are launched.

The New London Airport located approximately 6.4 km (4 mi) north-northeast of the site is limited to handling small, private aircraft. Seven persons are employed at the airport on a full-time basis. Approximately 12 additional part-time persons may be employed at this airport, primarily during the summer months and on weekends. Scheduled commercial aircraft do not use this airport. The maximum runway length is 2,000 ft. The largest aircraft known to use this airport on a regular basis is a Piper Aztec, with a gross weight of 5,200 lb.

The Groton/New London Airport, located 11.3 km (7 mi) east-northeast of the site, handles regularly scheduled commercial passenger flights. Two hundred persons are employed at Groton/New London Airport on a full-time basis. The airport has three runways. These range from 3,000 to 5,000 ft. The largest commercial aircraft to use this airport on a regular basis is a Fokker-F27, with a gross weight of 45,200 lb, plus a fuel load of 11,000 lb.

The U.S. Navy Submarine Base, Groton, Connecticut, is located on the east bank of the Thames River, 11.3 km (7 mi) northeast of the site. There are about 14,000 military and civilian personnel stationed on or near this base. The U.S. Navy Submarine Base provides logistics as well as training and operation of the base and its ships (nuclear and non-nuclear). All materials are moved by truck, railroad, barge, and/or ship to and from this facility.

The U.S. Coast Guard Academy, New London, Connecticut, is located on the west bank of the Thames River, approximately 9.7 km (6 mi) northeast of the site. Over 900 cadets attend the academy, whereas approximately 500 military and 150 civilian personnel are employed here. All materials used at the academy are nonhazardous and are moved by truck.

Camp O'Neill, located approximately 3.2 km (2 mi) northwest of the site, is a training headquarters for the Connecticut Army National Guard. It is owned and operated by the Military Department of the State of Connecticut. On a full-time basis, it employs 24 persons (military and civilian) including the headquarters personnel for the Connecticut Military Academy, Post Operations personnel, and the 745th Signal Company. On a part-time basis, during various weekends, Camp O'Neill is occupied by varying numbers of troop units for administrative training, billeting, and supply functions for the Connecticut Army National Guard.

In addition to Camp O'Neill, the Military Department of the State of Connecticut also maintains a field training facility known as Stone's Ranch Military Reservation, located 11.3 km (7 mi) northwest of the site. Nineteen persons are employed here full time for two regional motor vehicle and equipment maintenance shops. It is also occupied on a part-time basis by varying numbers of troop units for periods of field training for the Connecticut Army National Guard. No significant ordnance is stored or used at this facility.

On the basis of the separation distances and the type of activities conducted at the above facilities, the staff concludes that these activities do not represent a hazard to the safe operation of the plant.

2.2.3 Conclusion

The staff has conducted its review based on the criteria given in 10 CFR 50, Appendix A, General Design Criterion (GDC) 4, and in SRP Section 2.2.3. The staff concludes that the plant is adequately protected and can be operated with an acceptable degree of safety as a result of activities at nearby transportation, industrial, and military facilities.

2.3 Meteorology

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

2.3.1 Regional Climatology

Cold air moving southeastward into the area of the Millstone site is modified by Long Island Sound south of the plant. The Atlantic Ocean to the east also has a moderating effect on climate. Continental air masses dominate the region in winter and alternate with maritime tropical air masses in summer. The mean annual temperature in the area is about 52°F, ranging from about 30°F in January to about 74°F in July. Annual precipitation in the area is about 39 in.

The site lies near a principal track of storms that move northeast along the Atlantic coast and result in a variety of severe weather phenomena that affect the site area. Thunderstorms can be expected on about 22 days each year. About 60% of these thunderstorms occur between the months of June and August. Considering the frequency of thunderstorms, the applicant has estimated the number of lightning strikes to the ground per year at the Millstone site to be about two per year. Hail often accompanies severe thunderstorms.

During the period 1955 through 1967, an average 1.4 occurrences per year of hail with diameters 19 mm (3/4 in.) or greater were reported in the one-degree latitude-longitude square containing the site.

Tornados are uncommon in the region. For the two-degree latitude-longitude "square" (14,125 mi^2) containing the site, an average of about two tornados per year were reported for the period 1954 through 1981. On the basis of calculated expected mean tornado path area of 0.18 mi^2, the computed probability of occurrence for a tornado at the plant site is about 3.2×10^{-5} per year. The applicant has computed a higher probability of occurrence ($\sim 5.5 \times 10^{-4}$ per year) based on a larger tornado path area (2.82 mi^2) and a smaller annual frequency (0.7 tornado per year). The characteristics of the design-basis tornado considered by the applicant for the plant are based on the recommendations of Regulatory Guide (RG) 1.76, "Design Basis Tornado for Nuclear Power Plants," for this region of the country. The applicant's design-basis tornado has a 360-mph rotational velocity, with a translational velocity of 60 mph, a total pressure drop of 3 psi, and a rate of pressure drop of 2 psi/sec.

Hurricanes occasionally track northward along the Atlantic coast. In the period 1871 through 1981, about 15 tropical depressions, tropical storms, and hurricanes have passed within about 50 mi of the plant. Wind speeds associated with these storm systems are usually highest along the coast, with wind speeds diminishing inland.

High wind speed occurrences in the area are associated with severe thunderstorms, extratropical cyclones, tropical storms, and hurricanes. The highest "fastest mile" wind speed reported at Bridgeport, Connecticut, was 67 mph in January 1964. For plant design, the applicant has selected an operating-basis wind speed to be 82 mph. The operating-basis wind speed is defined as the "fastest mile" wind speed at a height of 30 ft with a return period of 100 years.

Heavy snowfall is not uncommon in the region, and roof loads may accumulate as a result of wintertime precipitation mixture of snow, ice, and rain. The maximum monthly snowfall and the maximum snowfall in a 24-hour period observed in Bridgeport, Connecticut, were 74 in. and 16.7 in., respectively, during the month of February. Ice storms, which may disrupt offsite power, are relatively infrequent. The applicant estimates that ice or freezing rain may occur about one time per year in the Millstone region, with a glaze accumulation of 0.25 in. Amounts greater than 0.5 in. can be expected about every 2 years. The applicant has estimated the weight on the ground of the 100-year return period snowpack to be 31 psf. To determine the probable maximum snowload for consideration in the design of safety-related structures, the applicant has added 29 psf, the weight of the 48-hour probable maximum winter precipitation, to the weight of the 100-year return snowpack for a total weight of 60 psf.

Occasional large-scale episodes of atmospheric stagnation occur in the region. From 1960 through 1970, 10 atmospheric high pollution potential periods were identified in the area.

As discussed above, the staff has reviewed available information relative to the regional meteorological conditions of importance to the safe design and siting of this plant, in accordance with SRP Section 2.3.1. On the basis of this review, the staff concludes that the applicant has identified appropriate regional meteorological conditions for consideration in the design and siting of this plant and has met the requirements of 10 CFR 100.10 and GDC 2 and 4.

2.3.2 Local Meteorology

Climatological data from Bridgeport, Connecticut, and available onsite data have been used to assess meterological characteristics of the plant site.

Extreme temperatures of -10°F to 103°F have been measured during 1901 through 1981 at Bridgeport, Connecticut. Onsite temperature extremes have ranged from -4.9°F to 88.7°F during the period of January 1974 through December 1981.

Precipitation is well distributed throughout the year, ranging from about 2.6 in. in June to almost 4 in. in November. The maximum amount of precipitation in a 24-hour period at Bridgeport, Connecticut, was 6.89 in.

Long Island Sound, adjacent to the plant site, will be used as the plant ultimate heat sink and is acceptable in accordance with the requirements of RG 1.27.

The wind distribution in the area, as determined on site, has an occurrence of 42% from a northwesterly quadrant direction with about 21% from the southwesterly direction. Remaining winds are distributed fairly uniformly in the remaining directions.

As discussed above, the staff has reviewed available information relative to local meteorological conditions of importance to the safe design and siting of this plant in accordance with SRP Section 2.3.2. The staff concludes that the applicant has identified and considered appropriate local meteorological conditions in design and siting and, therefore, meets the requirements of 10 CFR 100.10 and GDC 2.

2.3.3 Onsite Meteorological Measurements Program

Onsite meteorological measurements are made on a 465-ft tower situated south-southeast of the plant located nearly 1/4 mi away from the main plant structure in proximity to the Long Island Sound shore. Measurements of wind speed, wind direction, vertical temperature difference, ambient temperature, and dew point temperature are all made on the tower. Visibility and solar radiation measurements are made near ground level at the tower base. The elevations of the instruments are shown in Table 2.2.

The original onsite meteorological measurements program at Millstone began in 1965 and has continued in support of Units 1 and 2. The program has been upgraded to conform to RG 1.23 and will continue as the operational program.

The data being collected are recorded on strip chart recorders in the instrument shed near the tower, as well as in the control room. In addition, the information is recorded on magnetic tape for use on the plant microcomputers and the larger corporate computer system.

Onsite data collected from January 1981 through December 1982 had a joint data recovery rate of over 90% for wind speed and wind direction, measured at the 10-m level, and temperature difference between 10 and 43 m. These data were used to evaluate both short- and long-term gaseous dispersion as described in Sections 2.3.4 and 2.3.5 of this report.

The meteorological data also will be available to the emergency operations facility and technical support center for use in the dose calculation model following an accidental release at the plant.

The staff has reviewed the onsite meteorological measurements system in accordance with SRP Section 2.3.3 and concludes that the current meteorological measurements program has provided data to represent onsite meteorological conditions as required in 10 CFR 100.10 and 10 CFR 50, Appendix I. The staff concludes that the historical site data provide a reasonable basis for making conservative estimates of atmospheric dispersion conditions for estimating consequences of design-basis-accident and routine releases from the plant.

2.3.4 Short-Term (Accident) Diffusion Estimates

Short-term accidental releases lasting up to 30 days were evaluated at the exclusion area and low population zone (LPZ) boundaries. The relative concentration (χ/Q) values assuming a ground-level release were determined on the basis of the onsite meteorological data in joint frequency form from January 1, 1981, through December 31, 1982. The data from observations of wind speed and wind direction at the 10-m tower levels were combined with the atmospheric stability that was based on the vertical temperature difference between the 10- and 43-m tower levels. These data were used in the direction-dependent model described in RGs 1.145 and 1.111. The methods described in NUREG/CR-2858 and NUREG/CR-2919 were followed. The model includes credit for building wake effects, effluent plume meander, and recirculation. The maximum 0-to-2-hour χ/Q value at the exclusion area boundary, which is expected to be exceeded less than 0.5% of the time, is 5.3×10^{-4} sec/m^3 at the 525-m boundary south-southwest of the plant. At the LPZ boundary distance of 3,862 m, the χ/Q values for time periods to 30 days following an accidental release are given in Table 2.3. These χ/Q values were determined for the area south-southwest of the plant, which is an offshore direction.

This independent staff analysis provides a conservative assessment of short-term gaseous releases and confirms the results presented by the applicant. The staff has confirmed that the applicant has used adequately conservative diffusion estimates to assess the consequences of radioactive releases for design-basis accidents and has demonstrated compliance with the guideline dose levels for 10 CFR 100.

2.3.5 Long-Term (Routine) Diffusion Estimates

To audit the applicant's estimates, the staff has performed an independent calculation of annual average relative concentration (χ/Q) and relative deposition (D/Q) values.

Annual average χ/Q and D/Q values at specific receptor points were calculated using the straight-line Gaussian atmospheric dispersion model, corrected for plume recirculation, described in RG 1.111 and NUREG/CR-2919. A continuous ground-level release was assumed in these calculations.

In addition, χ/Q and D/Q values were determined to calculate the population dose assessment to 50 mi from the plant in each of the sixteen 22-1/2 degree sectors around the plant. Maximum values at nearby receptors are given in Table 2.4. The onsite meteorological data described in Section 2.3.4 was used for this evaluation.

The staff values for the annual average, which were slightly more conservative than the applicant's values, are used to determine the dose resulting from normal operational releases.

On the basis of the above evaluation performed in accordance with SRP Section 2.3.5, the staff concludes that the applicant has considered representative atmospheric dispersion estimates and has demonstrated compliance with the numerical guides for doses in 10 CFR 50, Appendix I. The atmospheric dispersion estimates developed by the staff have been used in its assessment of normal operational releases.

2.4 <u>Hydrologic Engineering</u>

The staff has reviewed the hydrologic engineering aspects of the applicant's design, design criteria, and design bases of safety-related facilities at the Millstone Unit 3 station. The acceptance criteria include the GDC, the reactor site criteria (10 CFR 100), and standards for protection against radiation (10 CFR 20, Appendix B, Table II). Guidelines for implementation of the requirements of the acceptance criteria are provided in RGs, American National Standards Institute (ANSI) standards, and branch technical positions (BTPs) identified in SRP Sections 2.4.1 through 2.4.14. Conformance to the acceptance criteria provides the bases for concluding that the site and facilities meet the requirements of 10 CFR 20, 50, and 100 with respect to hydrologic engineering.

2.4.1 Hydrologic Description

The Millstone site is located in the town of Waterford, New London County, Connecticut, on the north shore of Long Island Sound. Millstone Unit 3 shares the site with two other nuclear units located on the tip of Millstone Point. The site is located between Niantic Bay on the west and Jordan Cove on the east and is situated 3.2 mi west-southwest of the city limits of New London and 40 mi southeast of Hartford. The ground elevation at the site ranges from sea level to 40 ft above mean sea level (MSL). Mean high tide is about 1.3 MSL. Unit 3 plant grade is at el 24 ft MSL. Except for the two watertight doors into the service water pump cubicles located in the circulating and service water pumphouse (intake structure), all accesses to safety-related structures and facilities are at or above el 24 ft 6 in. MSL or 6 in. above plant grade. Significant hydrologically related plant features include the intake structure and adjacent shore protective structures located at Niantic Bay. Surface drainage from the plant, yard, roofs, and catch basins flows into underground stormwater conduits or surface channels that discharge into Niantic Bay.

There are no perennial streams on or adjacent to the site. Precipitation falling on the site is conveyed to Long Island Sound by surface runoff and groundwater flow. Normal variations in the water levels of Long Island Sound and at the shores of the plant site are induced primarily by semidiurnal tides. Extreme variations in water levels are storm induced and result from tropical windstorms (hurricanes) and extratropical windstorms. During the past 45 years, six hurricanes have given rise to abnormally high stillwater levels ranging from 5 ft to approximately 10 ft MSL, not including waves.

Plant operation will utilize once-through cooling, extracting water from Niantic Bay and discharging it into an abandoned quarry that is connected to Long Island Sound.

The applicant has provided hydrologic descriptions of the site. The staff has reviewed the applicant's information in accordance with procedures in SRP Sections 2.4.1 and 2.4.2 and concludes that the general hydrologic descriptions of the site meet the applicable requirements of GDC 2 and 10 CFR 100.

2.4.2 Floods

2.4.2.1 Flood Potential

Flooding near the site has historically been caused by hurricanes. The applicant reported that the maximum historical flooding was the result of a hurricane on September 21, 1938, which produced a flood level elevation of 9.7 ft MSL at New London, Connecticut.

The applicant considered several flooding sources in establishing the flooding design basis for the site. These events include stream flooding, precipitationinduced flooding, flooding caused by seismically induced dam failure, ice flooding, tsunami-induced flooding, and surge and seiche flooding. The applicant states that the only sources of flooding that could affect Millstone Unit 3 are direct rainfall and storm surges and that the controlling event for flooding at the site is the result of a storm surge induced by the probable maximum hurricane (PMH). The staff concurs that the PMH and the local probable maximum precipitation (PMP) are the appropriate design-basis events for this site.

2.4.2.2 Intense Local Precipitation

The applicant has indicated that the roofs of safety-related buildings are designed to dispose of local severe precipitation up to and including the local PMP as determined using Hydrometeorological Reports (HMRs) Nos. 51 and 52 (U.S. National Weather Service, 1978 and 1982).

These reports indicate that the probable maximum precipitation accumulation for a 1-square-mile area at the Millstone 3 site is as follows:

Duration	Depth (inches)
5 min.	5.9
15 min.	9.2
30 min.	13.2
1 hour	17.4
6 hours	26.0

The applicant states that the parapets on safety-related buildings have been designed either with scuppers or as low curbing so that if internal roof drains become clogged, the accumulated precipitation would overflow the parapets before either exceeding the design-basis roof loading or flooding roof ventilators. The applicant has indicated that the roof ponding level on the control building will exceed the roof hatch seal by about 3 in., but that the seal will be made water-tight to prevent in-leakage.

The staff has analyzed roof drainage systems including the ponding levels on roofs of safety-related structures using the PMP accumulations shown in HMRs 51 and 52. On the basis of its analysis according to SRP Section 2.4.2 and RG 1.102, the staff concludes that subject to the installation of a water-tight seal for the roof hatch of the control building, the plant meets the requirements of GDC 2 with respect to the effects of local intense precipitation on the roof.

The applicant evaluated the effect of local intense precipitation of the magnitude of a PMP concentrated on the plant site using HMRs 51 and 52, assuming complete blockage of the underground storm drainage system. The applicant concluded that the site grading would limit ponding of water to a level less than el 24.5 ft MSL, which is the minimum elevation of entrances to safety-related buildings except at three normally closed and locked outward-swinging doors. One door is located in the hydrogen recombiner building, the second door is in the main steam valve building, and the third door is to the auxiliary building. Analysis has shown that the potential in-leakage around the door seals and the resultant water level within the hydrogen recombiner and main steam valve buildings would be substantially lower than the base of any safety-related equipment. The applicant plans to install a protective curb outside the auxiliary building door to prevent any potential in-leakage.

The staff has reviewed the applicant's analysis, which used the most recent PMP guidance, and concludes that subject to installation of the protective curbing at the auxiliary building door, the plant meets the requirements of GDC 2 with respect to the effects of local intense precipitation runoff on site drainage relative to flooding of safety-related structures.

2.4.3 Probable Maximum Flood on Streams and Rivers

There are no major rivers or streams in the vicinity of Millstone Point, nor are there any watercourses on the site. A number of small brooks flow into Jordan Cove, east of the site, and into the Niantic River and then to Niantic Bay, west of the site. Any flooding of these brooks, even as a result of the PMP, would not significantly raise the water levels in Niantic Bay, Jordan

Cove, or Long Island Sound in the vicinity of the site. Additionally, local topography precludes flooding of any portion of the site from these landward areas.

The staff has visited the site and reviewed the FSAR material in accordance with the procedures described in SRP Sections 2.4.2 and 2.4.3 and concludes that the plant meets the requirements of GDC 2 with respect to precipitation-induced flooding on streams and rivers.

2.4.4 Potential Dam Failures

Because there are no major rivers or streams in the vicinity of Millstone Point, the applicant concludes that there is no dam failure hazard to the safety-related facilities of the plant. Using the procedures described in SRP Section 2.4.4, the staff has reviewed the applicant's submittal and concludes that the plant meets the requirements of GDC 2 with respect to hydrologic aspects of dam failures.

2.4.5 Probable Maximum Surge and Seiche Flooding

Major historical surge flooding on the southern New England coast has been induced by both hurricanes and northeasters. The applicant analyzed the potential surge flooding from a PMH at the CP stage and concluded that the PMH would produce the design-basis water level at the plant site. The parameters for the PMH were obtained from "Interim Report - Meteorological Characteristics of the Probable Maximum Hurricane, Atlantic and Gulf Coasts of the United States" (HUR 7-97) (National Oceanic and Atmospheric Administration, 1968). That reference provides a range of characteristics to define the size (radius to maximum winds) and speed (forward translation) of the hurricane. The PMH results from the most critical combination of these and other characteristics, including a postulated critical track to the site. Characteristics identifying the PMH for the site are a central pressure index (CPI) of 27.26 in. of mercury, asymptotic pressure of 30.56 in. of mercury, radius of maximum winds of 48 nautical miles, forward speed of translation of 15 knots, and maximum 10-min sustained gradient wind speed of 124 mph. The applicant also assumed a 10% exceedance high spring astronomical tide of 2.4 ft MSL. The applicant's coastal surge estimate also includes consideration of an initial rise of 1.0 ft (sea level anomaly). During the CP review, the staff concluded that these PMH parameters were conservative and provide acceptable values for determining the design-basis flood.

The applicant calculated that the design-basis open coast surge near the site had a stillwater level of 19.7 ft MSL. The staff concurred with this estimate on the basis of an independent assessment conducted at the CP stage of review. The staff has reviewed the FSAR in accordance with SRP Section 2.4.5 and concludes that the estimate is in accordance with RG 1.59.

The applicant has calculated the wave heights and wave runup at the site coincident with the PMH surge level. These calculations were based on the consideration of both modified deepwater waves entering Long Island Sound from the Atlantic Ocean and shallow-water waves that are generated within Long Island Sound. As the surge level would begin to rise, resulting from the approaching eye of the postulated hurricane, the increasing wind speed would progressively change direction from the north through east to the south. The deepwater waves would be refracted around the tip of Millstone Point and reduced in height in the area of the intake structure. Shallow-water waves from the southwest would impact and run up on the intake structure without any reduction in height from refraction.

As the surge reaches the maximum elevation of 19.7 ft MSL, waves in the order of 16 ft would be approaching the area of the intake structure. The areas adjacent to the intake structure are protected by a vertical concrete seawall to the east and a quarrystone revetment to the west. However, during the postulated design flooding event, both the surge and waves will overtop these structures. The resulting wave runup on the slopes fronting the site will not exceed plant grade (el 24.0 ft MSL). The water depths at the intake structure allow only nonbreaking waves (clapotis) to impinge on the front face of the intake structure. The maximum wave runup on the intake structure would be to el 42.4 ft MSL.

As indicated in Section 2.4.1, except for the two watertight doors (at el 14.5 ft MSL) into the service water pump cubicles located within the intake structure, all accesses to safety-related structures are at or above el 24 ft 6 in. MSL 6 in. above plant grade). The applicant has indicated that the watertight service water pump cubicles are flood protected to el 25.5 ft MSL and that if the water level were to exceed this flood protection level, any potential in-leakage would be directed to the service water cubicle sump pump and discharge outside the cubicle into the circulating water pumphouse. As indicated in Section 2.4.14, the applicant will be required to ensure that the watertight doors into the service water cubicles are closed and secured during any hydrologic event that will result in water levels in excess of el 14.5 ft MLS.

On the basis of its analyses according to SRP Sections 2.4.2, 2.4.5, and 2.4.10, the staff concludes that the plant design for PMH storm surge flooding and associated wave overtopping and runup meets the guidelines of RGs 1.59 and 1.102 and the requirements of GDC 2.

2.4.6 Probable Maximum Tsunami Flooding

The applicant has stated that the occurrence of tsunamis on the Atlantic coast has an extremely low probability. The staff concludes that although tsunamis may occur, they will not exceed the design-basis water level associated with the PMH.

On the basis of its analysis according to SRP Section 2.4.6, the staff concludes that the occurrence of a tsunami will not adversely affect plant operation and the plant meets the guidelines of RG 1.102 and the requirements of GDC 2 with respect to tsunami flooding.

2.4.7 Ice Effects

The applicant states that there is no history of ice in Niantic Bay or ice jam formation in the vicinity of the circulating and service water pumphouse. Additionally, no credible ice blockage of waterways could produce water levels in excess of the design-basis water level (from the PMH) of 19.7 ft MSL.

The applicant considers it highly unlikely that ice would form or collect in a manner or amount sufficient to obstruct the flow to safety-related pumps. However, as a preventive measure, part of the condenser discharge water will be recirculated into the mixing box above the intake and distributed into the pump bays.

A reinforced concrete curtain wall located at the seaward face of the pumphouse and extending to -7.0 ft MSL (extreme predicted astronomical low tide is -2.3 ft MSL) precludes floating or partially submerged ice from entering the pump intake bays and damaging or blocking the bar racks.

On the basis of its review of the FSAR using the procedures in SRP Section 2.4.7, the staff concludes that the effects of ice at the intake structure would not adversely affect plant operation, and, therefore, the plant meets the guidelines of RG 1.102 and the requirements of GDC 2 with respect to ice effects.

2.4.8 Cooling Water Canals and Reservoirs

There are no canals or reservoirs used to transport or impound plant cooling water. A review under the procedures of SRP Section 2.4.8 is not applicable to this plant.

2.4.9 Channel Diversions

There are no natural stream channel diversions to adversely affect the plant and essential water supplies. A review under the procedures of SRP Section 2.4.9 is not applicable to this plant.

2.4.10 Flood Protection Requirements

The flood design bases for the site consist of intense local precipitation (PMP) and the PMH-induced stillwater level with coincident waves, wave runup, and wave overtopping.

Flood protection from the design-basis flood and associated wave runup is provided by vertical concrete seawalls west and south of the intake structure. The vertical seawalls would be subjected to both surge and wave overtopping during this event. The maximum water level resulting from the PMH would be below the levels to which the plant is flood protected.

As discussed in Section 2.4.2.2, the staff concludes that the plant meets the requirements of GDC 2 with respect to the effects of flooding of safety-related facilities resulting from local intense precipitation runoff from site drainage.

On the basis of its analysis using SRP Sections 2.4.2, 2.4.5, and 2.4.10, the staff concludes that the applicant has shown that, except for the effects of flooding on roofs, the plant design flood protection meets the guidelines of RG 1.102 and the requirements of GDC 2.

2.4.11 Cooling Water Supply

2.4.11.1 Description of Cooling Water Supply

The condenser cooling water and service water are supplied to the plant from Long Island Sound (the ultimate heat sink) through the intake structure located on the shoreline of Niantic Bay adjacent to the plant. The normal once-through circulating water system provides cooling water at a flow rate of 912,000 gpm. For normal full-power operation, service cooling water is supplied at the rate of 29,410 gpm. A minimum safety-related cooling water flow required during accident conditions is 16,050 gpm.

The service water pumps are located in the intake structure and are housed in watertight compartments to provide protection against the design-basis flood.

2.4.11.2 Adequacy of Cooling Water Supply

The applicant has analyzed the ability of the cooling water supply to withstand the effect of severe natural phenomena. As indicated in Section 2.4.7, the effects of ice blockage would not obstruct the flow to safety-related pumps. Thus, the staff concludes that the intake structure and essential service water flow are adequately protected against ice effects.

On the basis of records collected since 1956, the applicant has indicated that the maximum daily average water temperature at the intake structure is 75°F and that all components cooled by the service water system are designed to perform on the basis of the maximum intake service water temperature of 75°F. Historical data show that the intake water temperature at Millstone Unit 3 has risen as much as 8F° during a 24-hour perod and that the maximum daily temperature has reached 78°F. A Technical Specification will be required by the staff to monitor the intake water temperature at 6-hour intervals when the intake water temperature exceeds 70°F.

The applicant reported that the minimum historical water level at New London, Connecticut, is -4.8 ft MSL. The maximum as a result of the PMH wind field blowing off shore was estimated to be 3.7 ft and was assumed to coincide with a 10% exceedance low spring astronomical tide of -2.2 ft MSL. The resultant water level would be -5.9 ft MSL. This is 2.1 ft above the minimum water level (el -8.0 ft MSL) required for service water pumps.

On the basis of its evaluation, the staff concludes the ultimate heat sink (UHS) can supply cooling water that meets the requirements of RG 1.27, GDC 2, 10 CFR 100, and Appendix A to 10 CFR 100, with respect to hydrologic characteristics.

2.4.12 Groundwater

The water table aquifer in the plant area lies in the overburden, which consists of varying thicknesses of both ablation and basal tills with occasional permeable lenses of sand. Below these tills is a hard crystalline bedrock with tight, moderately spaced joints. In the area of the emergency generator enclosure and the control building, the surface of the basal till is about el 13 ft (9 ft below plant grade); while the bedrock is about el -7 ft (33 ft below plant grade). In the area of the intake structure, bedrock is at about el -40 ft MSL. Both the basal till and overlying ablation till are relatively impervious, with the ablation tills more pervious than the basal tills. Very little inflow of water was observed entering the excavations through the bedrock. This would indicate that the permeability of the bedrock is very low and that very little groundwater or seawater seeps through the site bedrock.

The prevalence of bedrock outcrops at higher elevations approximately 8,000 ft north and 3,500 ft northeast of the site indicates that the bedrock acts as a groundwater divide, isolating the overburden adjacent to the plant from soils further inland. The groundwater level is subject to considerable seasonal fluctuations. The recharging of the groundwater would primarily be due to infiltration of local precipitation with probable migration to the adjacent waters of Long Island Sound. The groundwater surface has a gradient generally sloping from northeast to southwest. Essentially all of the groundwater movement is restricted to the soil overburden. Groundwater observation near the shoreline exhibited tidal fluctuations suggesting that the occasional sand lenses can be quite permeable.

The applicant has identified three onsite shallow wells all of which are up gradient from the site: one about 6,600 ft to the north-northeast, one about 4,000 ft to the northeast, and the third about 1,800 ft to the northwest. None of these wells provide domestic drinking water; only the well 4,000 ft northeast is used to supply drinking water to a nearby baseball field. The nearest publicly used wells are upgradient and about 2 mi from the site.

The applicant has concluded that since there is no plant use of groundwater, and the plant area is isolated from inland soils, there is no effect on groundwater on the site or surrounding areas.

The applicant has stated that there is no safety-related permanent dewatering system for Millstone Unit 3 and that safety-related structures are designed for groundwater pressure and buoyancy forces consistent with the design groundwater surface levels. These design groundwater levels vary from el 23 ft MSL (1 ft below plant grade) at the auxiliary, fuel, and waste disposal buildings to el 18 ft MSL (6 ft below grade) at the hydrogen recombiner building. On the basis of information provided in the FSAR, the staff believes that the groundwater levels are a conservative basis for design.

The staff has reviewed the material in the FSAR in accordance with procedures described in SRP Section 2.4.12 and has determined that the site does not affect safety of the neighboring groundwater supplies and that emergency shutdown does not depend on groundwater supplies. The staff also concurs that the design-basis groundwater levels for safety-related structures have been determined to be a conservative level. Therefore, the staff concludes that the site meets the requirements of GDC 2, 10 CFR 100, and Appendix A to 10 CFR 100 with respect to groundwater.

2.4.13 Accidental Releases of Liquid Effluents in Groundwater and Surface Water

The accidental release of radioactive liquids to the ground or directly to surface water would not affect drinking water users. Radioactive water that might be accidentally released to the circulating water system would be carried to the quarry where it would be highly mixed and then discharged into Long Island Sound and be carried away by ambient ocean currents.

The applicant has determined that the postulated failure of the boron recovery tank located at plant grade (el 24.0 ft MSL) has the greatest potential to result in high offsite radionuclide concentration. The contents of the tank were conservatively assumed to enter the groundwater instantaneously, as a slug release. For the radionuclides to be released to the groundwater, the building walls containing the tanks would have to be cracked. However, to bound the problem, the staff assumed that the contents of the tank did enter the groundwater system. The nuclides were then assumed to travel by way of the backfilled trench containing the circulating and service water pipelines to the bay with the travel time controlled by the groundwater velocity.

The groundwater gradient for this part of the Millstone Unit 3 site is toward the adjacent saltwater of Niantic Bay and away from areas of groundwater usage. The staff has conservatively estimated the travel time from the site to the waters of the bay to be approximately 6.6 years. For those nuclides that are affected by ion exchange processes the travel time would be longer. Upon release into the tidal waters, all radionuclides would be dispersed and diluted to concentrations well below those listed in 10 CFR 20, Appendix B, Table II (water), by the mixing process associated with the tidal, wind, and wave-induced currents of Long Island Sound.

Therefore, the staff concludes that accidental releases of liquid radioactivity from accidents within the design basis would not pose a threat to public health and safety and that the plant meets the requirements of 10 CFR 100 with respect to potential accidental releases of radioactive effluent. In performing its analysis, the staff relied on the guidance of SRP Sections 2.4.12 and 2.4.13, RG 1.113, and 10 CFR 20 and 100.

2.4.14 Technical Specifications and Emergency Operation Requirements

On the basis of its review in accordance with SRP Section 2.4.14, the staff concludes as indicated in Section 2.4.5 that a Technical Specification or Emergency Operating Plan will be required to ensure that the watertight doors to the service water pump cubicles are closed and secured well in advance of a hydrologic event that is predicted to produce water levels in excess of el 14.5 ft MSL (floor elevation of the intake structure). As discussed in Section 2.4.11.2, the staff also concludes that a Technical Specification will be required to monitor the service water intake temperature.

2.4.15 Conclusions

On the basis of its review and analysis as described above, the staff has asked the applicant to provide additional information on roof drainage as a result of intense local precipitation (PMP). Until the applicant provides this additional information, the staff cannot conclude that the plant meets the requirements of GDC 2 with respect to flooding.

The staff concludes that the plant will not adversely affect groundwater users and any accidental spill will migrate toward the adjacent tidal water areas rather than toward existing wells. Such spills, if allowed to proceed, would be diluted by tidal action in Niantic Bay and Long Island Sound. The staff will require a Technical Specification or Emergency Operating Plan for both the closing of the watertight doors to the service water pump cubicles and the monitoring of the service water intake temperature.

2.5 Geology and Seismology

2.5.1 Basic Geologic and Seismic Information

The geology and seismology of the site were reviewed in detail before (1) the construction permits and operating licenses were issued for Millstone Units 1 and 2, (2) the systematic evaluation program (SEP) for Millstone Unit 1, and (3) the construction permit was issued for Millstone Unit 3. The reviews were performed by the staff of the U.S. Atomic Energy Commission (AEC), the predecessor to the U.S. Nuclear Regulatory Commission (NRC), and its geologic advisors, the U.S. Geological Survey (USGS), and its seismological advisors, the National Oceanic and Atmospheric Administration (NOAA). The findings of those reviews were published in the safety evaluation reports (SERs) relating to the construction permits and operating licenses for the Millstone Point Nuclear Power Station Units 1 and 2 and in the SER relating to the construction permit for Millstone Unit 3.

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

(1) faulting exposed in the excavations for plant structures

(2) additional offshore seismic reflection profiling by the USGS

(3) the choice of tectonic provinces and the occurrence of the January 9, 1982, magnitude 5.75 earthquake in south central New Brunswick, Canada

(4) USGS clarification regarding the localization of the Charleston earthquake

Although no faults are shown adjacent to the site on the geologic maps, in the process of mapping the excavation at the Millstone Unit 3 site, 11 fault zones were uncovered. Potassium-argon dating of clay gouge from some of these fault zones indicates that the last activity along these zones occurred about 142 million years before the present (mybp). Excavations along the discharge tunnel uncovered slumped and faulted ablation till and outwash deposits. These features were found to be quite common in the outwash and are believed to be related to penecontemporaneous soft sediment deformations.

The staff considers the deterministic investigations performed by the applicant at the Millstone site adequate, and, on the basis of present knowledge, the NRC staff concludes that the site faults do not present a surface displacement hazard to the site, nor do they have the potential to localize earthquakes in the site vicinity. During past licensing decisions the NRC and AEC have held to the position that the relatively high seismic activity within the Coastal Plain Province in the vicinity of Charleston, South Carolina, including the 1886 modified Mercalli intensity (MMI) X earthquake, was, for licensing decisions, related to a unique tectonic structure there. Therefore, in the context of the tectonic province approach, an MMI X earthquake should not be assumed to occur anywhere else. This conclusion was based primarily on the persistent historical seismicity that has characterized the meizoseismal zone of the 1886 Charleston earthquake. It was also based on evidence, though not strong, of unique geologic structure. Lacking definitive information, the NRC (AEC) based its conclusion in part on advice from the USGS.

In 1973, with AEC funding, the USGS began extensive geologic and seismic investigations in the Charleston, South Carolina, region. These studies are still under way. As a result of these investigations, a great deal of information has been obtained, but the source mechanism of the seismicity still is not known. Many working hypotheses have been developed based on the research data. These hypotheses are described in the Virgil C. Summer Safety Evaluation Report (NUREG-0717) and will not be discussed here, except to say that some of these theories postulate that an earthquake the size of the Charleston earthquake of 1886 could recur in other areas of the Piedmont, Atlantic Coastal Plain, New England, and continental shelf in addition to the epicentral area.

Because of the wide range of opinions within the scientific community concerning the tectonic mechanism for the 1886 Charleston earthquake, the USGS clarified its position regarding the localization of this earthquake in the vicinity of Charleston, South Carolina (see November 18, 1982, letter from James F. Devine (USGS) to Robert E. Jackson (NRC)). The staff has formulated an interim position concerning eastern seismicity in general and the 1886 Charleston earthquake in particular (see Appendix E and March 2, 1983, memorandum from R. Vollmer to H. Denton). As part of future research efforts described in that position, the staff is addressing the uncertainties about eastern seismicity by probabilistic studies funded by NRC and conducted by Lawrence Livermore National Laboratory (LLNL). At the conclusion of these studies, the staff will assess the need for a modified position with respect to specific sites. In the interim, considering the speculative nature of most of the eastern seismicity hypotheses, the low probability of large earthquakes in the Eastern United States, and the present knowledge of the geology and seismology of the region, the NRC staff considers that the Millstone Unit 3 design basis does not need to consider the Charleston earthquake of 1886 as a potential near source event at this time.

In licensing decisions since about 1976 regarding the seismic design basis of nuclear power plants located in the Precambrian-Paleozoic crystalline section of the Appalachian orogeny, particularly in New England and the northernmost Piedmont, the staff has recognized the New England-Piedmont Tectonic Province. Because the maximum historic earthquake in different parts of this province was an MMI VII, magnitude about 5.3 (Nuttli and Herrmann, 1978), it has not been important to critically consider subdividing this province. On January 9, 1982, a body-wave magnitude (m_b) 5.75, MMI VI, earthquake occurred in south central New Brunswick, Canada. Extensive research by Canadian scientists, the USGS, universities, consulting firms, and the New England utility companies is under way regarding that earthquake. The staff is monitoring the results of these studies and assessing them with respect to nuclear power plant sites in the region. Also the deterministic and probabilistic studies concerning seismicity of the eastern seaboard and New Brunswick that are described in the NRC Eastern Seismicity Plan (Appendix E) will be evaluated by the staff as the results become available. In addition, as noted in NRC questions Q230.4, Q230.6, and Q230.7, the staff and applicant maintain different positions regarding the New England-Piedmont Tectonic Province, the 1982 New Brunswick earthquake, and its potential association with a specific tectonic structure. However, as supported by this report, the staff finds that there is no need to change the existing design basis for Millstone. This conclusion is supported by the earthquake recurrence statistics and the valuable insights gained as part of the probabilistic seismic hazard studies.

The staff concludes that the applicant has satisfied the requirements of Appendix A to 10 CFR 100. The staff also finds that the FSAR conforms to the applicable sections of the SRP and RG 1.70, Revision 2.

The NRC staff has completed its review of the geological and seismological sections of the FSAR. On the basis of its review of the FSAR and pertinent documents from the published scientific literature, the staff concludes that the applicant has conducted an adequate investigation of the site and the region around it. On the basis of its present knowledge, the staff concludes that there are no capable faults in the site region and there are no geologic conditions that pose a hazard to the nuclear plant and its facilities.

2.5.1.1 Regional Geology

The Millstone site is located in the Seaboard Lowland section of the New England Physiographic Province, which is primarily a northward extension of Appalachian geology and topography. The Seaboard Lowland is a maturely eroded and glaciated peneplain, where bedrock is generally overlain by a few feet to a few hundred feet of glacial debris.

Although the FSAR points out that the New England Physiographic Province is divided into five sections (Fenneman, 1938) - the Taconic, the Green Mountain, the White Mountain, the New England Upland, and the Seaboard Lowland - the staff has in the past assumed a uniform New England-Piedmont Tectonic Province for its reviews of sites located in New England and the northernmost Piedmont since 1976. The staff concludes that the Millstone site is within the New England-Piedmont Tectonic Province. The original basement and sedimentary rocks of the province have been extensively folded, faulted, metamorphosed, and intruded by igneous rock during successive episodes of orogenic activity.

The Devonian Acadian orogeny, 410 to 360 mybp, was the major event affecting the New England portion of the province, producing a northerly trending complex series of anticlinoria and synclinoria that are characteristic of New England.

The site is located 22 to 25 km (14 to 15 mi) south of the Honey Hill fault where it truncates the southern end of the Merrimack synclinorium. The Honey Hill-Lake Char thrust fault is the boundary between the Southeastern New England Platform, which is a part of the Avalon Terrain, and the New England-Piedmont Tectonic Province. Rodgers (1972) suggests that this boundary represents a Paleozoic collision zone between a plate containing the Southeastern New England Platform and a plate containing the New England fold belt. O'Hara and Gromet (1983) indicate that the Honey Hill fault zone was active during the Late Paleozoic and that ductile deformation and metamorphism associated with the Alleghenian orogeny extend well into southern Connecticut, although much of the deformation that occurred in the Millstone area has been attributed to the Acadian orogeny. An Alleghenian age for mylonitization within the Honey Hill fault zone suggests it should be considered as a possible site for the major Late Paleozoic strike-slip displacement inferred from paleomagnetic studies (Kent and Opdyke, 1978, 1979; Van der Voo, French, and French, 1979) for parts of coastal New England and maritime Canada. However, this possible major late Paleozoic boundary is not reflected in Neogene tectonics in the site region. The brittle structures across such fundamental Paleozoic structures as the Honey Hill-Lake Char fault zone are generally consistent in physical characteristics and orientation.

Glaciation has greatly changed much of New England. The rock outcrops have been rounded and smoothed and the valleys filled with glacial deposits. The rivers in many cases had to develop new channels after being dammed by glacial deposits. Pleistocene glacial deposits are widespread throughout New England. End moraines occur along the southern margins and are prominent along Long Island, Block Island, Martha's Vineyard, and Cape Cod, and in southern Rhode Island and Connecticut.

2.5.1.2 Site Geology

Millstone Unit 3 is located on a low peninsula bounded on the west by Niantic Bay, on the south by Twotree Island Channel, and on the east by Jordan Cove. With the exception of scattered shoreline and inland exposures of Paleozoic bedrock, the plant area is blanketed by glacial debris, and some fill, ablation and basal tills, and unclassified stream deposits, ranging up to 48 ft in thickness.

Bedrock consists primarily of early Paleozoic metamorphic rock, the Monson gneiss. The Monson gneiss has been intruded by the sill-like Westerly granite of Pennsylvanian or younger age. Both the gneiss and the granite have been observed to be moderately jointed, hard, sound crystalline rocks with occasional weathering limited to the gneiss-granite contacts, joint planes, and foliation partings. With the exception of a few scattered shoreline exposures, the irregular bedrock surface lies beneath a soil cover (fill and glacial) ranging from 8 to as much as 48 ft in thickness.

Within the area of the major plant structures, overburden thickness and type is variable. Generally, the soil consists of fill derived from construction activities connected with Millstone Units 1 and 2. This heterogeneous loose material, up to 8 ft in thickness, consists of quarry waste, sands, gravels, silt, clay, and boulders and occurs sporadically throughout the Millstone Unit 3 area. Where not covered by fill, poorly stratified ablation till up to 20 ft thick overlies dense basal till as much as 37 ft in thickness. Occasionally, the basal till is absent with the ablation till resting directly over bedrock.

2.5.1.3 Reevaluation of the New Shoreham Fault

In a May 3, 1983, NRC letter to the applicant, question Q231.1 requested that the applicant assess new information being developed in Block Island Sound and Long Island Sound by the USGS and the Connecticut Geological and Natural History Survey. Additional offshore reflection profiles have been run by these organizations in a joint effort since the Weston Geophysical Study made for the New England Power Company (NEPCO, 1976) to investigate the New Shoreham fault. As a result of this study (NUREG-0424) the staff concluded that

(1) There is no well-defined relationship of seismicity with the New Shoreham fault. Also, the seismicity in the vicinity of the New Shoreham fault is low for the region.

(2) On the basis of a careful study made of the sedimentary sequence overlying the fault to determine the age of the youngest layers that have been displaced, the minimum date for last movement along the New Shoreham fault is at least 43,800 years ago and may well be 20 million years ago.

(3) The New Shoreham fault is not a capable fault as defined in 10 CFR 100, Appendix A.

The new seismic reflection profile lines will appear in USGS (Preliminary Open-File Report 83-XXX). Preliminary indications from interpretations by USGS of the profiles were

(1) The landward limit of the New Shoreham fault, previously traced to 16 km south of the Rhode Island coast by Weston Geophysical, extends an additional 6 km to the northwest to 10 km south of Rhode Island.

(2) The fault, only observed within the coastal plain strata, has a vertical offset of 15 m. Vertical offset is not observed along the surface of the coastal plain strata.

(3) The New Shoreham fault was active in Block Island Sound after the deposition of the Late Cretaceous-Early Tertiary coastal plain strata and before the deposition of glacial drift during the Pleistocene (2 mybp).

The applicant obtained and evaluated the new seismic reflection profiles. The staff also evaluated copies of the profiles at the Woods Hole office of the Connecticut Geological and Natural History Survey. The applicant and the staff agree that the Pleistocene material above the New Shoreham fault shows no evidence of offset or disturbance.

The new profiles as interpreted by the USGS, the applicant, and the NRC staff support the conclusions summarized above in the NEPCO SER (NUREG-0424) that the New Shoreham fault is not capable as defined in 10 CFR 100, Appendix A.

2.5.1.4 New Brunswick Earthquake Study

The applicant has submitted reports prepared by Weston Geophysical (1983, 1984) to demonstrate that the Millstone site is situated in a tectonic environment that is dissimilar to that of the epicentral area of the January 1982 earthquakes. These reports also present data to demonstrate a reasonable correlation of the New Brunswick seismicity to definable tectonic structure, as stated in Appendix A to 10 CFR 100. The New Brunswick tectonic structure is defined on the basis of surface geology, geophysical data, gravity modeling, and seismicity. The applicant described and discussed exploratory trenches which intersect faults in the epicentral area, one of which exhibits emplacement of glacial till within the fault gouge.

As noted by the applicant (Weston Geophysical, 1984, Vol. 1) the accumulated data on the New Brunswick 1982 earthquake sequence are currently being discussed in the literature, and various agencies are planning additional field investigations in the region in 1984. The applicant compared the geological and geophysical environment and the seismicity between the Millstone site and New Brunswick epicentral area using currently available data. The comparisons are restated in items 1-4, below. Following that is the staff's evaluation of the evidence for the proposed New Brunswick tectonic structure and, with the exception of the seismicity comparison (item 4) which is briefly mentioned here but discussed in detail in Section 2.5.2.4 of this SER, a discussion of the comparisons is presented.

(1) The geological environment and geophysical signature of the Millstone site region can be generally characterized as being typical, with respect to brittle fracture, of Northern Appalachian terrain. However, the New Brunswick epicentral region has been identified as a distinctive tectonic block that can be geologically and geophysically distinguished from immediately adjacent terrain.

(2) Local faults in the Millstone site area have been dated to be pre-Cenozoic (older than 142 mybp); faults in the New Brunswick epicentral area are characterized by an uncertain date of last movement because of the problematic emplacement of Wisconsin age glacial till (11,000 years before present) within one fault zone.

(3) Lithologic contrasts exist between the Millstone site area and the New Brunswick proposed tectonic structure. Also, metamorphic structural elements of the two areas are significantly different.

(4) The seismicity in the immediate vicinity of the site is relatively less than in adjacent terrain. It is below the average seismicity of the Northern Appalachians and has no historical record of significant seismicity. The immediate vicinity of the New Brunswick epicenter, however, is an order of magnitude more seismically active than the adjacent terrain. It is more active than the average seismicity of the Northern Appalachians, and in particular than the immediate vicinity of the Millstone site.

2.5.1.4.1 New Brunswick Tectonic Structure

The tectonic structure to which the New Brunswick seismicity is spatially correlated is defined by the applicant as a fault bounded, apparently counterclockwise-rotated crustal block, approximately 1650 km^2 (644 mi^2) in area. The boundaries of this crustal block include the Catamaran fault to the south, a steep gravity gradient corresponding to a zone of significant reverse faulting to the west, brittle faults to the north, and northwest-trending gravity gradients to the east.

It is the staff's position that the evidence for the crustal block along the boundaries given are not conclusive. The Catamaran fault is a reasonable southern boundary; it has the proper right lateral strike slip motion to fit the model. The mylonitic zones, brittle faults, and cleavage on the west could indicate a western boundary. However, rotation along discontinuous brittle faults which appear to be normal faults (see Weston Geophysical, 1984, Fig. 3-1A) to the north, and along northwest-trending gravity gradients to the east which may or may not indicate faulting, are not sufficient evidence for rotational tectonics. More and better supportive data are required for this interpretation to be adequately supportable.

The staff has observed and would suggest that the Rocky Brook-Millstream fault is a more demonstrable northern boundary and possibly constitutes part of the western fault boundary. In addition, the applicant has not demonstrated how this block is uniquely structured to cause a localization of seismicity in the New Brunswick region. The three-dimensional gravity model which the applicant proposed to illustrate that the January 1982 earthquake sequence and the June 1982 Trousers Lake event are associated with the rotated tectonic structure, is one of several interpretive models that could fit the data. It is not a unique interpretation, and therefore, it is not a reliable correlation of seismicity and structure.

2.5.1.4.2 Trenched Faults in the Epicentral Area

The time of last movement on the northwest-trending faults in the epicentral area is not known, nor is there any evidence that these faults were affected by the 1982 earthquakes. However, one of the faults exposed in trench I exhibits Pleistocene deformation of the glacial till overlying the bedrock and emplacement of till within the fault zone. At least two deformational events are postulated by the applicant: (1) squeezing and injection of fault gouge/breccia into the overlying till near ground surface, and (2) dilation of the fault to allow emplacement of till within the fault. The applicant postulates that the deformations may stem from a combination of tectonic and glacial activity and dates them at approximately 11,000 years before present.

The staff is of the opinion that the gouge/breccia was squeezed into the overlying till while in a plastic state because of large overburden pressure induced upon the fault gouge by the over-riding glacier. The emplaced till, which was mapped in the south wall of trench I, measures 0.5 m by 0.2 m and is located approximately 1 m below the till/rock interface. According to the applicant, its presence in the fault zone may indicate post-Wisconsinan deformation and tectonic activity in the epicentral area. In the opinion of the staff the emplaced till may also have several non-tectonic explanations. Such a small amount of material may have been emplaced by animals burrowing into the soft fault gouge, which was later filled with the overlying Wisconsin-aged till, or the emplacement could have resulted from glacial penetrations. The staff concludes that a tectonic explanation for the emplacement till has not been demonstrated and, therefore, evidence of post-Wisconsin faulting in the epicentral area has not been demonstrated.

2.5.1.4.3 Conjugate Fault Set

The New Brunswick earthquake sequence was concentrated in a volume approximately 6 km (3.8 mi) north-south by 6 km (3.8 mi) east-west by 8 km (5 mi) deep and arranged in a north-trending conjugate "V" pattern (Wetmiller et al., 1983). The larger earthquakes of magnitude greater than 4.5 were predominantly reverse motions occurring on north striking mid- to high-angle faults in the upper 8 km

(5 mi) of the crust, indicating an east-west compressive, horizontal regional stress field. The earthquakes induced a 2.5-cm break in the bedrock surface, which was probably not a primary movement. The offset occurred on a preexisting joint that was oriented north-south, parallel to the trend of the aftershocks. Wetmiller et al. (1983) state that a more fundamental relationship between the joints and the earthquakes may be possible, as they may be products of the same contemporary tectonic environment. Although field investigators have not yet been able to associate the earthquake sequence with known preexisting faults, the staff is of the opinion that there are compelling reasons to believe that a conjugate set of reverse faults, which are concordant with the north-south-trending joints, are the generators of the 1982 earthquake sequence. Furthermore, the conjugate faults may be associated with a larger still-undefined tectonic structure within the Miramichi Anticlinorium.

2.5.1.4.4 Comparative Analyses and Conclusions

The staff agrees with some of the results of the applicant's comparative analyses, namely, that lithologic contrasts exist between the Millstone site area and the New Brunswick epicentral area. Also, the structural elements of the two areas are significantly different. The epicentral area is underlain by a Devonian granitic body, the North Pole pluton (Fyffe, 1982), which intrudes metamorphosed Cambro-Ordovician sedimentary, volcanic, and plutonic rocks of the Tetagouche Group. The Millstone site is underlain by early Paleozoic metamorphic rocks, the Monson gneiss, which has been intruded by the Westerly granite of Pennsylvanian or Permian age.

Like most of the New England portion of the New England-Piedmont Tectonic Province, the New Brunswick region is characterized by northeasterly trending anticlinoria and sysclinoria that were formed as a result of extensive deformation, primarily during the Acadian orogeny. Whereas, the Bronson Hill anticlinorium and Merrimack synclinorium, which are southwesterly extensions of the regional structures that characterize New Brunswick, are truncated by the Honey Hill-Lake Char fault zone at approximately 24 km (15 mi) north of the Millstone site. This fault zone is a low angle thrust fault boundary between the Avalonian rocks to the south and the units of the Merrimack sequence. Extensive Alleghenian deformation characterizes the Millstone site. The Catamaran fault, a major fault system, strikes north-south as it approaches a few miles to the south of the North Pole pluton and veers to the east. The fault exhibits from 12 to 33 km (7.5 to 21 mi) of right-lateral displacement. Little Mesozoic deformation is evident in the New Brunswick epicentral region; however, considerable Mesozoic tectonics, such as the Connecticut Valley Triassic Basin containing thick Mesozoic clastic sediments and diabase intrusives, are present in the Millstone site region.

In the applicant's comparison of the brittle structural elements at the Millstone site and at the New Brunswick epicentral area, the staff agrees that, at the site, the brittle structural elements are generally consistent in physical characteristics and orientation across fundamental lithotectonic boundaries such as the Honey Hill-Lake Char fault zone. The evidence provided by the applicant is not sufficient in establishing that in the New Brunswick epicentral area, the brittle structures are apparently bounded by and/or uniquely oriented within the proposed lithotectonic boundaries defining the proposed tectonic structure.

In conclusion, there is no evidence of capable faulting in the Millstone site area. On the basis of petrographic and radiometric studies, the latest movement on the site faults was determined to have occurred during the Triassic-Jurassic rifting of the continent, approximately 142 mybp. And there are no known tectonic structures that could be characterized as possible localizers of seismicity in the site vicinity.

In the opinion of the staff, the data presented by the applicant have not satisfactorily demonstrated that the rotated block represents a unique tectonic structure in the New Brunswick epicentral area. However, it is the staff's opinion that the conjugate faults defined by the aftershocks might reflect the existence of faults which could localize seismicity in the central New Brunswick region. There is no assurance, however, that similar faults do not exist in other areas of the Eastern United States. On the basis of available geological and geophysical information, there is no evidence of capable faulting in the Millstone site area, and there are no known tectonic structures that could be characterized as possible localizers of seismicity in the site vicinity.

2.5.2 Vibratory Ground Motion

2.5.2.1 Seismicity

In FSAR Section 2.5.2.1 and in response to staff questions Q230.4, Q230.6, and Q230.7, the applicant presented the results of a complete study of historical seismicity in New England up to June 1983. The historical record of earthquakes in New England is the longest of any region in the United States. This record extends back more than 400 years (NUREG/CR-2309) when the early settlers and missionaries made notes of the larger earthquakes. New England is characterized by the infrequent occurrence of low-to-moderate-intensity earthquakes; considering past patterns of population density and other factors, the early record can be considered complete only for events with epicentral MMIs of VII or greater.

The largest earthquakes within a 200-mi (320-km) radius of the site have occurred in a region extending from Cape Ann and Boston in Massachusetts to central New Hampshire. These include the 1727 Newbury event (MMI VII), the 1755 Cape Ann event (MMI VIII), the 1817 Woburn event (MMI VI), and two MMI VII events in 1940 at Lake Ossippee, New Hampshire. In addition to this region, MMI VII events were reported within 200 mi (320 km) of the site in 1737 and 1884 near New York City, and in 1791 near Moodus, Connecticut. The MMIs of some of the events discussed above have been reduced since the SER-CP (March 1974) following reevaluation by Coffman (Feb. 24, 1976), the USGS (1980), and other investigators (NUREG/CR-2309).

On the basis of the work of Street and Lacroix (1979), which utilized intensity information such as the total felt area, total area with MMI IV, or the rate of intensity falloff with distance from the event, some of the above earthquakes have assigned magnitudes. These include m_{blg} = 5.0 for the 1727 event, m_{blg} = 6.0 for the 1755 event, m_{blg} = 4.35 for the 1791 event, m_{blg} = 4.3 for the 1817 event, and an instrumental m_{blg} = 5.4 for the 1940 event. The 1884 event has an m_{blg} = 4.9 based on the work of Kafka (1980).

The Northeastern United States Seismic Network (NEUSSN), partially funded by the NRC, has been in operation since 1975 and provides locations and magnitudes of the more numerous, smaller earthquakes. Since the startup of the NEUSSN, the area within a 50-mi (80-km) radius of the site shows some clustering of microearthquakes in central Connecticut northwest of the site, and in the Narragansett Bay region east of the site. The significance of this activity is discussed in Section 2.5.2.3 of this report.

On January 9, 1982, a magnitude 5.75 earthquake occurred in south central New Brunswick, Canada, about 485 mi (775 km) north of the site. As discussed in Section 2.5.1, the applicant has provided a significant amount of information in an attempt to relate this earthquake to a specific tectonic structure in the New Brunswick region. This information includes geologic data, geophysical data and interpretation, and earthquake recurrence comparisons. One of the conclusions of Section 2.5.1.4 was that insufficient geologic and/or geophysical evidence exists to associate the 1982 New Brunswick earthquake with a specific structure. However, as discussed in Section 2.5.2.3, there is some indication that the level of seismicity in central New Brunswick is higher than that of the Millstone site area. The staff's use of these data and information is discussed in subsequent sections of this SER.

2.5.2.2 Geologic and Tectonic Characteristics of the Site and Region

As discussed in Section 2.5.1, the staff and applicant maintain different positions regarding the New England-Piedmont Tectonic Province and the 1982 New Brunswick earthquake and its potential association with a specific tectonic structure. These issues are extremely complex, particularly considering that on the basis of both historic and recent seismicity, different areas of the New England-Piedmont Tectonic Province appear to exhibit different levels of earthquake activity. On the basis of detailed earthquake recurrence information (see Section 2.5.2.3), it is apparent that significant different levels of seismicity exist in the NEPTP and that the definition of this province using surficial geologic features alone may result in the selection of safe shutdown earthquake (SSE) levels of varying degrees of conservatism, depending on the site location. Subdivision of the NEPTP has been recognized by the staff (NUREG-0896) in delineating the Cape Ann-New Hampshire seismic zone which intersects the NEPTP. This issue is further discussed in Section 2.5.3.

Other tectonic provinces within 200 mi of the Millstone site include the Central Stable Region, the Valley and Ridge, the Appalachian Plateau, and the Coastal Plain provinces. Other than the Coastal Plain, the above provinces and all other provinces outside the 200-mi radius are at a sufficient enough distance to have no impact on the Millstone site seismic design. The Coastal Plain also has no impact on the Millstone seismic design, although it may approach within 6 mi (10 km), because the controlling earthquake will be equal to or less than the controlling earthquake for the host tectonic province.

2.5.2.3 Correlation of Earthquake Activity With Geologic Structures or Tectonic Provinces

As discussed in the Safety Evaluation Report for Seabrook Station Units 1 and 2 (NUREG-0896, 1983), a zone, which extends from Cape Ann to central New Hampshire (and not beyond), is characterized by higher seismicity than the remainder of the New England-Piedmont Tectonic Province. Additionally the staff has found that the New Hampshire-Cape Ann seismic zone is spatially associated with a zone of shallow intrusives and volcanic rocks of predominantly Jurassic-Cretaceous age. This zone approaches to within about 106 mi (175 km) of the Millstone site. Assuming a recurrence of the largest historic earthquake within this zone, results in ground motion that is well below that of the SSE.

About 35 km (22 mi) northwest of the site there is a concentration of seismicity, in terms of the numbers of earthquakes, in the central Connecticut region around the towns of Moodus and Haddam. This region was the location of an intense microearthquake swarm from August through October 1981 and has been the location of repeated felt earthquakes that date back as far as 1568. The most recent swarm events have been discussed by both Ebel et al. (1982) and Weston Geophysical (September 1982). The depth of these small earthquakes was 1 km or less; the largest event was about a magnitude of 2.1. Although depths are not available for the historic earthquakes in this region, there is some indication that the 1791 earthquake also may have been relatively shallow, based on the small area that felt the earthquake compared with that which is expected for an MMI VII event. This is reflected in the magnitude estimate of Street and Lacroix (1979) (m_{blg} = 4.35) compared with that which is typically expected for MMI VII (m_{blg} = 5.25, Nuttli and Herrmann, 1978). Although the staff has not identified a specific structure associated with seismicity near Moodus, it appears as if it can be expected that shallow events will continue to occur there. The impact of the Moodus seismicity on the likelihood of earthquakes near Millstone is discussed in the following section.

2.5.2.4 Comparison of Millstone and New Brunswick Seismicity

As a result of staff questions, the applicant has submitted detailed seismicity and earthquake recurrence comparisons between the Millstone site region and the central New Brunswick region. It is the applicant's position that the immediate vicinity of the inferred tectonic structure is approximately an order of magnitude more seismically active than the surrounding New Brunswick region and in particular the 25-km (15.6-mi) radius region of the Millstone site.

Earthquake recurrence estimates involve the assessment of the annual likelihood of different magnitude events for a given geographic region. In order to accurately assess earthquake recurrence, the accuracy and completeness of the earthquake data base must be thoroughly assessed. The applicant has adequately addressed these issues in providing recurrence estimates to the staff. All recurrence data for each geographic region described below have been expressed in terms of (normalized to) the same size area (radius of 25 km (15.6 mi), making comparisons possible. The results discussed below would be similar if other normalized areas had been used. Least squares linear regression was performed on the annual cumulative number of earthquakes for different magnitude ranges for each geographic region, and sensitivity tests were provided showing which assumptions significantly altered the recurrence results.

In terms of the New Brunswick region, three different geographic areas were considered. The first conforms to the specific inferred tectonic structures supported by the applicant; the second conforms to a square region slightly larger than the inferred tectonic structure, and the third is a fairly large rectangular two-degree latitude by three-degree longitude block centered around the New Brunswick earthquake. Conclusions regarding the two smaller regions are the same and are not repeated. A general comment regarding the entire New Brunswick region is that the ability to detect and locate earthquakes, particularly prior to about 1975, is very poor. This adds uncertainty to the recurrence statistics.

With regard to the inferred tectonic structure, there were five earthquakes between magnitude 2.5 and 4.0 in the five years proceeding to the 1982 sequence. The applicant emphasizes that this rate is very high and that when projected over longer periods of time infers shorter recurrence of larger earthquakes (magnitude 5 to 6) particularly compared to the Millstone areas discussed below. Although the staff agrees that the apparent rate of earthquakes recorded prior to 1982 is high, because of the extremely short time frame (less than 10 years) for which there are accurate earthquake parameters, less emphasis is placed on these data. The staff notes, however, that the immediate vicinity of the Millstone site (25-km (15.6-mi) radius) has had no earthquakes greater than a magnitude of 2.5 for the past 30 to 40 years.

For the larger New Brunswick region, recurrence statistics are fairly stable, including sensitivity on the completeness assumed and whether the 1982 mainshock is included or excluded. The annual probability of a New Brunswick-size earthquake in a 25-km (15.6-m) radius is about 1×10^{-4} to 2×10^{-3}. On the average, this probability is higher than that calculated for the region of the Millstone site discussed below. This is important considering that the actual inferred tectonic structure appears to have a higher rate compared with the larger New Brunswick region.

A wide variety of geographic areas was considered for the Millstone site region. These include a 25-km (15.6-mi) radius, a 50-km (31-m) radius, a 20,000-km^2 (7,800-mi^2) and a 55,600-km^2 (21,700-mi^2) rectangular block, and two smaller blocks - one from the site eastward into the Naragansett Bay region, the other from the site north and west into central Connecticut. All recurrence rates were normalized to a 25-km (15.6-mi) radius area. For the large regions considered around Millstone, the recurrence estimate of an M = 5.8 is dependent on whether, and to what extent, historic seismicity is included in the calculations. Including historic seismicity the annual probability of an M = 5.8 is about 3×10^{-5} to 5×10^{-5} years for a 25-km (15.6-mi) radius area. Using only recently recorded instrumental data, the return period for an M = 5.8 event is similar to that calculated for the large New Brunswick region. This is consistent with Ebel (1984) who found that recently recorded instrumental data appear to reflect a higher rate of activity compared with that which has been experienced in the past 200 to 300 years. It is the applicant's position that the recurrence estimates using the complete earthquake catalog are more accurate.

For the smaller geographic regions, in particular the region including the site and Naragansett Bay area conforming roughly to the southeast New England platform portion of the NEPTP, the probability of an M = 5.8 event appears to be about 5×10^{-5}. Other sensitivity tests show that the Moodus seismicity can affect the recurrence statistics, particularly because these are numerous small events. The annual probability of exceedance ranges from being roughly equal to that of New Brunswick to being lower than that for New Brunswick. It is clear that for some of the local regions around the Millstone site, the rate of seismicity is significantly lower than that of New Brunswick.

It is the applicant's position that the recurrence information provided demonstrates that an M = 5.25 event is appropriate for the Millstone site. In general, results of the recurrence comparisons show that the annual probability of an M = 5.25 event near the Millstone site is lower than that of the New Brunswick region. Thus, the staff agrees with the applicant that the recurrence data support the conclusion that there is no need to the change the existing design basis. However, sufficient uncertainty exists with these recurrence comparisons that ground motion from an M = 5.75 earthquake is compared with the existing design basis. This issue is further discussed in Section 2.5.2.7 including comparisons and differences between M = 5.25 and 5.75 site specific spectra definitions, the SSE accepted response spectrum at the CP stage of review, and two detailed probabilistic seismic hazard studies completed for the Millstone site.

2.5.2.5 Maximum Earthquake Potential

As discussed in previous sections of this SER, although the positions of the staff and applicant regarding the association of the New Brunswick earthquake with a specific geologic structure are different, the earthquake recurrence statistics demonstrate that the likelihood of moderate earthquakes is higher in New Brunswick than at the Millstone site. Before 1982 the largest events in the NEPTP were MMI VII, being equivalent to about a magnitude of 5.25 (Nuttli and Herrmann 1978); the 1982 New Brunswick mainshock is equivalent to the M = 5.75. Although the implications of an M = 5.75 event is assessed in Section 2.5.2.7, the staff is not explicitly recommending its use at Millstone. The staff has used comparative seismicity in other cases (for example, Midland SER, NUREG-0793) and agrees with the applicant that the data presented demonstrate that there is no need to change the existing design basis at Millstone. However, because the staff has not been able to associate the M = 5.75 earthquake with a specific tectonic structure, the differences between the ground motion associated with this earthquake and the CP-approved SSE spectrum are discussed.

2.5.2.6 Seismic Wave Transmission Characteristics of the Site

Most of the seismic Category I structures at the Millstone site are founded on competent crystalline bedrock of the Monson gneiss formation. However, the emergency generator building and the control building along with parts of the circulating water discharge tunnel and service water intake lines are founded on either glacial till or structural fill. The thickness of the soil material ranges from a few feet up to about 30 ft. The applicant has analyzed for potential ground motion modifications through the soil material to assess both the potential for liquefaction and amplification. The applicant has assumed that the model used in the SHAKE analysis is sufficiently conservative to account for local variations in the subgrade and their effect on structural response. The staff has reviewed the assumed amplification of ground motion using the SHAKE code and has found that results for the top of the basal till are conservative.

2.5.2.7 Safe Shutdown Earthquake

2.5.2.7.1 Deterministic Estimates of Ground Motion

The applicant has used a Newmark-type response spectrum anchored to 0.17-g peak horizontal ground acceleration for the SSE. As stated in previous sections, the staff will compare this to the ground motion generated by M = 5.25 and M = 5.75 site-specific spectra. In addition, Section 2.5.2.7.3 discusses these differences in a probabilistic context using two seismic hazard studies which have specific results for the Millstone site.

In recent NRC staff SERs (Catawba, NUREG-0954; Wolf Creek, NUREG-0881; Limerick, NUREG-0991), the staff has indicated that site-specific spectra obtained from appropriate suites of strong motion records of earthquakes are more in accord with the controlling earthquake size, frequency content of response spectrum, and local site conditions than are standard response spectra such as the RG 1.60 spectra. In this method, spectra obtained from earthquakes within half a magnitude of the SSE recorded at a distance less than about 25 km (15.6 mi), with geologic conditions similar to those at the site are assembled. It is the staff's position that the 84th percentile spectrum is appropriate for describing ground motion to be used in evaluating the design spectra of nuclear power plants.

Various site-specific spectra have been generated for both an M = 5.25 and an M = 5.75 earthquake using rock records (Millstone site conditions) recorded at distance less than about 25 km. These include those generated by Lawrence Livermore National Laboratory for the NRC staff (NUREG/CR-1582, Vol. 4; Wolf Creek, NUREG-0881; Limerick, NUREG-0991) and those generated by consultants to various applicants (Sequoyah, NUREG-0011; Fermi, NUREG-0798; Perry, NUREG-0887). The review of these data indicates that for the M = 5.25 earthquake case, the design spectrum for Millstone is approximately equal to or larger than the 84th percentile of the various site-specific spectra. For the M = 5.75 case, the 84th percentile would exceed the design spectrum by up to 40 to 50% for frequencies above about 3 Hz. However, on the basis of the recurrence statistics, this exceedance is judged to be an upper bound and would be reduced if a smaller earthquake were used. In addition, as discussed in Section 2.5.2.7.3, these differences are relatively minor when judged from a probabilistic viewpoint.

2.5.2.7.2 Probabilistic Estimates of Ground Motion at Millstone

Three different estimates of probabilistic seismic hazard are available for the Millstone site. The first is that which is available from the NRC's Systematic Evaluation Program (SEP), the second is a study completed by a consultant to the applicant as part of the applicant's Probabilistic Safety Study (PSS), (NNECo, 1983), and the third is the ongoing joint Office of Nuclear Regulatory Research and Office of Nuclear Reactor Regulation Seismic Hazard Characterization Program (SHCP).

In studying earthquake hazards, there is concern about the probability that an earthquake or its associated ground motion will occur at a site during a specified period of time. The exceedance probability is the probability over some period of time that an earthquake will generate a level of ground shaking greater than some specified level. The return period is the reciprocal of the annual probability of exceedance on the average over long periods of time between events causing ground shaking exceeding a particular level at a site.

An important part of completing a seismic hazard analysis involves the selection of an approach to incorporate the uncertainty of all input parameters into the analysis. Difficulty in accounting for this uncertainty is one of the reasons the staff has used probability studies in a limited sense. All the studies which have specific results for Millstone allow for the incorporation of uncertainties and alternative hypotheses on many of the hazard input parameters. These include seismic source zonation, earthquake occurrence rates and the upper magnitude cutoff, and ground motion attenuation equations and their associated uncertainty. Results for the SEP are not discussed here as they have been updated as part of the SHCP.

The applicant has concluded that the return period (50th percentile) for a peak acceleration of 0.17 g at the Millstone site is about 12,000 years. Results included in the SHCP interim report (NUREG/CR-3756) are more conservative than this by about one order of magnitude. This large difference reflects input assumption differences, particularly with respect to ground motion attenuation. At this time the staff does not necessarily believe that one is wrong and the other is right. This difference also points out that specific reliance upon the absolute numbers is not warranted.

Both studies however provide valuable insights regarding the impact of ground motion above the 0.17-g response spectra. Although large absolute differences exist, both studies show that the actual seismic hazard would decrease by only a factor of about 3 if ground motion were 40 to 50% higher than the 0.17-g response spectra. The staff judges that the factor of 3 reduction in hazard is likely to be an upper bound because (see Section 2.5.2.7.1) the 40 to 50% exceedance is likely an upper bound. The implication of this, is that a change in the acceleration of 40 to 50% implies a small relative risk difference. The PSS provides insights regarding the contribution of various acceleration ranges to the total probability of core melt. Results of both the applicant's and the staff's preliminary review indicate that the contribution to core melt from the seismic hazard for peak accelerations less than about 0.30 g are small. This would tend to support the conclusion that differences of 40 to 50% in ground motion at accelerations less than or equal to about 0.17 g to 0.25 g are not significant when viewed from the perspective of risk.

Although the Millstone Unit 3 PSS provides strong insight as to the limited significance, in terms of risk, of seismic ground motion up to about 0.3 g at Millstone Unit 3, the record does not adequately reflect the more specific capabilities of individual structures, components, and equipment to withstand seismic ground mitions above the existing design basis. As confirmation of the conclusions drawn in this evaluation, the staff will require the applicant to utilize the results of the PSS to document the seismic capacity, at accelerations up to 0.25 g, for high confidence of low probability of failure for individual controlling failure modes for structures and equipment. In addition the applicant will be required to assess plant fragilities, for various acceleration levels considering those risk scenarios that include the majority of seismic risk to the plant. These confirmatory studies will be required to be completed before specified low power test levels are exceeded.

2.5.2.7.3 High Frequency and Vertical Ground Motion

As has been reported by Chang (NUREG/CR-3327), Cranswick et al. (1982), and Weichert et al. (1982), strong ground motion from the Gaza, New Hampshire earthquake of January 18, 1982, m_b = 4.8, and aftershocks of the New Brunswick earthquake of January 9, 1982, m_b up to 4.8, appear to indicate that the recorded motion was enriched in high frequencies (above ~10 Hz) when compared with typical recordings from California. This observation raises the question of whether earthquakes in the Eastern United States produce more high-frequency ground motion than they were previously thought to produce.

Considerable effort has been and is being expended, including effort by specific NRC Office of Nuclear Regulatory Research contractors, in attempting to determine the cause of the high-frequency energy. It is apparent that this issue is extremely complex and it is likely to be some time before the cause of the recorded high frequencies is fully understood. The following examples illustrate some of the parameters that complicate the issue. Most of the strong-motion instruments in New Brunswick which recorded aftershock data in 1982 were not rock sites and thus site effects may be one reason some of the high frequencies were recorded. The one hard-rock station, which also showed enriched high frequencies, recorded only one earthquake above a magnitude of 4. Different types of strong-motion instrumentation have also been used at different times, each having different frequency response and high-frequency limitations. This makes it extremely difficult to draw conclusions regarding high frequencies in general, and in particular for magnitudes above ~5.0, the size of the events which we are concerned about in the Eastern United States.

As reported by Mueller and Cranswick (1982) and Boatwright and Astrue (1983) for the New Brunswick aftershock data recorded in 1982, there does not appear to be systematic source differences between New Brunswick data and data recorded in California. Preliminary results of work undertaken in 1983 (Cranswick 1984) appear to show that stress drops for very small magnitude (less than ~2.0) events decreases as the magnitude decreases, and that the site response has an effect on the high-frequency content of the recorded ground motion. These examples demonstrate the complexity of this issue. The staff currently judges that little evidence exists to support systematic source differences, although this issue is still under investigation, and that the "high-frequency issue" may have generic implications. The NRC will continue to monitor and support generic work on this topic, and any impact this work has on the design basis of power plant sites. Regarding the Millstone 3 site, the staff presently believes that the most appropriate and applicable strong-motion information is found in the accelerograph data used to develop the rock site specific spectra discussed in Section 2.5.2.7.1.

The vertical-component design response spectra for the Millstone 3 site have been taken as two-thirds of the horizontal spectra for all frequencies. It is important to both compare this assumption to the available data from the New Hampshire and New Brunswick earthquakes, and to determine the significance of the two-to-three ratio assumption in light of recommendation of a higher ratio for frequencies above 4 Hz contained in RG 1.60.

In examining the existing eastern data (New Hampshire and New Brunswick), no consistent trend in the vertical to horizontal ratio is evident and the scatter in results at a given frequency is at least an order of magnitude. Although Newmark and Hall (NUREG/CR-0098) recommended the use of two to three for all frequencies, they recognized that large scatter exists in this ratio. Studies of Western U.S. earthquakes (e.g., NUREG/CR-1175; Agbabian, 1983) have shown that the assumption of the vertical taken as two-thirds of the horizontal is generally conservative. The staff has also examined peak acceleration data included in site-specific spectra and found that the two-thirds assumption is a good average although individual values show wide scatter.

Considering the above, the staff concludes that the vertical-design spectrum at Millstone is adequate. The current Eastern U.S. strong-motion data are presently insufficient to draw conclusions regarding this issue. The staff will continue to monitor ongoing generic work and any new data that become available, and will assess these results with respect to both the high frequencies and the ratio of vertical to horizontal ground motion.

2.5.2.7.4 Conclusion

In conclusion, the staff finds the existing design basis adequate with respect to the impact of the New Brunswick earthquake on the Millstone site. This conclusion is supported by the earthquake-recurrence statistics and the valuable insights gained as part of the probabilistic seismic hazard studies. Additionally, on the basis of available geological and geophysical information, there is no evidence of capable faulting in the Millstone site area, and there are no known tectonic structures that could be characterized as possible localizers of seismicity in the site vicinity. The NRC is currently undertaking a generic program regarding the quantification of seismic design margins. A confirmatory program using available plant-specific information on insights regarding seismic capabilities beyond the design is recommended for the Millstone 3 site.

2.5.2.8 Operating-Basis Earthquake

As currently defined, the operating-basis earthquake (OBE) and its associated response spectrum is one-half the SSE response spectrum. The staff considers the OBE acceptable even in light of the unresolved issue of the 1982 New Brunswick, Canada, earthquake. The basis for this conclusion is the staff's review of various probabilistic peak acceleration maps (Applied Technology, 1978; Algermissen et al., 1982; NUREG/CR-3756; NNECO, 1984), which indicate that the return period of the OBE is estimated to be on the order of hundreds of years. This is consistent with hazard results provided by the applicant. The staff finds this to be acceptable in light of the Appendix A to 10 CFR 100 definition of the OBE as "that earthquake which...could reasonably be expected to affect the plant site during the operating life of the plant."

2.5.3 Surface Faulting

Although none of the published geology maps show faults in the vicinity of the plant site, 62 faults were found during the mapping of the rock excavation for Millstone Unit 3. Forty of the faults have apparent displacements of less than 1 ft, with the remaining faults having apparent displacements greater than 1 ft. Eleven separate fault zones (T-1, T-2, T-3, 18, 471-1541, 1599, 1940, 2250, 2282-2295, 2339-2347, and 2380) were identified. The faults are evaluated and discussed in several reports (NNECO, 1975, 1976, 1977, and 1982).

Most of the faults trend northerly and dip at high angles either to the east or to the west. Table 2.5.3-1 of the FSAR shows the apparent displacements of the faults in the horizontal plane and lists the calculated displacements, determined from the offset pegmatite veins and from slickenside information. The eastern blocks of T-3, T-2, and 1599 and most of the faults in the pump house appear to be downthrown relative to the western block, whereas with faults T-1, 18, 461 (1541), 368, 2251, and 2426, the western block appears to have been downthrown relative to the eastern block. A complex irregular pegmatitic intrusion has obscured the contacts in the vicinity of faults 2250 and 2282, making it impossible to determine the sense of displacement. Fault 1940 shows low-angle thrust displacement of between 1 and 2 in. toward the northeast.

The applicant performed potassium/argon age dating, petrographic analysis, x-ray diffraction studies, soils mapping, and detailed mapping of the fault zones, which indicates that the faults at the Millstone site are noncapable features. The petrographic analysis shows that the cataclasite in the faults has been silicified and hydrothermally altered, and that the fractures and cracks have been filled with chlorite. Prismatic quartz crystals, drusy quartz, and the silicified cataclasite found in the fault zones would be fractured and/or granulated if any additional movement had occurred. The radiometric age dates on the fault gouge indicate that the last activity along the faults occurred approximately 142 mybp. The petrographic and radiometric studies are reinforced by the published geologic history of the region.

Considering all the geologic data presented, the applicant concluded, and the staff concurs, that the faults at the Millstone site are not capable. The last activity along them occurred approximately 142 mybp. This indicates that the faulting at the site is related to the Triassic-Jurassic rifting or older events as in the case of the 1940 fault. There is no evidence of capable faults within the 5-mi radius of the site.

2.5.4 Stability of Subsurface Materials and Foundations

2.5.4.1 Site Conditions

2.5.4.1.1 General Plant Description

Major structures at Millstone Unit 3 are the containment structure, auxiliary building, fuel building, control building, emergency generator enclosure, and circulating and service water pumphouse. These safety-related structures, systems, and components that were reviewed for foundation and slope stability are listed in FSAR Table 3.2-1.

2.5.4.1.2 Site Investigation

The field investigation consisted of borings, standard penetration tests, piezometer installations, water pressure tests, geologic mapping, and seismic surveys to determine compressional and physical properties of the soil and rock. A total of 78 test borings, both vertical and inclined, were drilled in the rock and soil at the site. Groundwater elevations were monitored over a period of 2 years in some borings before construction. Water pressure tests were performed in three borings to assess the degree of weathering and permeability of the bedrock. Seismic surveys (consisting of refraction, cross-hole and up-hole surveys) were employed at the site to determine the values of dynamic soil and rock properties.

2.5.4.1.3 Properties of Subsurface Materials

Natural materials underlying the site include beach sand, unclassified stream deposits, ablation till, basal till, and hard crystalline bedrock of the Monson gneiss formation.

The beach sand deposits of uniform silty sands are present for the most part only in the cove east of Bay Point, in the area of the circulating and service water pumphouse. The beach sand is a loose-to-medium-dense deposit.

Unclassified glacial stream deposits west and southwest of Millstone Unit 3 consist of sands with some silts and gravels. Thicknesses of the deposits vary. In general, the deposits are medium-dense to dense, cohesionless materials, with loose deposits occurring within a limited zone above basal till.

No major plant structure, pipeline, or duct is founded on either the beach deposits or the unclassified stream deposits.

The ablation till overlies the basal till in the area where the major plant structures are located. It consists of glacially transported debris of medium-dense to dense silty sand with 20 to 40% fines, finer than the No. 200 sieve. Approximately 500 ft of the circulating water discharge tunnel is founded on compacted fill overlying ablation till.

Basal till overlies bedrock at the site area, varying in thickness from less than 5 ft in the pumphouse area to over 40 ft under the turbine building. It is a very dense, glacial compacted material of low permeability consisting of a widely graded mixture of cobble, gravel, sand, and some silt. The control building, emergency generator oil tank, and waste disposal are founded on basal till.

The bedrock at the site is thinly layered with light feldspathic and dark biotitic and hornblendic layers. The foliation is well defined and exhibits a consistent northwest trend. The unconfined compressive strength of the bedrock varies from 4,000 to 14,000 psi. All major structures founded on bedrock are listed in Table 2.5.

The laboratory tests included index property and grain-size distribution of onsite soils, moisture-density and direct shear testing of backfill, cyclic triaxial tests of beach sands, and unconfined compression testing of bedrock. The results of these tests are tabulated in FSAR Tables 2.5.4-10 through 12.

The geophysical surveys using both explosive and impact sources were conducted at the site to determine compressional "P" and shear "S" wave velocities of the underlying materials. The seismic velocities and shear moduli are given in Table 2.6.

2.5.4.1.4 Groundwater Conditions

The groundwater conditions are discussed in FSAR Section 2.4.12. A piezometric surface map showing the groundwater flow and the hydraulic gradient is presented in FSAR Figure 2.5.4-37. The staff's evaluation of groundwater is presented in Section 2.4 of this report.

2.5.4.2 Excavation and Backfill

Soil and rock-excavation were required to reach bearing elevations of seismic Category I structures at the site. The extent of excavations and backfill for major structures is shown on FSAR Figures 2.5.4-33 through -35.

Controlled rock-blasting techniques, including line drilling, cushion blasting, presplitting, and smooth-wall blasting, were used when blasting near the perimeter of structures to limit overbreak and minimize damage to adjacent rock. Rock bolts were installed in the southwest sector of the containment excavation to prevent potential sliding failures along the foliation. The inflow of water into the excavation was controlled by pumping from local sumps.

Some softening of the basal till in sections of the excavation was observed. The softening was attributed to the exposure of the till to the effects of weathering and construction traffic. The softened material was hand excavated to firm till and replaced with either fill concrete or compacted structural backfill.

The structural backfill was obtained from glacial outwash deposits located at the Romanella pit in North Stonington, Connecticut. All structural backfill was processed at the borrow pit to ensure that the gradation limits met the specification requirements. All backfill was compacted to 95% modified Proctor (ASTM D-1557, Method D). A continuing program of testing, inspection, and documentation was in effect during construction to ensure satisfactory placement of backfill. The backfill and its placement are acceptable to the staff.

2.5.4.3 Foundation Stability

2.5.4.3.1 Static Stability

The static foundation loads of major structures range from 1.6 to 8.25 ksf and are listed in FSAR Table 2.5.4-23. The applicant has calculated the factor of safety for bearing capacity to be in excess of 3. The calculated static settlements, essentially elastic displacements under the foundation loads, range from 0.01 to 0.22 in. (Table 2.5). However, the measured settlements during construction, presented in FSAR Figures 2.5.4-60 through -64, show some erratic movements. The applicant has attributed these movements to excavation rebound and has stated in the FSAR that the future settlements of these structures will be monitored periodically after construction until the rate of change of structure movement decreases. The staff concurs in the applicant's assessment that the factor of safety against bearing capacity failure is adequate and the after-construction settlement monitoring for structures founded on soils is acceptable.

2.5.4.3.2 Dynamic Loading

The SSE of 0.17 g and an OBE of 0.09 g in the horizontal direction, input at the bedrock surface, have been used as the design basis for seismic loading at the site. Further discussion of the design earthquake is presented in Section 2.5.2 of this report.

The lumped mass-spring methods were used for structures founded on rock to determine the dynamic response to seismic loadings. The staff evaluation is presented in Section 3.7.2 of this report.

The computer program PLAXLY-3 was used for structures founded on soils to determine the amplification effects of the soil-structure system. The computer program SHAKE was used to determine the strain-compatible soil properties. However, the applicant has not provided sufficient information for the staff to confirm the acceptability of the analyses. In a letter dated June 26, 1984, the applicant committed to perform additional analyses incorporating the as-built soil/foundation conditions to confirm these data.

Sliding stability resulting from seismic loading was analyzed for the service water pipe encasement in the FSAR (Amendment 8). The applicant has provided additional information to confirm that the service pipe encasement is stable unders seismic conditions. This is acceptable to the staff.

2.5.4.4 Liquefaction Potential

Most plant structures are founded on rock; therefore, liquefaction is not a concern for those structures.

For structures founded on either basal till or structural backfill, liquefaction potential of these foundation materials has been assessed by the applicant in the FSAR (Amendment 8). The staff concurs in the applicant's conclusion that both the compacted structural backfill and the dense basal till are not susceptible to liquefaction.

The dynamic response analysis of the shorefront sand deposits has been performed by the applicant to assess the potential amplification of ground motions applied to the bedrock surface. The calculated shear stresses were used to assess the liquefaction potential of the beach sand. A one-dimensional model was used to represent the beach sand deposits in the dynamic analysis. Since the beach sand deposits have thicknesses varing from a few feet to about 50 ft overlying the basal till, which dips at an angle toward the Long Island Sound, the one-dimensional idealization is not representative of the in situ conditions.

A two-dimensional analysis was performed by the applicant in May 1984. The results indicate that the beach sand deposits may be liquefiable under seismic conditions. The applicant will evaluate the effects of the liquefied beach sand to confirm that this will not have safety consequences.

2.5.4.5 Lateral Loads

The below-grade walls of structures were designed to resist both the static and dynamic pressure resulting from the surrounding earth and water. The value of the lateral earth pressure coefficient at rest used in the design is 0.5. The dynamic lateral earth pressure on the below-grade walls was determined in accordance with Seed and Whitman's procedure (1970). The procedure used to obtain the dynamic lateral earth pressure is in accordance with the state-of-the-art methods required by the Standard Review Plan (NUREG-0800) and is acceptable. Although the design lateral earth pressure at rest is low, the applicant has provided additional analyses to confirm that the as-built below-grade walls possess adequate safety margin against at-rest later earth pressure coefficient up to 0.7 and is, therefore, acceptable to the staff.

2.5.4.6 Conclusion

On the basis of the applicant's design criteria and construction specifications and on the results of the applicant's investigations, laboratory and field tests, and analyses, the staff has concluded that the plant foundation will be adequate to safely support the plant structures if the following can be confirmed:

(1) The input parameters used in the soil-structure interaction analyses for the emergency diesel generator enclosure will be acceptable. (Section 2.5.4.3.2)

(2) The liquefaction of the beach sand deposits will not impair the function of the pumphouse. (Section 2.5.4.4)

2.5.5 Stability of Slopes

The topography in the plant area is generally flat with the final grade at el 24 ft in the major plant area. There are two manmade slopes, the beach sand and armor stone slope at the shoreline and the vertical rock cut excavated for the containment structure.

2.5.5.1 Shoreline Slope

A plan of the shoreline in the vicinity of the Millstone Unit 3 pumphouse is shown on FSAR Figure 2.5.4-41. To the east of the pumphouse, a reinforced concrete seawall with post-tensioned rock anchors has been built between the pumphouse and the Millstone Unit 2 intake structure to retain the earth and protect the structure from wave action. To the west of the pumphouse a counterfoot retaining wall has been built to retain the earth and a variable slope has been cut in the beach sand to provide for a transition from the offshore intake channel at el -32 ft to the pumphouse area at el 14 ft. The slope varies from 5 horizontal to 1 vertical (5H:1V) immediately adjacent to the pumphouse to 10 horizontal to 1 vertical (10H:1V) near Bay Point, the western extent of the beach. Compacted backfill was placed in areas where additional fill was required to meet these grade requirements. A multilayer stone armor zone was placed on the slope for protection against wave action. FSAR Figure 2.5.5-1 shows a detailed cross-section of the slope protection system. Stone for the slope protection was obtained from bedrock previously blasted during excavation at the site and from offsite sources.

There were 22 borings drilled in the vicinity of the pumphouse. A geologic profile across this area is shown on FSAR Figure 2.5.4-35. The depth of sand along the beach varies from zero at Bay Point to a maximum of approximately 40 ft in the vicinity of the pumphouse. The beach sand deposits overlie a thin layer of basal till, generally less than 5 ft thick, which covers the bedrock. Properties of the beach sand are shown in FSAR Table 2.5.4-12 and Appendices 2.5F and G.

The stability of the shoreline slope was analyzed by the applicant using computer program LEASE II (Limiting Equilibrium Analysis of Slopes and Embankments). Both static and dynamic loads were considered in the analysis. The calculated factor of safety against a slope failure through the 5H:1V section of shoreline slope is 1.44 under the seismic loading condition.

However, the staff has found that the as-built and subsurface conditions of the slope are not compatible with those used in the applicant's original analyses.

The applicant performed a two-dimensional dynamic response analysis in May 1984, and the results indicate that the beach sand deposits may not be stable under seismic condition. The applicant will evaluate the effect of the lateral movements of the beach sand to confirm that this will not impair the safety function of the pumphouse. The applicant will also provide additional loading information about the retaining wall design. The staff will provide its evaluation in a future SER supplement.

2.5.5.2 Containment Rock Cut

The containment building is founded on bedrock at approximately el -39 ft. Top of rock varies from approximately el 0 ft to el 20 ft. The excavation walls are vertical, with a 9-in. bench at el -17 ft.

During construction, detailed geologic mapping has uncovered some additional preferred joint sets, shown on FSAR Figure 2.5.5-2. The failure in the bedrock along these joints and foliation surfaces could affect the lateral loading of the containment structure.

The applicant developed two computer programs to evaluate field data and compute the stability of rock slopes. A ring beam was designed and constructed to transfer the rock load around the excavation, maintaining the isolation of containment structure from these external loads. The staff has reviewed the information provided by the applicant in a letter dated March 27, 1984, concerning the rock loads used in the ring beam design and found that the rock loads used are acceptable.

2.5.5.3 Conclusion

On the basis of the applicant's information presented in the FSAR through Amendment 8, the staff confirms that the design of the ring beam is adequate. However, the effect of (1) the potentially unstable beach sands during an earthquake and (2) the retaining wall design must be confirmed.

Figure 2.1 General vicinity of the Millstone Nuclear Power Plant, Unit 3
Source: FSAR Figure 2.1-2

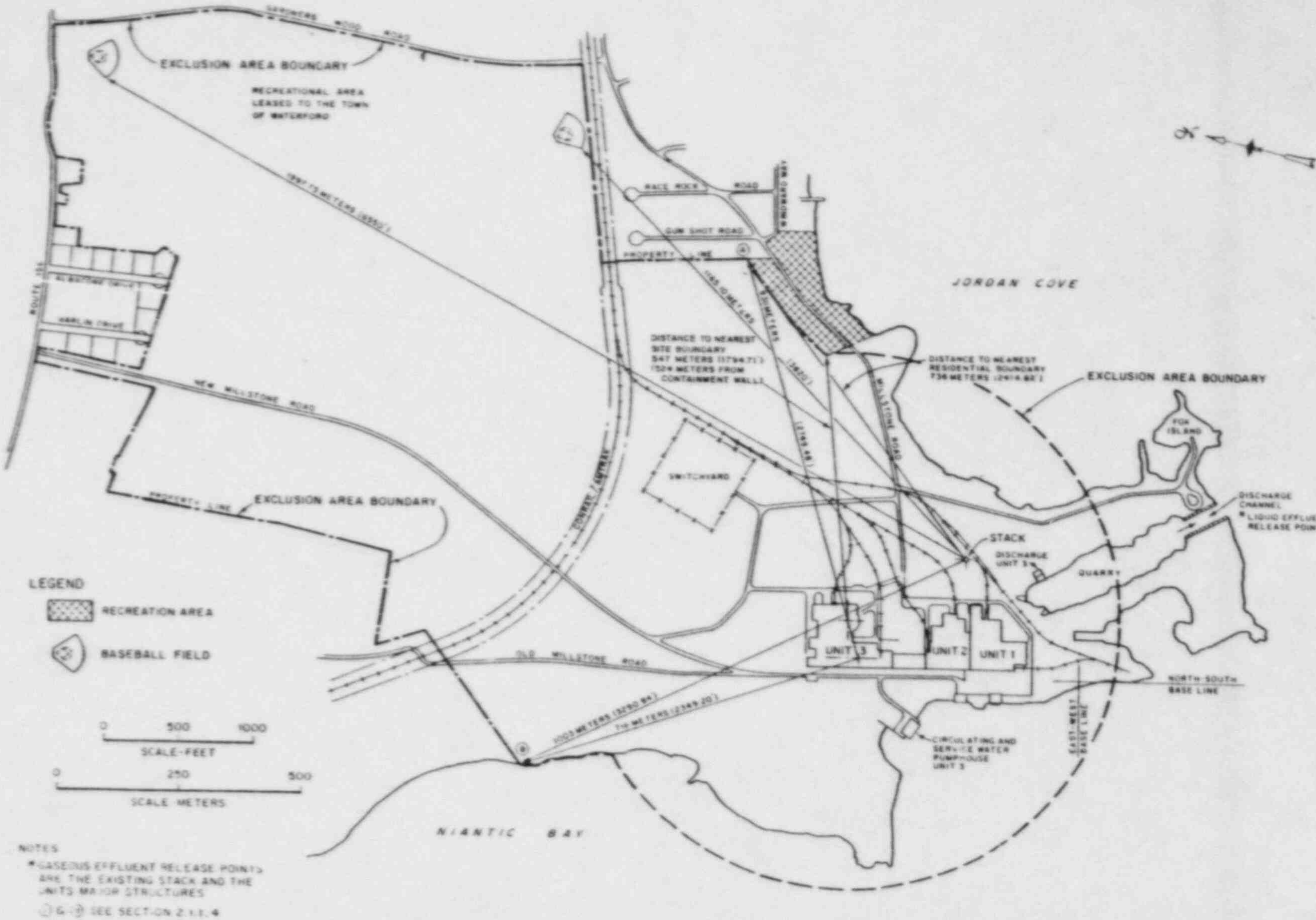

Figure 2.2 Site layout
Source: ER-OL Figure 2.1-3, Amendment 5, Jan. 1984

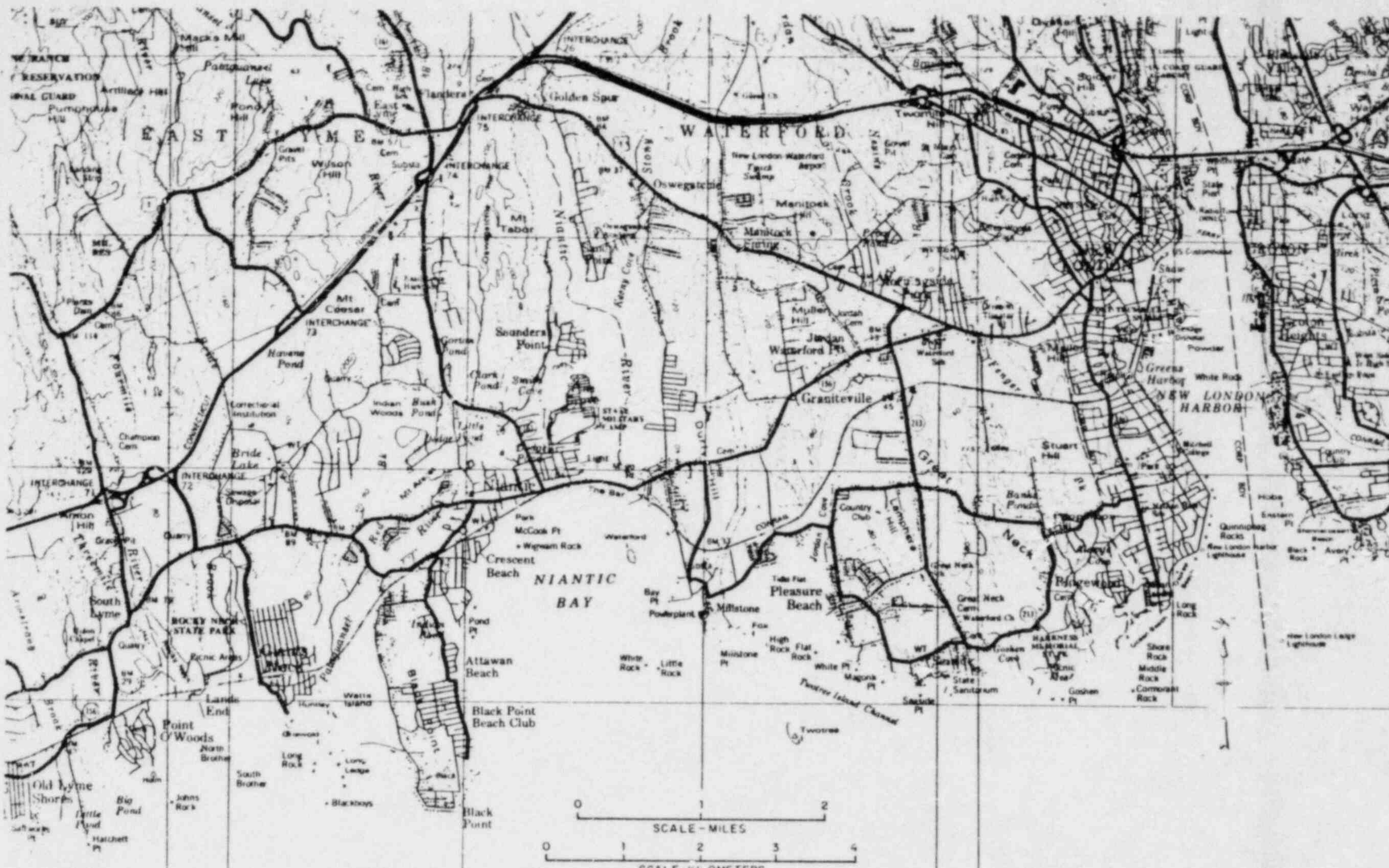

Figure 2.3 New London County - state highways and town roads
Source: FSAR Figure 2.2-4

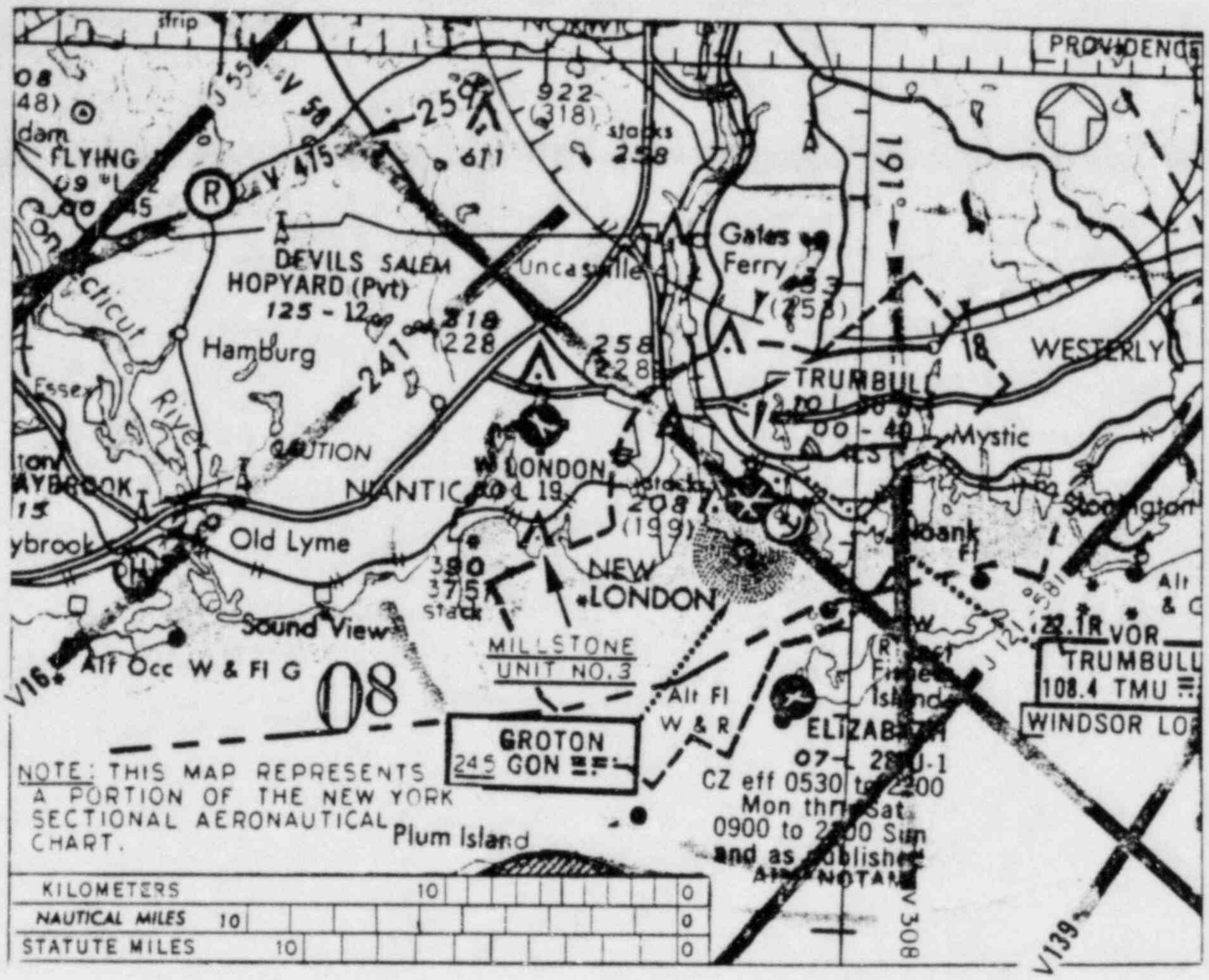

Figure 2.4 Air lanes adjacent to Millstone Point
Source: FSAR Figure 2.2-3

Table 2.1 Resident population in the vicinity of Millstone site

	Distance from plant			
Year	0-2 mi	2-5 mi	5-10 mi	0-10 mi
1980	5,122	43,111	61,204	109,437
1982*	5,202	47,601	61,539	114,342
2000	5,348	44,483	66,730	116,561
2030	5,543	55,255	73,595	134,393

*Staff estimates.

Table 2.2 Meteorological measurements at Millstone

Elevation* (ft)	Measurements
447	Wind speed Wind direction and variance Air temperature Dew point temperature Temperature difference to 10-m level
374	Wind speed Wind direction Temperature difference to 10-m level
142	Wind speed wind direction Temperature difference to 10-m level
64	Air temperature Dew point temperature
33	Wind speed Wind direction and variance Air temperature Dew point temperature
14	Visibility
5	Solar radiation**

*Above base of tower at 15 ft mean sea level.
**Mounted on a platform to south of tower.

Table 2.3 Millstone relative concentration values

Time period	χ/Q (sec/m^3)
0-8 hours	2.7×10^{-5}
8-24 hours	1.9×10^{-5}
1-4 days	8.4×10^{-6}
4-30 days	2.7×10^{-6}

Table 2.4 Millstone maximum annual average relative concentration* and deposition

Location	χ/Q (sec/m^3)	D/Q (1/m^2)
Nearest goat		
NNE 1.5 mi	6.8×10^{-7}	3.1×10^{-9}
ENE 2 mi		3.4×10^{-9}
Cow		
WNW 4.5 mi	6.4×10^{-8}	2.3×10^{-10}
Residence		
ENE 0.52 mi	9.4×10^{-6}	7.4×10^{-8}
Vegetable garden		
ENE 0.52 mi	9.4×10^{-6}	7.4×10^{-8}
Site boundary		
SSW 0.39 mi	2.0×10^{-5}	9.2×10^{-8}
ENE, SE		1.2×10^{-7}

*Undecayed, undepleted.

Table 2.5 Foundation settlement data for major structure

Structure	Foundation Bearing Load (psf)	Enbedment Depth (ft)	Founding Material	Average Thickness Till (ft)	Average Thickness Structural Fill (ft)	Dimensions of Foundation Material (ft)	Foundation Type	Maximum Calculated Static Settlement (in)
Containment	8,000	62.7	Rock	-	-	158 diameter	Mat	0.04
Main Steam Valve	5,000	15.0	Rock	-	-	70 × 60	Mat	0.01
Auxiliary	5,000	24.5	Rock	-	-	177 × 102	Mat	0.02
Engineered Safety Features	3,500	24.5	Rock	-	-	139 × 23	Mat	0.01
Control	3,500	24.5	Till	0 to 10	-	125 × 105	Mat	0.02 to 0.03
Emergency Generator Enclosure	1,600	15.0	Structural Fill	10	20	65 × 72	Mat and Strip Footings	0.22
Emergency Generator Oil Tank	1,600	22.5	Till	10	4	65 × 32	Mat	0.01
Refueling Water Storage Tank	4,000	9.0	Rock	-	-	45 diameter	Mat	less than 0.01
Demineralized Water Storage Tank	4,000	9.5	Rock	-	-	35 diameter	Mat	less than 0.01
Fuel	5,500	21.0	Rock		-	93 × 112	Mat	less than 0.01
Waste Disposal (Liquid)	3,500	23.5	Till	2 to 8	-	112 × 36	Mat	0.04
Hydrogen Recombiner	3,000	4.0	Concrete Fill	-	13	39 × 27	Mat	less than 0.01

Source: FSAR Table 2.5.4-14

Table 2.6 Seismic velocities and shear moduli

Material	P-wave (fps)	S wave (fps)	Shear* modulus G (psi)	Poisson's* ratio
Fill	1,363-3,060	814-1,238		
Alluvium	4,820-5,818	383-684		
Ablation till	6,033-6,597	398-654	9×10^3	0.49
Basal till	7,539-7,603	1,246-2,387	1.4×10^5	0.44
Rock	12,800-13,500	6,500	1.5×10^6	0.33

*Moduli used in the response analyses of structures.

3 DESIGN CRITERIA FOR STRUCTURES, SYSTEMS, AND COMPONENTS

3.1 General

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

The staff 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 are cited in this report and have been previously reviewed by the staff, found acceptable, and incorporated into the SRP (NUREG-0800).

3.2 Classification of Structures, Systems, and Components

3.2.1 Seismic Classification

GDC 2 requires that nuclear power plant structures, systems, and components important to safety be designed to wthstand the effects of earthquakes without loss of capability to perform their safety function. Certain of these plant features are necessary to ensure (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, and (3) the capability to prevent or mitigate the consequences of accidents that could result in potential offsite exposures comparable to 10 CFR 100 guidelines exposures.

The earthquake for which these safety-related plant features are designed is defined as the safe shutdown earthquake (SSE) in 10 CFR 100, Appendix A. The SSE is based on an evaluation of the maximum earthquake potential and is that earthquake that produces the maximum vibratory ground motion for which structures, systems, and components are designed to remain functional. Those plant features that are designed to remain functional if an SSE occurs are designated seismic Category I in RG 1.29, "Seismic Design Classification." RG 1.29 is the principal document used in the NRC staff review for identifying those plant features that, as a minimum, should be designed to seismic Category I requirements. Millstone Unit 3 was reviewed in accordance with SRP Section 3.2.1.

The structures, systems, components, and equipment of Millstone Unit 3 that are required to be designed to withstand the effects of an SSE and remain functional have been identified in an acceptable manner in FSAR Table 3.2-1. This table, in part, identifies the major components in fluid systems, mechanical systems, and associated structures designated as seismic Category I. In addition, piping and instrumentation diagrams in the FSAR identify the interconnecting piping and valves and the boundary limits of each system classified as seismic Category I.

The staff has reviewed Table 3.2-1 and the fluid system piping and instrumentation diagrams and concludes that the structures, systems, and components of Millstone Unit 3 have been properly classified as seismic Category I items in conformance with RG 1.29, Revision 3.

In its review of FSAR Section 3.9, the staff confirmed that acceptable design interfaces exist between seismic Category I and nonseismic portions of piping systems. All other structures, systems, and components that may be required for operation of the facility are not required to be designed to seismic Category I requirements, including those portions of Category I systems such as vent lines, fill lines, drain lines, and test lines on the downstream side of isolation valves and portions of these systems that are not required to perform a safety function.

The staff concludes that the structures, systems, and components of Millstone Unit 3 are properly classified as seismic Category I items in accordance with RG 1.29. This constitutes an acceptable basis for satisfying, in part, the requirements of GDC 2.

3.2.2 System Quality Group Classification

GDC 1 requires that nuclear power plant systems and components important to safety be designed, fabricated, erected, and tested to quality standards commensurate with the importance of the safety function to be performed. These pressure-retaining components of fluid systems are part of the reactor coolant pressure boundary (RCPB) and other fluid systems important to safety, where reliance is placed on these systems to (1) prevent or mitigate the consequences of accidents and malfunctions originating within the RCPB, (2) permit shutdown of the reactor and maintain it in a safe shutdown condition, and (3) retain radioactive material. RG 1.26, "Quality Group Classification and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," is the principal document used in the NRC staff review for identifying on a functional basis the components of those systems important to safety as NRC Quality Groups A, B, C, or D. 10 CFR 50.55a identifies those American Society of Mechanical Engineers (ASME) Section III, Class 1 components that are part of the RCPB.

Conformance of these RCPB components with 10 CFR 50.55a is discussed in Section 5.2.1.1 of this report. These RCPB components are designated in RG 1.26 as Quality Group A. Certain other RCPB components that meet the exclusion requirement of Footnote 2 of 10 CFR 50.55a are classified Quality Group B in accordance with RG 1.26. Millstone Unit 3 was reviewed in accordance with SRP Section 3.2.2.

The applicant used the American Nuclear Society (ANS) Safety Classes 1, 2, 3, and non-nuclear safety (NNS) as defined in American National Standards Institute (ANSI) N18.2-1973, "Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," and ANSI N18.2a-1975, "American National Standard Revision and Addendum to Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants," in the classification of system components as an alternative acceptable method of meeting the guidance of RG 1.26. Safety Classes 1, 2, 3, and NNS correspond to the Commission's Quality Groups A, B, C, and D in RG 1.26.

The relationship of the NRC quality groups and ANS safety classes can be summarized as follows:

NRC quality group	Millstone 3 PWR safety class
A	1
B	2
C	3
D	NNS

The staff has reviewed the use of ANS safety classes in FSAR Table 3.2-1 and finds the classification of components acceptable. Quality Group A (Safety Class 1) components of the RCPB are constructed* in accordance with ASME Code, Section III, Division 1, Class 1. Components in fluid systems that are classified Quality Group B (Safety Class 2) are constructed in accordance with ASME Code, Section III, Division 1, Class 2. Components in fluid systems that are classified Quality Group C (Safety Class 3) are constructed in accordance with ASME Code, Section III, Division 1, Class 3. The NRC staff finds the codes and standards used in the construction of components acceptable.

The safety-related systems and components that are important to safety have been identified in an acceptable manner in FSAR Table 3.2-1. As noted above this table, in part, identifies major components in fluid systems (such as pressure vessels, heat exchangers, storage tanks, pumps, piping, and valves) and in mechnaical systems (such as cranes, refueling platforms, and other miscellaneous handling equipment). In addition, piping and instrumentation diagrams in the FSAR identify the classification boundaries of interconnecting piping and valves. The staff has reviewed FSAR Table 3.2-1 and the fluid system piping and instrumentation diagrams and concludes that pressure-retaining components have been properly classified in conformance with RG 1.26, Revision 3.

The staff concludes that the construction of components in fluid systems identified in FSAR Table 3.2-1 is in conformance with the ASME Code and industry standards, the Commission's regulations, and the guidance provided in RG 1.26. This provides assurance that component quality is commensurate with the importance of the safety function of these systems and constitutes an acceptable basis for satisfying the requirements of GDC 1.

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*Constructed, as used herein, is an all-inclusive term comprising materials certification, design, fabrication, examination, testing, inspection, and certification required in the manufacture and installation of components.

3.3 Wind and Tornado Loadings

3.3.1 Wind Design Criteria

All Category I structures exposed to wind forces were designed to withstand the effects of the design wind. The design wind specified has a velocity of 115 mph based on the wind experienced at the Millstone site in a 1960 hurricane.

The procedures that were used to transform the wind velocity into pressure loadings on structures and the associated vertical distribution of wind pressures and gust factors are in accordance with ANSI A58.1 and American Society of Civil Engineers (ASCE) Paper 3269, "Wind Forces on Structures," as appropriate. These documents are acceptable to the staff.

The staff concludes that the plant design is acceptable and meets the recommendations of SRP Section 3.3.1 and the requirements of GDC 2 with respect to the capability of the structures to withstand design wind loading so that their design reflects

(1) appropriate consideration for the most severe wind recorded for the site with an appropriate margin

(2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena

(3) the importance of the safety function to be performed

The applicant has met these requirements by using ANSI A58.1 and ASCE Paper 3269, which the staff has reviewed and found acceptable, to transform the wind velocity into an effective pressure on structures and for selecting pressure coefficients corresponding to the structures' geometry and physical configuration.

The applicant has designed the plant structures with sufficient margin to prevent structural damage during the most severe wind loadings that have been determined appropriate for the site so that the requirements of Item 1 listed above are met. In addition, the design of seismic Category I structures, as required by Item 2 listed above, has included, in an acceptable manner, load combinations that occur as a result of the most severe wind load and the loads resulting from normal and accident conditions.

The procedures used to determine the loadings on structures induced by the design wind specified for the plant are acceptable because these procedures have been used in the design of conventional structures and been proven to provide a conservative basis that, together with other engineering design considerations, ensures that the structures will withstand such environmental forces. The use of these procedures provides reasonable assurance that, in the event of design-basis winds, the structural integrity of the plant structures that have to be designed for the design wind will not be impaired. As a result, safety-related systems and components located within these structures are adequately protected and will perform their intended safety functions if needed. Thus, the requirement of Item 3 above is satisfied.

3.3.2 Tornado Design Criteria

All Category I structures exposed to tornado forces and needed for the safe shutdown of the plant were designed to resist a tornado of 290-mph tangential wind velocity and a 70-mph translational wind velocity. The simultaneous atmospheric pressure drop was assumed to be 3 psi in 1.5 sec. Tornado missiles also are considered in the design as discussed in Section 3.5 of this report.

The staff concludes that the applicant has met the recommendations of SRP Section 3.3.2 and the requirements of GDC 2 with respect to the structures' capability to withstand design tornado wind loading and tornado missiles so that their design reflects

(1) appropriate consideration for the most severe tornado recorded for the site with an appropriate margin

(2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena

(3) the importance of the safety function to be performed

The applicant has met these requirements by using ANSI A58.1 and ASCE Paper 3269, which the staff has reviewed and found acceptable, to transform the wind velocity generated by the tornado into an effective pressure on structures and for selecting pressure coefficients corresponding to the structures' geometry and physical configuration.

The applicant has designed the plant structures with sufficient margin to prevent structural damage during the most severe tornado loadings that have been determined appropriate for the site so that the requirements of Item 1 listed above are met. In addition, the design of seismic Category I structures, as required by Item 2 listed above, has included, in an acceptable manner, load combinations that occur as a result of the most severe tornado wind load and the loads resulting from normal and accident conditions.

The procedures used to determine the loadings on structures induced by the design-basis tornado specified for the plant are acceptable because these procedures have been used in the design of conventional structures and proven to provide a conservative basis that, together with other engineering design considerations, ensures 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 tornados will not be impaired. As a result, safety-related systems and components located within these structures will be adequately protected and may be expected to perform necessary safety functions, as required. Thus, the requirement of Item 3 listed above is satisfied.

3.4 Water Level (Flood) Design

3.4.1 Flood Protection

The design of the facility for flood protection was reviewed in accordance with SRP Section 3.4.1 (NUREG-0800).

To ensure conformance with the requirements of GDC 2, the staff's review of the overall plant flood protection design included all systems and components whose failure as a result of flooding could prevent safe shutdown of the plant or result in uncontrolled release of significant radioactivity.

The applicant has sited the plant on Millstone Point where Millstone Units 1 and 2 are already located. There are no major rivers or streams in the vicinity of Millstone Point. The design-basis flood (maximum combination of storm surge and wave runup) established for Millstone Unit 3 is el 23.8 ft mean sea level (MSL), and the maximum still water level is el 19.7 ft. All of the unit's safety-related structures and equipment, except the circulating and service water pumphouse, are protected from flooding by the site grade of el 24 ft MSL.

Each pair of service water pumps and their motors are located at el 14.5 ft MSL inside individual watertight cubicles in the seismically designed pumphouse. The walls of those cubicles are watertight up to el 25.5 ft MSL. Therefore, the pump motors and associated electrical equipment are protected from wave action and probable maximum hurricane (PMH) surge. The applicant has shown that the possibility of water entering the pumphouse through the pump shaftways and rendering the pumps inoperable is not credible.

All access openings to safety-related structures and facilities are at an elevation of 24 ft 6 in. which is above the nominal site grade elevation of 24 ft (except two access openings to the service water cubicles inside the pumphouse) and are consequently protected from flooding resulting from groundwater, storm surge, and direct rainfall. The two access openings to the service water cubicles are at el 23.8 ft MSL and are fitted with watertight steel doors capable of withstanding the maximum hydrostatic loads occurring at their respective locations. Pumphouse roof ventilators are weatherproof and are located above the 100-year accumulated snow depth of 52 in. Equipment access openings on the service water pumphouse roof are fitted with watertight covers.

Foundations of safety-related structures are constructed of reinforced concrete. All subgrade joints between walls and slabs are protected with waterstops cast in concrete.

The storm drain system uses catch basins and underground conduits and/or drainage ditches to convey runoff to Niantic Bay. Flooding of safety-related buildings during maximum precipitation will be prevented because of grade elevation and because access openings are located 6 in. above the nominal grade elevation.

The applicant stated that nonseismic Category I tanks and vessels in safety-related structures do not contain sufficient inventory to cause flooding of safety-related equipment resulting from a hypothetical worst-case single tank or vessel failure. In addition, safety-related equipment required for safe shutdown of the plant is located in cubicles or on elevated platforms, which would preclude damage as a result of flooding resulting from postulated failures of nonseismic Category I tanks or vessels during a seismic event.

Internal flooding resulting from postulated piping failures was developed by considering the worst-case fluid release from a single piping failure according to Branch Technical Position ASB 3-1, Section 3.3.a. Tanks outside safety-related structures are located in areas that would preclude flooding of safety-related equipment.

On the basis of its review, the staff concludes that the design of the facility provides adequate protection against flooding as a result of probable maximum precipitation (PMP). Compliance with the guidelines of RGs 1.59 and 1.102 is discussed in Section 2.4 of this SER. The plant is protected from flooding as a result of high- or moderate-energy-line breaks within a structure. Therefore, the staff concludes that the plant design complies with the requirements of GDC 2. The flood protection meets the acceptance criteria contained in SRP Section 3.4.1.

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 Section 2.4 of this report. The hydrostatic effect of the flood was considered in the design of all Category I structures exposed to the water head.

The staff concludes that the plant design is acceptable and meets the recommendations of SRP Section 3.4.2 and the requirements of GDC 2 with respect to the structures' capability to withstand the effects of the flood or highest groundwater level so that their design reflects

(1) appropriate consideration for the most severe flood recorded for the site with an appropriate margin

(2) appropriate combinations of the effects of normal and accident conditions with the effects of the natural phenomena

(3) the importance of the safety function to be performed

The applicant has designed the plant structures with sufficient margin to prevent structural damage during the most severe flood or groundwater and the associated dynamic effects that have been determined appropriate for the site so that the requirements of Item 1 listed above are met. In addition, the design of seismic Category I structures, as required by Item 2 above, has included, in an acceptable manner, load combinations that occur as a result of the most severe flood or groundwater-related loads and the loads resulting from normal and accident conditions.

The procedures used to determine the loads on seismic Category I structures induced by the design flood or highest groundwater level specified for the plant are acceptable because these procedures have been used in the design of conventional structures and proven to provide a conservative basis that, together with other engineering design considerations, ensures that the structures will withstand such environmental forces.

The use of these procedures provides reasonable assurance that, in the event of floods or high groundwater levels, the structural integrity of the plant seismic Category I structures will not be imparied. As a result, seismic Category I systems and components within these structures will be adequately protected and may be expected to perform necessary safety functions, as required. Thus, the requirement of Item 3 listed above is satisfied.

3.5 Missile Protection

3.5.1 Missile Selection and Description

3.5.1.1 Internally Generated Missiles (Outside Containment)

The design of the facility for providing protection from internally generated missiles outside containment was reviewed in accordance with SRP Section 3.5.1.1 (NUREG-0800). Conformance with the acceptance criteria, except as noted below, formed the basis for the staff evaluation of the design of the facility for providing protection from internally generated missiles outside containment with respect to the applicable regulations of 10 CFR 50.

The acceptance criteria for the design of the facility for providing missile protection includes meeting RG 1.115. The review of turbine missiles is discussed separately in Section 3.5.1.3 of this report.

GDC 4 requires protection of those plant structures, systems, and components whose failure could lead to offsite radiological consequences or that are required for safe plant shutdown against postulated missiles associated with plant operation. The missiles considered in this evaluation include those missiles generated by rotating or pressurized (high-energy fluid system) equipment.

Protection is provided by any one or a combination of the following: compartmentalization, barriers, separation, orientation, and equipment design. The primary means of providing protection to safety-related equipment from damage resulting from internally generated missiles is through the use of plant physical arrangement. Safety-related systems are physically separated from non-safety-related systems, and components of safety-related systems are physically separated from their redundant components. Stored spent fuel in the fuel building is protected by the fuel pool walls from damage by internal missiles that could result in radioactive release. The spent fuel is also protected by not locating any high-energy piping system or rotating machinery in the vicinity of the fuel. The applicant has identified the safety-related structures, systems, and components (SSCs) outside containment that are required for safe shutdown and has evaluated the potential for internally generated missiles affecting these safety-related SSCs. Pressurized system equipment and rotating components were evaluated as potential missile sources. The evaluation identified valves, pressurized gas bottles, accumulators, and instrument wells in high-energy systems as potential missiles. The applicant's evaluation verified that plant design features, such as walls or separation of redundant systems, will prevent these missiles from causing adverse effects on safety-related systems and components.

Pumps and fans outside the containment were evaluated to determine their potential for missile generation as a result of a failure or an overspeed condition. Regarding pumps and fans other than the turbine-driven auxiliary feedwater (AFW) pump, the applicant stated that no credible missiles are postulated because the maximum no-load speed is equivalent to the maximum operating speed of the motors; therefore, a pipe break or single failure could not result in a speed in excess of the no-load condition (no overspeed). On the basis of recent experience with fan fracture resulting in a missile, the staff requires further justification from the applicant that adequate protection is provided. In response to the staff request, the applicant stated he will provide the results of an analysis showing that the potential missiles from the turbine-driven AFW pump will not damage other safety-related equipment. On the basis of the results of the analysis, any required inside barriers will be installed.

The rod control motor generator sets were evaluated for potential flywheel missile generation. It was determined that because of material and fabrication standards, nondestructive testing, design stresses, and the speed and torque limitations on the 1,800-rpm induction motor, it is not credible for the motor generator sets to generate missiles.

The applicant evaluated the potential for gravitational missiles. All non-safety-related components are supported to prevent their collapse in a safe shutdown earthquake, or if they cannot be properly supported, they will be removed during plant operation so as to prevent them from becoming gravitational missiles.

On the basis of this review, the staff cannot conclude that the design conforms with GDC 4 as it relates to protection against internally generated missiles assuming a single active failure until the applicant provides additional information. The staff will report resolution of this item in a supplement to this SER.

3.5.1.2 Internally Generated Missiles (Inside Containment)

The design at the facility for providing protection from internally generated missiles inside containment was reviewed in accordance with SRP Section 3.5.1.2 (NUREG-0800).

Plant structures, systems, and components inside containment whose failure could lead to offsite radiological consequences or that are required for safe plant shutdown must be protected against the effects of internally generated missiles in accordance with the requirements of GDC 4. Potential missiles that could be generated inside containment result from failures of rotating components, pressurized component (high-energy fluid system) failures, and gravitational effects.

The applicant's analysis of potential missiles indicates that the catastrophic failure of the reactor vessel, steam generators, pressurizer, and reactor coolant pump casings that would lead to the generation of missiles is not considered credible because of the material characteristics, inspections, conservative design, and quality control during fabrication and erection. Nuts and bolts were of negligible concern because of the small amount of energy contained in them. The reactor coolant pump flywheel is not considered a credible source of missiles for the reasons presented in Section 5.4.1.

The applicant evaluated the potential of the control rod drive mechanism (CRDM) for becoming a missile as a result of gross failure of the CRDM housing and concluded that the CRDM was not a credible missile source because of its material properties and the conservative design as demonstrated by hydrostatic testing. This analysis also considered the consequences should the CRDM top plug become loose and be ejected. The control rod cluster would travel upward until it would impact the upper support plate. This impact would cause a fracture that would allow the drive shaft to continue accelerating upward until it hit the missile shield, thus preventing it from affecting safety-related equipment.

Pumps and fans inside containment were evaluated to determine their potential for missile generation as a result of a failure or an overspeed condition. The applicant stated that no credible missiles are postulated because the maximum no-load speed is equivalent to the maximum operating speed of the motors; therefore, a pipe break or single failure could not result in a speed in excess of the no-load condition (no overspeed). The staff requires further justification from the applicant that adequate protection is provided.

The applicant also considered the following as sources of potential missiles from pressurized high-energy fluid systems: control rod drive mechanism housing plug and drive shaft, valves, temperature and pressure sensor assemblies, pressurizer heaters, pressurized gas bottles, and accumulators. The applicant performed analyses that demonstrate that the designs of the above components either prevent the generation of missiles as a result of a single failure or, if generated, the missiles have insufficient energy to cause unacceptable damage or compartmentalization, separation, or barriers are adequate to provide protection of safety-related equipment.

In addition, the applicant evaluated the potential for gravitational missiles inside containment. All nonsafety-related components are supported to prevent their collapse in an SSE.

On the basis of its review, the staff cannot conclude that the design is in conformance with GDC 4 as it relates to protection against internally generated missiles, assuming a single active failure until the applicant provides additional information. The staff will report resolution of this item in a supplement to this SER.

3.5.1.3 Turbine Missiles

3.5.1.3.1 Review Basis

During the past several years the results of turbine inspections at operating nuclear facilities have indicated that cracking to various degrees has occurred at the inner radius of turbine discs, particularly those of Westinghouse design. Within this time period, there has actually been a Westinghouse disc failure at one facility owned by the Yankee Atomic Electric Company. Furthermore, recent inspections of General Electric turbines have also resulted in the discovery of disc keyway cracks. Stress corrosion has been identified by both manufacturers as the operative cracking mechanism.

The staff has followed these developments closely. Its primary safety objective is the prevention of unacceptable doses to the public from releases of radioactive contaminants that could be caused by damage to plant safety-related structures, systems, and components as a result of missile-generating turbine failures. On the basis of previous staff reviews and various estimates by others (Bush, 1973; Twisdale, Dunn, and Frank, 1982) for a variety of plant layouts, the staff concludes that if a turbine missile is generated the probability of unacceptable damage to safety-related structures, systems, and components is in the neighborhood of 10^{-3} or 10^{-2} per year depending on whether the turbine orientation is favorable or unfavorable. In view of this and operating experience, the staff has shifted the review emphasis to the prevention of missile-generating turbine failures. In keeping with this shift of emphasis, the staff has recently set turbine missile generation probability guidelines for determining (1) turbine disc ultrasonic inservice inspection frequencies and (2) turbine control and overspeed protection system's maintenance and testing schedules. It should be noted that (1) no change in safety criteria is associated with this change in review emphasis and (2) the major domestic turbine manufacturers are already in the process of establishing models and methods for calculating turbine missile generation probabilities for their respective turbine generator systems.

This shift of emphasis helps improve turbine generator system reliability by focusing on review and evaluation of the probability of missile-generating turbine failure and, in the process, provides a logically consistent method for establishing inservice inspection and testing schedules. Furthermore, it reduces considerably the analytical burden placed on licensees by eliminating the need for elaborate and ambiguous analyses of strike and damage probabilities and, at the same time, better ensures the protection of public health and safety by better maintaining turbine system integrity.

According to GDC 4, nuclear power plant structures, systems, and components important to safety shall be appropriately protected against dynamic effects, including the effects of missiles. Failures of large steam turbines of the main turbine generator have the potential for ejecting large high-energy missiles that can damage plant structures, systems, and components. The overall safety objective of the staff is to ensure that structures, systems, and components important to safety are adequately protected from potential turbine missiles. Of those systems important to safety, this topic is primarily concerned with safety-related systems, that is, those structures, systems, or components necessary to perform required safety functions and to ensure

(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

(3) the capability to prevent accidents that could result in potential offsite exposures that are a significant fraction of the guideline exposures of 10 CFR 100, "Reactor Site Criteria"

Typical safety-related systems are listed in RG 1.117.

The probability of unacceptable damage resulting from turbine missiles (P_4) is generally expressed as the product of (1) the probability of turbine failure resulting in the ejection of turbine disc (or internal structure) fragments through the turbine casing (P_1), (2) the probability of ejected missiles perforating intervening barriers and striking safety-related structures, systems, or components (P_2), and (3) the probability of struck structures, systems, or components failing to perform their safety function (P_3).

According to NRC guidelines in SRP Section 2.2.3 (NUREG-0800)and RG 1.115, the probability of unacceptable damage from turbine missiles should be less than or equal to about 1 chance in 10 million per year for an individual plant, that is, $P_4 \leq 10^{-7}$ per year.

In the past, analyses for CP and OL reviews assumed the probability of missile generation (P_1) to be approximately 10^{-4} per turbine-year, based on the historical failure rate (Bush, 1973). The strike probability (P_2) was estimated (SRP Section 3.5.1.3) on the basis of postulated missile sizes, shapes, and energies and on available plant-specific information such as turbine placement and orientation, number and type of intervening barriers, target geometry, and potential missile trajectories. The damage probability (P_3) was generally assumed to be 1.0. The overall probability of unacceptable damage to safety-related systems (P_4), which is the sum over all targets of the product of these probabilities, was then evaluated for compliance with the NRC safety objective. This logic places the regulatory emphasis on the strike probability; that is having established an individual plant safety objective of about 10^{-7} per year, or less, for the probability of unacceptable damage to safety-related systems resulting from turbine missiles, this procedure requires that P_2 be less than or equal to 10^{-3}.

It is well known that nuclear turbine discs crack (NUREG/CR-1884; Northern States Utilities, 1981) and that disc ruptures can result in the generation of high-energy missiles (Kalderon, 1972). Furthermore, analyses (Burns 1977; Clark, Seth, and Shaffer, 1981) clearly demonstrate the large effects of inservice testing and inspection frequencies on missile generation probabilities (P_1). It is the staff's view that sufficiently frequent turbine testing and inspection are the best means of ensuring that the criteria on the probability of unacceptable damages to safety-related structures, systems, and components (P_4) are met. Therefore, it is prudent for turbine manufacturers to perform, and the NRC to review, analyses of turbine reliability that include known and likely failure mechanisms expressed as a function of time (i.e., inservice inspection or test intervals).

Although the calculation of strike probability is not difficult in principle, for the most part reducing to a straightforward ballistics analysis, it presents a problem in practice. The problem stems from the fact that numerous modeling approximations and simplifying assumptions are required to make tractable the incorporation into acceptable models of available data on the

(1) properties of missiles, (2) interactions of missiles with barriers and obstacles, (3) trajectories of missiles as they interact with and perforate (or are deflected by) barriers, and (4) identification and location of safety-related targets. The particular approximations and assumptions made tend to have a large effect on the resulting value of P_2. Similarly, a reasonably accurate specification of the damage probability (P_3) is not a simple matter because of the difficulty of defining the missile impact energy required to render given safety-related systems unavailable to perform their safety function and the difficulty of postulating sequences of events that would follow a missile-producing turbine failure.

The new approach places on the applicant the responsibility for demonstrating and maintaining an NRC-specified turbine reliability by appropriate inservice inspection and testing throughout plant life. This shift of emphasis necessitates that the applicant show capability to have volumetric (ultrasonic) examinations performed that are suitable for inservice inspection of turbine discs and shaft and to provide reports for NRC review and approval that describe the applicant's methods for determining turbine missile generation probabilities.

Westinghouse and General Electric (GE) on behalf of applicants, have prepared reports for NRC review and approval that describe methods for determining turbine missile generation probabilities for their respective turbines. The design-speed missile generation probability is to be related to disc design parameters, material properties, and the inservice volumetric (ultrasonic) disc inspection interval (e.g., see Clark, Seth, and Shaffer, 1981). The destructive overspeed missile generation probability is to be related to the turbine governor and overspeed protection system's speed sensing and tripping characteristics, the design and arrangement of main steam control and stop valves and the reheat steam intercept and stop valves, and the inservice testing and inspection intervals for system components and valves (e.g., see Burns, 1977). Westinghouse and GE have submitted such reports to the NRC for review and approval. The manufacturer will provide applicants and licensees with tables of missile generation probabilities versus time (inservice volumetric disc inspection interval for design speed failure and inservice valve testing interval for destructive overspeed failure) for their particular turbine. These tables will be used to establish inspection and test schedules that will meet NRC safety objectives.

Because of the uncertainties involved in calculating P_2, the staff concludes that P_2 analyses are "ball park" or "order of magnitude" type calculations only. On the basis of simple estimates for a variety of plant layouts (e.g., Bush, 1973, and Twisdale, Dunn, and Frank, 1982), the staff further concludes that the strike and damage probability product can be reasonably taken to fall in a characteristic narrow range that is dependent on the gross features of turbine generator orientation: (1) for favorably oriented turbine generators, P_2 P_3 tend to lie in the range 10^{-3} to 10^{-2}. For these reasons (and because of weak data, controversial assumptions, and modeling difficulties), in the evaluation of P_4, the staff gives credit for the product of the strike and damage probabilities of 10^{-3} for a favorably oriented turbine and 10^{-2} for an unfavorably oriented turbine, and does not encourage calculations of them. These values represent the staff's opinion of where P_2 P_3 lie on the basis of its calculations and those of others.

It is the staff's view that the NRC safety objective with regard to turbine missiles is best expressed in terms of two sets of criteria applied to the missile generation probability (see Table 3.1). One set of criteria is to be applied to favorably oriented turbines, and the other is to be applied to unfavorably oriented turbines. Applicants and licensees with turbines from manufacturers who have had reports describing their methods and procedures for calculating turbine missile generation probabilities reviewed and accepted by the NRC are expected to meet the set of criteria appropriate to their turbine orientation, as shown in Table 3.1.

Applicants and licensees with turbines from manufacturers who have not yet submitted reports to the NRC describing their methods and procedures for calculating turbine missile generation probabilities or who have submitted reports that are still being reviewed by the NRC are expected to meet the following alternative criteria, regardless of turbine orientation:

(1) The inservice inspection program used for the steam turbine rotor assembly is to provide assurance that disc flaws that might lead to brittle failure of a disc at speeds up to design speed will be detected. The turbine rotor design should be such as to facilitate inservice inspection of all high-stress regions, including disc bores and keyways, without the need for removing the discs from the shaft. The volumetric inservice inspection interval for the steam turbine rotor assembly is to be established according to the following guidelines:

   (a) The initial inspection of a new rotor or disc should be performed before any postulated crack is calculated to grow to more than 1/2 the critical crack depth. If the calculated inspection interval is less than the scheduled first fuel cycle, the licensee should seek the manufacturer's guidance on delaying the inspection until the refueling outage. If the calculated inspection interval is longer than the first fuel cycle, the licensee should seek the manufacturer's guidance for scheduling the first inspection at a later refueling outage.

   (b) Discs that have been previously inspected and found to be free of cracks or that have been repaired to eliminate all indications should be reinspected using the same criterion as for new discs, as described in (a), calculating crack growth from the time of the last inspection.

   (c) Discs operating with known and measured cracks should be reinspected before 1/2 the time calculated for any crack to grow to 1/2 the critical crack depth. The guidance described in (a) should be used to set the inspection date based on the calculated inspection interval.

   (d) Under no circumstances is the volumetric inservice inspection interval for low-pressure (LP) discs to exceed approximately 3 years or two fuel cycles.

   Inspections during these refueling or maintenance shutdowns should consist of visual, surface, and volumetric examinations, according to the manufacturer's procedures, of all normally inaccessible parts such as couplings, coupling bolts, LP turbine shafts, blades, discs, and high-pressure rotors. Shafts and discs with cracks of depth near to or greater than 1/2 the critical crack depth are to be repaired or replaced. All cracked couplings and coupling bolts should be replaced.

(2) The inservice inspection and test program used for the governor and overspeed protection system should provide assurance that flaws or component failures in the overspeed sensing and tripping subsystems, in the main steam control and stop valves, reheat steam intercept and stop valves, or extraction steam nonreturn valves that might lead to an overspeed condition above the design overspeed will be detected. The inservice inspection program for governor and overspeed protection system's operability should include, as a minimum, the following provisions:

(a) For typical turbine governor and overspeed protection systems, at approximately 3-year intervals during refueling or maintenance shutdowns, at least one main steam control valve, one reheat stop valve, and one of each type of steam extraction valves are to be dismantled and visual and surface examinations conducted of valve seats, discs, and stems. Valve bushings should be inspected and cleaned, and bore diameters should be checked for proper clearance. If any valve is shown to have hazardous flaws or excessive corrosion or improper clearances, the valve is to be repaired or replaced and all other valves of that type dismantled and inspected.

(b) Main steam control and stop valves, reheat intercept and stop valves, and steam extraction nonreturn valves are to be exercised at least once a week during normal operation by closing each valve and observing directly the valve motion as it moves smoothly to a fully closed position.

(c) At least once a month during normal operation, each compartment of the electrohydraulic governor system (which modulates control and intercept valves) and the mechanical overspeed trip mechanism and backup electrical overspeed trip (both of which trip the main steam control and stop valves and reheat intercept and stop valves) are to be tested.

On-line test failures of any one of these subsystems require repair or replacement of failed components within 72 hours or the turbine is to be isolated from the steam supply until repairs are completed.

3.5.1.3.2 Evaluation

For Millstone Unit 3, the reactor coolant system transfers the heat generated in the core to the steam generators where steam is produced to drive the turbine generator manufactured by General Electric Corporation. The placement and orientation of the turbine generator is unfavorable with respect to the station reactor buildings; that is, there are safety-related targets inside the low trajectory missile strike zone. The turbine unit consists of one double-flow high-pressure turbine, three double-flow low-pressure turbines, and a rated rotational speed of 1,800 rpm.

Destructive Overspeed Failure Prevention

The turbine generator has a turbine control and overspeed protection that is designed to control turbine action under all normal or abnormal conditions and to ensure that a turbine trip from full load will not cause the turbine to overspeed beyond acceptable limits so as to minimize the probability of generating turbine missiles in accordance with the requirements of GDC 4. The turbine control and overspeed protection system is, therefore, essential to the overall safe operation of the plant.

Turbine control is accomplished with an electrohydraulic control (EHC) system. The EHC system consists of an electronic governor using solid-state control techniques in combination with a high-pressure hydraulic actuating system. The system includes electrical control circuits for steam pressure control, speed control, load control, and steam control valve positioning.

Three methods provide turbine overspeed control protection: the normal speed governor (EHC), the mechanical overspeed trip mechanism, and the electrical overspeed trip. The EHC modulates the turbine control valves to maintain desired speed load characteristics. At 103% of rated speed, the EHC will close the governor and intercept valves. The mechanical overspeed sensor trips the turbine stop, control, and combined intermediate valves by deenergizing the hydraulic fluid system when 110% of rated speed is reached. The electrical backup overspeed sensor trips these same valves when 112% of rated speed is reached by independently deenergizing the hydraulic fluid system. These overspeed trip systems can be tested while the unit is on line.

The staff has reviewed these systems and has concluded that the turbine overspeed protection system meets the guidelines of SRP Section 10.2 and can perform its design safety function.

According to the applicant's inservice inspection and testing program, each compartment of the mechanical and electrical overspeed protection systems will be tested during normal operation, on a weekly basis, by the following tests:

(1) a mechanical overspeed trip test at the EHC panel to test for operation of the overspeed trip device and mechanical trip valve

(2) a mechanical trip piston test at the EHC panel to test for electrical activation of the trip mechanism

(3) an electrical trip test at the EHC panel to test for operation of the electrical trip valve

(4) a backup overspeed trip test at the EHC panel to test the 2 out of 3 logic circuits

In addition, inservice inspection of the main steam and reheat valves will include the following:

(1) At least one main steam stop valve, on main steam control valve, and one reheat intercept valve will be dismantled at approximately 3-1/3-year intervals during refueling or maintenance shutdowns coinciding with the

inservice inspection schedule required by ASME Code, Section XI, and a visual and surface examination of valve seats, discs and stems will be conducted. If unacceptable flaws or excessive corrosion are found in a valve, all valves of that type will be inspected. Valve bushings will be inspected and cleaned, and bore diameters will be checked for proper clearance.

(2) Main steam stop and control and reheat stop and intercept valves and the turbine overspeed trip mechanism will be exercised at least once a week by closing each valve or performing the overspeed trip test and observing, by the valve position indicator, that the valves move smoothly to a fully closed position. This observation will be made in accordance with Technical Specification requirements by actually watching the valve motion.

GE has completed an analysis of turbine missile generation probabilities at destructive overspeed that can serve as a basis for evaluating the adequacy of the applicant's overspeed protection system inspection and testing program. The reports have been submitted to the NRC (Timo, February 6, 1984) and are under review by the staff. Until the review is complete, the NRC alternate criteria described above apply to Millstone Unit 3.

Design Speed Failure Prevention

Failures of turbine discs at or below the design speed (nominally, 120% of normal operating speed) are caused by a nonductile material failure at nominal stresses lower than the yield stress of the material. Since 1982, the staff has known of the stress corrosion cracking problems in low-pressure rotor discs of GE turbines. GE has developed and implemented procedures for inservice volumetric inspection of the bore and keyway areas of low-pressure turbine discs. It has prepared and submitted reports for NRC review that describe its methods for determining turbine disc inspection intervals and relating them to missile generation probabilities resulting from stress corrosion cracking. The GE reports have been submitted to the NRC and are being reviewed and evaluated by the staff. Until the review is complete, NRC alternate criteria described above will apply to Millstone Unit 3.

3.5.1.3.3 Summary

The staff has reviewed the Millstone Unit 3 facility with regard to the turbine missile issue and concluded that the probability of unacceptable damage to safety-related structures, systems, and components resulting from turbine missiles is acceptably low (i.e., less than 10^{-7} per year) provided the total turbine missile generation probability is such that conformance with the criteria presented in Table 3.1 is maintained, throughout the life of the plant, by acceptable inspection and test programs. In reaching this conclusion, the staff has factored into consideration the unfavorable orientation of the turbine generators. The relevant GE analyses have been submitted to the staff for review and acceptance.

Within 3 years of startup, no cracks have been observed in a GE turbine wheel with depths greater than 1/2 the critical crack depth calculated for that wheel. For these reasons, the staff is allowing the applicant up to 3 years from initiation of power output to propose a revised turbine maintenance (which establishes, with NRC-approved methods, inspection and testing procedures and schedules) and obtain NRC approval of his program. In response to an NRC request, the applicant has agreed to

(1) submit for NRC approval, within 3 years of obtaining an operating license, a turbine system maintenance program based on the manufacturer's calculations of missile generation probabilities, or

(2) volumetrically inspect all low-pressure turbine rotors at the second refueling outage as stated above and every other (alternate) refueling outage thereafter until some other maintenance program is approved by the staff, and

(3) conduct turbine steam valve maintenance (following initiation of power output) in accordance with NRC recommendations as stated above.

On the basis of its review and this agreement, the staff concludes that the turbine missile risk for the proposed plant design is acceptable and meets the requirements of GDC 4.

3.5.1.4 Missiles Generated by Natural Phenomena

The tornado missile spectrum was reviewed in accordance with SRP Section 3.5.1.4 (NUREG-0800).

GDC 2 requires that structures, systems, and components essential to safety be designed to withstand the effects of natural phenomena, and GDC 4 requires that these same plant features be protected against missiles. The missiles generated by natural phenomena that are of concern are those resulting from tornados. The applicant has identified the plant site as being in tornado Region I as defined in RG 1.76.

The applicant selected a spectrum of missiles for a tornado Region I site. The staff has evaluated this spectrum and concludes that it is representative of missiles at the site and is, therefore, acceptable. A discussion of the protection afforded to safety-related equipment from the identified tornado missiles including compliance with the guidelines of RG 1.117 is provided in Section 3.5.2 of this report. A discussion of the adequacy of barriers and structures designed to withstand the effects of the identified tornado missiles is provided in Section 3.5.3 of this report.

On the basis of its review of the tornado missile spectrum, the staff concludes that the spectrum was properly selected and meets the requirements of GDC 2 and 4 with respect to protection against natural phenomena and missiles and meets the guidelines of RGs 1.76 and 1.117 with respect to identification of missiles generated by natural phenomena. It is, therefore, acceptable. The tornado missile spectrum meets the acceptance criteria of SRP Section 3.5.1.4.

3.5.1.5 Site Proximity Missiles (Except Aircraft)

This topic is discussed in Section 3.5.1.6.

3.5.1.6 Aircraft Hazards

The results of the staff reviews for SRP Sections 3.5.1.5 and 3.5.1.6 are incorporated into the evaluation of nearby industrial, transportation, and military facilities in Section 2.2 of this SER.

3.5.2 Structures, Systems, and Components To Be Protected From Externally Generated Missiles

The design of the facility for providing protection from tornado-generated missiles was reviewed in accordance with SRP Section 3.5.2 (NUREG-0800).

GDC 2 requires that all structures, systems, and components important to the safety of the plant be protected from the effects of natural phenomena, and GDC 4 requires that all structures, systems, and components important to the safety of the plant be protected from the effects of externally generated missiles. The Millstone Unit 3 site is located in tornado Region I as identified in RG 1.76. The tornado missile spectrum is discussed in Section 3.5.1.4 of this report.

The applicant has identified all safety-related structures, systems, and components requiring protection from externally generated missiles. All safety-related structures are designed to withstand postulated tornado-generated missiles without damage to safety-related equipment. All safety-related systems and components and stored fuel are located within tornado-missile-protected structures or are provided with tornado missile barriers, except for the diesel generator exhaust piping as discussed in Section 9.5.8. The ventilation openings in buildings housing essential equipment are protected from tornado missiles by reinforced concrete labyrinths. Buried safety-related systems such as piping and electrical circuits are adequately protected by the overlying earth. The ultimate heat sink, Long Island Sound, has inherent protection against natural phenomena. The requirements of GDC 2 and 4 with respect to missile protection and the guidelines of RGs 1.13, 1.27, and 1.117 concerning tornado missile protection for safety-related structures, systems, and components including stored fuel and the ultimate heat sink are met with the exception of diesel generator exhaust piping. Protection from low-trajectory turbine missiles including compliance with RG 1.115 is discussed in Section 3.5.1.3 of this report.

On the basis of this review, the staff cannot conclude that the applicant's list of safety-related structures, systems, and components to be protected from externally generated missiles and the provisions in the plant design providing this protection are in accordance with the requirements of GDC 2 and 4 and meet the guidelines of RGs 1.13, 1.27, 1.115, and 1.117 until the applicant resolves the open item regarding tornado-generated missile protection for diesel generator exhaust piping. The staff will report resolution of this item in a supplement to the SER.

3.5.3 Barrier Design Procedures

The plant 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 (LOCA).

Information has been provided indicating that the procedures that were used in the design of the structures, shields, and barriers to resist the effect of missiles are adequate. 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 is determined by using established methods of impactive analysis. The equivalent loads of missile impact, whether the missile is environmentally generated or accidentally generated within the plant, are combined with other applicable loads as discussed in Section 3.8 of this report.

The staff concludes that the barrier design is acceptable for local effects such as penetration and meets the recommendations of SRP Section 3.5.3. How ever, the procedure used by the applicant to evaluate the overall response of a barrier when impacted by a missile is different than the one recommended in SRP Section 3.5.3. The procedure used by the applicant takes into consideration the deformability of a nonpenetrating missile; the SRP procedure, however, considers the missile to be rigid. On the basis of discussions with the applicant in a meeting on June 14, 1984, the staff believes that considering the sizes of the missile barriers at Millstone Unit 3, the actual design of barriers will not be affected by the implementation of the staff position. The applicant has committed to provide the information to confirm the above finding. The staff will review the information and report its finding in a supplement to this SER.

Pending the satisfactory review of the information discussed above, the staff concludes the following:

The procedures used to determine the effects and loading on seismic Category I structures and missile shields and barriers induced by design-basis missiles selected for the plant provide a conservative basis for engineering design. These provisions, therefore, are acceptable to ensure that the structure or barriers are adequately resistant to and will withstand the effects of such forces. Conformance with these procedures is an acceptable basis for satisfying, in part, the requirements of GDC 2 and 4.

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

3.6.1 Plant Design for Protection Against Postulated Failures in Fluid Systems Outside Containment

The design of the facility for providing protection against postulated piping failures outside containment was reviewed in accordance with SRP Section 3.6.1 (NUREG-0800).

The staff's guidelines for meeting the requirements of GDC 4 concerning protection against postulated piping failure in high-energy and moderate-energy fluid systems outside containment are contained in Branch Technical Position (BTP) ASB 3-1. The applicant has used the guidance in SRP Section 3.6.1 and BTP ASB 3-1 in evaluating the effects of high- and moderate-energy-pipe breaks. The applicant has identified all high- and moderate-energy piping systems in accordance with these guidelines and also has identified those systems requiring protection from postulated piping failures. The means used to protect safety-related systems and components throughout the plant include physical separation, enclosure in suitably designed structures or compartments, drainage systems, pipe whip restraints, equipment shields, and necessary equipment environmental qualification.

The main steam and feedwater lines including the isolation valves are located in the main steam valve building and have been classified as part of the break exclusion boundary. The applicant has performed a subcompartment analysis for the main steam and feedwater piping in the main steam valve building to ensure that the resulting jet impingement and environmental effects of the postulated pipe break in one of these lines will not result in adverse consequences. The results of this analysis indicate that the integrity of the main steam valve building is not affected by the pressure increase from the resulting blowdown. Further discussion of the environmental qualification of safety-related equipment is contained in Section 3.11 of this SER.

The applicant has not completed the analysis of the rupture of high-energy piping systems and has stated that the complete analysis will be submitted in a future amendment to the FSAR. The applicant has recently provided additional information for the staff to perform an independent calculation to verify the applicant's analysis of the environmental conditions in a compartment after a high-energy-line break. This information is being reviewed.

On the basis of the above, the staff cannot conclude that the plant design meets the requirements of GDC 4, and the criteria set forth in BTP ASB 3-1 with regard to the protection of all high-and moderate-energy piping systems in accordance with these guidelines and also has identified those systems requiring protection from postulated piping failures. The means used to protect safety-related systems and components throughout the plant include physical separation, enclosure in suitably designed structures or compartments, drainage systems, pipe whip restraints, equipment shields, and necessary equipment environmental qualification.

The main steam and feedwater lines including the isolation valves are located in the main steam valve building and have been classified as part of the break exclusion boundary. The applicant has performed a subcompartment analysis for the main steam and feedwater piping in the main steam valve building to ensure that the resulting jet impingement and environmental effects of the postulated pipe break in one of these lines will not result in adverse consequences. The results of this analysis indicate that the integrity of the main steam valve building is not affected by the pressure increase from the resulting blowdown. Further discussion of the environmental qualification of safety-related equipment is contained in Section 3.11 of this SER.

Pending receipt of acceptable information as discussed above, the staff cannot find that the applicant has 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. On the basis of the above, the staff cannot conclude that the plant design meets the requirements of GDC 4, and the criteria set forth in BTP ASB 3-1 with regard to the protection of safety-related systems and components from a postulated high- and moderate-energy-line break. The staff will report resolution of these items in a supplement to this SER.

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

GDC 4 requires that structures, systems, and components important to safety shall be designed to be compatible with and to accommodate the effects of the environmental conditions as a result of normal operations, maintenance, testing, and postulated accidents, including LOCAs. These structures, systems, and components shall be adequately protected against dynamic effects, including the effects of missiles, pipe whipping, and discharging fluids, that may result from equipment failures and from events and conditions outside the nuclear power plant.

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

Pipe whip need only be considered in those high-energy piping systems having fluid reservoirs with sufficient capacity to develop a jet stream. The criterion for determining high- and moderate-energy lines is found in BTP ASB 3-1 of SRP Section 3.6.1. This criterion has been used correctly by the applicant. A list of all high-energy systems is included in the FSAR.

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

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

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

On the basis of the staff review of FSAR Section 3.6.2, the staff concludes that the pipe rupture postulation and the associated effects are adequately considered in the plant design and are, therefore, acceptable and meet the requirements of GDC 4. This conclusion is based on the following:

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

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

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

3.7 Seismic Design

3.7.1 Seismic Input

The horizontal peak acceleration value of the safe shutdown earthquake (SSE), chosen for the rock level, is 0.17g. The corresponding peak acceleration for the operating basis earthquake (OBE) is 0.09g. The horizontal design response spectra are smooth spectra anchored to the above accelerations. The vertical design response spectra are taken to be two-thirds of the horizontal design spectra. These spectra have been reviewed in Section 2.5.2 of this report and have been found to be acceptable as design-basis spectra.

The specific percentage of critical damping values used in the seismic analysis of Category I structures, systems, and components, in general, is more conservative than that specified in RG 1.61. The only exception is the value used for the bolted steel structures. The applicant has used 5% of critical damping for bolted steel structures when the stresses are limited to 0.5 yield stress. For this situation RG 1.61 specifies 4% of critical damping. The applicant in a structural audit meeting (February 27 - March 2, 1984) and in a letter dated March 24, 1984 informed the staff that the only structures which were determined to have used 5% damping for bolted steel members for OBE loading were the turbine building (non-seismic Category I and the containment structure enclosure (CSE).

The OBE structural response of the CSE was produced by a dynamic model which included the concrete reactor containment superstructure. The modal dampings utilized in the analysis were based on 2% structural damping for concrete members (as opposed to 4% specified in RG 1.61) and 5% structural damping for steel members. The modal dampings which resulted for all significant modes were less than 4%.

The OBE structural response of the turbine building was produced by a dynamic model which utilized a constant 5% damping for all modes. An OBE design response spectra based on 4% damping produces spectral accelerations no more than 10% greater than the spectral accelerations from 5% damping. The resulting increase in OBE structural responses for the building produces stress levels which remain within the allowables.

On the basis of the review of the above information, the staff considers this issue resolved.

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

Most of the major safety-related structures are founded on bedrock, with the exception of the control building, emergency diesel generator building, fuel building, and the hydrogen recombiner building. The control building is founded on 1 to 4 ft of compacted structural backfill overlying basal till of thickness varying between 1 ft on the east side and 15 ft on the west. The emergency diesel generator building is founded on approximately 20 ft of structural backfill overlying a 20-ft-thick layer of basal till. The fuel building is founded partially on basal till and partially on bedrock, and the hydrogen recombiner is founded on concrete fill overlying bedrock.

On the basis of the above findings, the staff concludes that the seismic design parameters used in the plant structural design are acceptable and meet the recommendations of SRP Section 3.7.1 and the requirements of GDC 2 and Appendix A to 10 CFR 100.

3.7.2 Seismic System Analysis

This topic is addressed in Section 3.7.3.

3.7.3 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 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 Category I structure overturning, and determination of composite damping. The review included design criteria and procedures for evaluation of interaction of non-Category I structures with Category I structures and the effects of parameter variations on floor response spectra. The review also included criteria and seismic analysis procedures for reactor internals and Category I buried structures outside the containment.

The system and subsystem analyses were performed by the applicant on an elastic and linear basis. Modal response spectrum multidegree-of-freedom methods form the bases for the analyses of all major Category I structures, systems, and components. In applying the modal response spectrum method, governing response parameters were combined in conformance with one position of RG 1.92. The absolute sum of the modal responses was used for modes with closely spaced frequencies. The square root of the sum of the squares (SRSS) of the maximum codirectional responses was used in accounting for three components of the earthquake motion. 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 used for all structures, systems, and components where analyses showed significant structural amplification in the vertical direction. Torsional effects and stability against overturning were considered.

The inertial effects as a result of an earthquake on buried systems and tunnels have been adequately accounted for in the analysis. The principles used to account for the effects of static resistance of the surrounding soil on buried system deformations were based on the theory of structures on elastic foundations, and they are acceptable.

The applicant has used discrete soil springs to analyze the structures directly founded on the bedrock to evaluate the soil-structure interaction effects. The emergency generator enclosure and the control building, which are founded on the shallow soil overburden overrock, have been analyzed using finite-element technique. The current staff position requires that the soil-structure interaction should include both elastic half-space and finite-element approaches for all Category I structures founded on soil. These Category I structures should be designed to responses obtained by any of the following methods:

(1) envelope of results of the two methods

(2) results of one method with conservative design considerations of impact from use of other method

(3) combination of (1) and (2) with provision of adequate conservatism in design

In a response to a staff question on the above position, the applicant stated that studies that have been conducted on the only structure that is completely soil founded (the emergency generator enclosure) indicate that the finite-element results provided more severe results than the half-space representation. In addition, the applicant also conducted half-space analysis for the control building.

The staff reviewed the results of the alternate half-space analysis for both structures during the structural audit. Comparisons between resulting fundamental frequencies and seismic responses from both methods indicate that they are in close agreement and that the finite element method produces slightly more severe responses. With this finding, the staff concludes that the applicant has complied with the staff's position on soil structure interaction and the issue is resolved.

The current staff position requires that an additional eccentricity effect based on a consideration of ±5% of the maximum building dimension at the level under consideration shall be assumed to account for accidental torsion. The applicant's analyses of Category I structures do not account for the accidental torsional effects. In a response to a staff question on the above issue, the applicant noted that the Millstone Unit 3 design was finalized before the development of the above staff position. However, the applicant has now analyzed four structures to assess the effects on the plant structures that might result from the implementation of the staff position. These structures include the fuel building, control building, containment structure internals, and containment structure shell. The results of the applicant's analyses (provided with letters dated March 23, 1984 and May 31, 1984) indicate that the critical wall sections for all four structures have adequate reinforcement to resist additional shear forces resulting from the consideration of the accidental torsion. On the basis of this information, the staff concludes that the applicant has met the intent of the current staff requirement and, therefore, this issue is resolved.

The applicant's procedure to generate the floor response spectra is not in compliance with the staff's accepted procedure as delineated in RG 1.122. In particular, the applicant did not consider effects of three components of earthquake in generating the floor response spectra and also used a peak broadening technique which is less severe than that recommended in RG 1.122. However, the procedure used by the applicant also includes conservatism in that the damping values used for both structures and equipments (5% for structures and 2% for equipment) are lower than those accepted by the staff and recommended in RG 1.61 (7% for structures and 2% for equipment). Therefore, in order to resolve this issue, the applicant regenerated floor response spectra for the containment structure, auxiliary building, and fuel building using RG 1.122 procedure and damping values consistent with RG 1.61.

The applicant provided results of his analyses with letters dated May 4, May 31, and June 28, 1984. A comparison between the RG 1.122 spectra (those developed using RG 1.122 procedure) and the design spectra (those developed using the applicant's procedure) indicates that the design spectra, in general, exceed RG 1.122 spectra. In fact, the peak levels of the design spectra are from 1.5 to 3.0 times as severe as those of the RG 1.122 spectra. RG 1.22 spectra, occasionally, exhibit exceedances over the design spectra in a very narrow frequency range. However, taking into account that these exceedances are minor and the design spectra exhibit severe exceedances at other frequencies, it is concluded that the design floor response spectra generated by the applicant have met the intent of RG 1.122 and this issue is considered resolved.

The floor response spectra at Millstone Unit 3 were developed with the assumption that all slabs behaved rigidly in the vertical direction. In order to examine the effects of floor flexibility on the floor response spectra, the applicant undertook a sensitivity study by examining the floor slabs in the containment, auxiliary, and control buildings. The containment and auxiliary building floor slabs were chosen as a representative slab for the seismic Category I structures at Millstone Unit 3. The control building was chosen because it has different characteristics from other Category I structures.

By examining floor slabs in the containment and auxiliary building, it was determined that only the 1-ft-thick slabs at the south end of the auxiliary building have a fundamental vertical frequency less than 33 Hz. The applicant generated new spectra at these floors accounting for the floor flexibility. Again, in generating new spectra, the applicant used 7% structural damping and 2% equipment damping as opposed to 5% structural damping and 1% equipment damping used in the applicant's original analysis. In addition, the applicant used a newly developed vertical time history which matched design response spectra more closely than the time history used in the original analysis.

The comparison between the new spectra (accounting for the floor flexibility and higher damping, and the old spectra (based on the rigid floor assumption and lower damping) indicates that the old spectra practically enveloped the new spectra in the entire frequency range. The minor exceedances in the frequency range of 25 to 30 Hz is of no practical consequence. Thus, consideration of the vertical floor flexibility has no impact on the vertical spectra generated in Millstone Unit 3 Category I structures except for the control building. The issue of the control building is discussed below.

The comparison between the new spectra and the old spectra of the control building (at el 47 ft) indicates that at many locations the new spectra are bounded by the old spectra. However, at a few locations, the new spectra exhibit a higher peak between the frequency range of 15 to 20 Hz. This exceedance, in some cases, is about 100%. At other frequencies, the new spectra are bounded by the old spectra for practical considerations.

The investigation of floor slabs in the control building indicated that there was no impact on the structural design of the slabs because of the flexibility consideration. In addition, the applicant also stated that the floor spectra used in the qualification of the equipment at these floors enveloped the new spectra. Thus, the consideration of the vertical floor flexibility has no impact on the control building design. On the basis of the above findings, the staff considers the issue of the vertical floor flexibility resolved.

The staff concludes that the plant design is acceptable and meets the recommendations of SRP Sections 3.7.2 and 3.7.3 and the requirements of GDC 2 and Appendix A to 10 CFR 100. The staff concludes that the plant design is acceptable and meets the recommendations of SRP Section 3.7.3 and the requirements of GDC 2 and Appendix A to 10 CFR 100 with respect to the capability of the structures to withstand the effects of the earthquakes so that their design reflects

(1) appropriate consideration of the most severe earthquake recorded for the site with an appropriate margin (GDC 2); consideration of two levels of earthquakes (Appendix A, 10 CFR 100)

(2) appropriate combination of the effects of normal and accident conditions with the effect of the national phenomena

(3) the importance of the safety functions to be performed (GDC 2); the use of a suitable dynamic analysis or a suitable qualification test to demonstrate that structures, system, and components can withstand the seismic and other concurrent loads, except where it can be demonstrated that the use of an equivalent static load method provides adequate consideration (Appendix A, 10 CFR 100)

The applicant has met the requirements of Item 1 above by use of the acceptable seismic design parameters according to SRP Section 3.7.1. The combination of earthquake-resultant loads with those resulting from normal and accident conditions in the design of Category I structures as specified in SRP Sections 3.8.1 through 3.8.5 will be in conformance with Item 2 above.

3.7.4 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 RG 1.12. Supporting instrumentation is being installed on Category I structures, systems, and components to provide data for the verification of the seismic responses determined analytically for such Category I items.

The applicant has met the recommendations of SRP Section 3.7.4 except that a seismic instrumentation surveillance scheme has not been provided. However, such a scheme, in accordance with stated staff requirements, will be incorporated in the Technical Specifications. The requirements of 10 CFR 100, Appendix A, are met by providing the instrumentation that is capable of measuring the effects of an earthquake, which meets the requirements of GDC 2. The applicant will have met the requirements of 10 CFR 50.55a by providing the inservice inspection program that will verify operability by the performance of channel checks, calibrations, and functional tests at acceptable intervals. In addition, the installation of the specific seismic instrumentation on the reactor containment structure and at other Category I structures, systems, and components constitutes an acceptable program to record data on seismic ground motion as well as data on the frequency and amplitude relationship of the seismic response of major structures and systems. A prompt readout of pertinent data at the control room can be expected to yield sufficient information to guide the operator on a timely basis for the purpose of evaluating the seismic response in the event of an earthquake. Data obtained from such installed seismic instrumentation will be sufficient to determine that the seismic analysis assumptions and the analytical model used for the design of the plant are adequate and that allowable stresses are not exceeded under conditions where continuity of operation is intended. Provision of such seismic instrumentation complies with RG 1.12.

3.8 Design of Seismic Category I Structures

3.8.1 Concrete Containment

The reactor coolant system is enclosed in the reinforced concrete containment as described in Section 3.8.1 of the FSAR. The containment structure is designed, primarily, in accordance with American Concrete Institute (ACI) 318-71 and the American Institute of Steel Construction (AISC) specification, 1969 edition, to resist various combinations of dead loads, live loads, environmental loads (including those resulting from wind, tornados, OBEs and SSEs), and loads generated by the design-basis accident (including pressure, temperature, and associated pipe rupture effects).

The liner design is based on the guidance provided in Sections III and VIII of the ASME Code, 1971 Edition.

The containment is designed for the peak tangential shear. The tangential shear stress capacity of concrete is limited to 40 psi. This value is acceptable to the staff.

The current staff position requires that the design of concrete containment be in accordance with applicable provisions of ASME Code, Section III, Division 2, except for the tangential shear criteria. Further, the liner design is acceptable if it is in accordance with Article CC 3000 and the provisions of Subsection NE, Division 1, Section III of the ASME Code. In addition, the provisions of RG 1.136 are also applicable to the containment design.

By a letter dated May 4, 1984, the applicant provided the following comparisons to address this issue

(1) a comparison between Millstone Unit 3 design criteria and ASME Code, Section III, Division 1 criteria to assess the impact of the current staff position on Millstone Unit 3 liner design

(2) a comparison between Millstone Unit 3 design criteria and ACI 359-80 (ASME Code, Section III, Division 2) to assess the impact of the current staff position on the Millstone Unit 3 containment design

These comparisons indicate that, in general, the applicant's design has met the intent of the current staff acceptance criteria. In particular, strain limits for the liner and the load combinations allowable stresses for the containment design have met the intent of the current requirements. On the basis of this information, the staff considers this issue resolved.

The applicant has performed the ultimate capacity analysis of Millstone Unit 3 containment in conjunction with the Millstone Unit 3 probabilistic safety study. The applicant has reported mean failure pressure ranging from 128 psi (2.84 x design pressure) to 155 psi (3.44 x design pressure) based on various components of the containment and their failure modes. The staff met with the applicant on June 14, 1984, to discuss this subject further and to establish the lower bound ultimate capacity consistent with the leak-tight performance. On the basis of discussions at the meeting, the staff anticipates that the lower bound ultimate capacity of Millstone Unit 3 should be about 2.5 x design pressure or greater as observed for some of the recent containments. The staff has requested additional information from the applicant to clearly establish an actual lower bound. The staff will report its findings in a supplement to this SER.

Pending the review of the above confirmatory information, the staff concludes that

(1) The applicant has met the recommendations of 10 CFR 50.55a and GDC 1 with respect to ensuring that the concrete containment is designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with the safety function to be performed by meeting the guidelines of RGs and industry standards indicated below.

(2) The applicant has met the requirements of GDC 2 by designing the concrete containment to withstand the most severe earthquake that has been established for the site with sufficient margin and the combinations of the effects of normal and accident condition with the effects of environmental loadings such as earthquakes and other natural phenomena.

(3) The applicant has met the requirements of GDC 4 by ensuring that the design of the steel containment is capable of withstanding the dynamic effects associated with missiles, pipe whipping, and discharging fluids.

(4) The applicant has met the requirements of GDC 16 by designing the concrete containment so that it is an essentially leaktight barrier to prevent the uncontrolled release of radioactive effluents to the environment.

(5) The applicant has met the requirements of GDC 50 by designing the concrete containment to accommodate, with sufficient margin, the design leakage rate, calculated pressure, and temperature conditions resulting from accident conditions, and by ensuring that the design conditions are not exceeded during the full course of the accident condition. In meeting these design requirements, the applicant has used the recommendations of RGs and industry standards indicated below.

The criteria used in the analysis, design, and construction of the concrete containment structure to account for anticipated loadings and postulated conditions that may be imposed on the structure during its service lifetime are in conformance with established criteria, and with codes, standards, guides, and specifications acceptable to the staff. These include meeting the intent of the positions of RG 1.136 and industry standard, ASME Code, Section III, Division 2.

The use of these criteria as defined by applicable codes, standards, guides, and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control programs, and special construction techniques; and the testing and inservice surveillance requirements provide reasonable assurance that, in the event of winds, tornados, earthquakes, and various postulated accidents occurring within and outside the containment, the structure will withstand the specified design conditions without impairment of structural integrity or safety function of limiting the release of radioactive material.

3.8.2 Steel Containment

Not applicable to this facility.

3.8.3 Concrete and Structural Steel Internal Structures

The containment interior structures consist of reinforced concrete and steel framed walls, compartments, and floors. The major code used in the design of concrete internal structures was ACI 318-71.

Steel internal structures were designed in accordance with the AISC specification, "Specifications for the Design, Fabrication, and Erection of Structural Steel for Buildings," 1969 Edition.

The current staff positions require that the concrete internal structure should be designed in accordance with ACI 349-76, "Requirements for Nuclear Safety-Related Structures," as amplified by RG 1.142. By letters dated May 4, 1984, and May 31, 1984, the applicants provided a comparison between the Millstone Unit 3 design criteria (ACI 318-71) and ACI 349-76 and RG 1.142 to assess the impact of the current staff position on the Millstone Unit 3 structural design (other than containment).

These comparisons indicate that the Millstone Unit 3 design, in general, has met the intent of the current staff requirement. In particular, the controlling load combinations for Millstone Unit 2 design are the same as or more severe than those currently required by the staff. On the basis of these findings, the staff considers this issue resolved.

The staff concludes that the design of the containment internal structures is acceptable and meets the relevant requirements of 10 CFR 50.55a and GDC 1, 2, 4, 5, and 50. This conclusion is based on the following:

(1) The applicant has met the requirements of 10 CFR 50.55a and GDC 1 with respect to ensuring that the containment internal structures are designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with their safety functions to be performed by meeting the guidelines of RGs and industry standards.

(2) The applicant has met the requirements of GDC 2 by designing the containment internal structures to withstand the most severe earthquake that has been established for the site with sufficient margin and the combinations of the effects of normal and accident conditions with the effects of environmental loadings such as earthquakes and other natural phenomena.

(3) The applicant has met the requirements of GDC 4 by ensuring that the design of the internal structures is capable of withstanding the dynamic effects associated with missiles, pipe whipping, and discharging fluids.

(4) The applicant has met the requirements of GDC 5 by demonstrating that structures, systems, and components are not shared between units or that if shared they have demonstrated that sharing will not impair their ability to perform their intended safety function.

(5) The applicant has met the requirements of GDC 50 by designing the containment internal structures to accommodate, with sufficient margin, the design leakage rate and calculated pressure and temperature conditions resulting from accident conditions and by ensuring that the design conditions are not exceeded during the full course of the accident condition. In meeting these design requirements, the applicant has used the recommendations of RGs and industry standards indicated below. The applicant has also performed appropriate analysis that demonstrates the ultimate capacity of the structures will not be exceeded and establishes the minimum margin of safety for the design.

The criteria used in the design, analysis, and construction of the containment internal structures to account for anticipated loadings and postulated conditions that may be imposed on the structures during their service lifetime are in conformance with established criteria and with codes, standards, and specifications acceptable to the staff. These include meeting the positions of RGs 1.57 and 1.42 and industry standards: ACI-349; ASME Code, Section III, Division 2, Subsections NE and NF; AISC specification (1969 Edition); and ANSI N45.2.5.

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

3.8.4 Other Seismic Category I Structures

Other seismic Catetory I structures are

(1) containment enclosure building
(2) auxiliary building
(3) fuel building
(4) control building
(5) cable tunnel
(6) emergency generator enclosure and diesel fuel oil tank vault
(7) engineered safety features building
(8) main steam valve building
(9) circulating and service water pumphouse
(10) hydrogen recombiner building
(11) circulating water discharge tunnel and discharge structure
(12) railroad canopy

Category I structures other than the containment and its interior structures are all constructed of structural steel and/or concrete. The structural components consist of slabs, walls, beams, and columns. The major code used in the design of concrete Category I structures was ACI 318-71. For steel Category I structures, the AISC specification, "Specifications for the Design, Fabrication and Erection of Structural Steel for Buildings," 7th Edition, was used.

The concrete and steel Category I structures were designed to resist various combinations of dead loads; environmental loads including winds, tornados, OBE, and SSE; 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 current staff position requires that the concrete internal structure should be designed in accordance with ACI 349-76 as amplified by RG 1.142. By letters dated May 4, 1984, and May 31, 1984, the applicants provided a comparison between the Millstone Unit 3 design criteria (ACI 318-71) and ACI 349-76 and RG 1.142 to assess the impact of the current staff position on the Millstone Unit 3 structural design (other than containment).

These comparisons indicate that the Millstone Unit 3 design, in general, has met the intent of the current staff requirement. In particular, the controlling load combinations for the Millstone Unit 3 design are the same as or more severe than those currently required by the staff. On the basis of these findings, the staff considers this issue resolved.

The applicant by a letter dated May 15, 1984, has informed the staff that the design of the spent fuel pool racks complies with the current staff acceptance criteria. The staff plans to review the information which will confirm such compliance.

Pending the review of the confirmatory information, the staff concludes that the design of safety-related structures other than containment or containment interior structures is acceptable and meets the relevant requirements of 10 CFR 50.55a and GDC 1, 2, 4, and 5. This conclusion is based on the following:

(1) The applicant has met the requirements of 10 CFR 50.55a and GDC 1 with respect to ensuring that the safety-related structures other than containment are designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with their safety functions to be performed by meeting the guidelines of RGs and industry standards.

(2) The applicant has met the requirements of GDC 2 by designing the safety-related structures other than containment to withstand the most severe earthquake that has been established for the site with sufficient margin and the combinations of the effects of normal and accident conditions with the effects of environmental loadings such as earthquakes and other natural phenomena.

(3) The applicant has met the requirements of GDC 4 by ensuring that the designs of the safety-related structures are capable of withstanding the dynamic effects associated with missiles, pipe whipping, and discharging fluids.

(4) The applicant has met the requirements of GDC 5 by demonstrating that structures, systems, and components are not shared between units or that if shared by demonstrating that sharing will not impair their ability to perform their intended safety function.

(5) The applicant has met the requirements of Appendix B to 10 CFR 50 because his quality assurance program provides adequate measures for implementing guidelines relating to structural design audits.

The criteria used in the analysis, design, and construction of all the plant Category I structures to account for anticipated loadings and postulated conditions that may be imposed on each structure during its service lifetime are in conformance with established criteria, codes, standards, and specifications acceptable to the staff. These include meeting the positions of RG 1.142 and industry standards, ACI-349 and AISC specification, 7th Edition.

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

3.8.5 Foundations

Foundations of Category I structures are described in FSAR Section 3.8.5. Primarily, these foundations are reinforced concrete of the mat type. The major code used in the design of these concrete mat foundations was ACI 318-71. These concrete foundations have been designed to resist various combinations of dead loads; live loads; environmental loads including wind, tornado, and seismic; and the loads postulated by ruptures of high-energy pipes.

The current staff position requires that the concrete internal structures should be designed in accordance with ACI-349 as amplified by RG 1.142. By letters dated May 4, 1984, and May 31, 1984, the applicants provided a comparison between the Millstone Unit 3 design criteria (ACI 318-71) and ACI 349-76 and RG 1.142 to assess the impact of the current staff position on the Millstone Unit 3 structural design (other than containment).

These comparisons indicate that the Millstone Unit 3 design, in general, has met the intent of the current staff requirement. In particular, the controlling load combinations for the Millstone Unit 3 design are the same as or more severe than those currently required by the staff. On the basis of these findings, the staff considers this issue resolved.

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

The staff concludes that the design of the seismic Category I foundations is acceptable and meets the relevant requirements of 10 CFR 50.55a and GDC 1, 2, 4, and 5. This conclusion is based on the following:

(1) The applicant has met the requirements of 10 CFR 50.55a and GDC 1 with respect to ensuring that the seismic Category I foundations are designed, fabricated, erected, constructed, tested, and inspected to quality standards commensurate with their safety functions to be performed by meeting the guidelines of RGs and industry standards.

(2) The applicant has met the requirements of GDC 2 by designing the seismic Category I foundations to withstand the most severe earthquake that has been established for the site with sufficient margin and the combinations of the effects of normal and accident conditions with the effects of environmental loadings such as earthquakes and other natural phenomena.

(3) The applicant has met the requirements of GDC 4 by ensuring that the design of seismic Category I foundations is capable of withstanding the dynamic effects associated with missiles, pipe whipping, and discharging fluids.

(4) The applicant has met the requirements of GDC 5 by demonstrating that structures, systems, and components either are not shared between units or that, if shared, by demonstrating that sharing will not impair their ability to perform their intended safety function.

The use of these criteria as defined by applicable codes, standards, and specifications; the loads and loading combinations; the design and analysis procedures; the structural acceptance criteria; the materials, quality control, and special construction techniques; and the testing and inservice surveillance requirements provide reasonable assurance that, in the event of winds, tornados, 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.

3.8.6 Structural Audit

From February 27 through March 2, 1984, the staff met with the applicant and his consultants to conduct the seismic and structural audit. The audit covered major safety-related structures at Millstone Unit 3. The staff conducted the audit in order to

(1) investigate in detail the manner in which the applicant has implemented the structural and seismic design criteria that he committed to use before obtaining construction permits for the facility

(2) verify that the key structural and seismic design and the related calculations have been conducted in an acceptable way

(3) identify and assess the safety significance of these areas where the plant structures were designed and analyzed using methods other than those recommended by the SRP

As a result of the audit, the staff identified several action items consolidating all the outstanding issues relative to the adequacy of the Millstone Unit 3 structural design. The review and evaluation of the information resulting from these action items provided a basis for the conclusion reached and reported in this SER.

3.9 Mechanical Systems and Components

The review performed under SRP Sections 3.9.1 through 3.9.6 (NUREG-0800) pertains to the structural integrity and functional capability of various safety-related mechanical components in the plant. The staff's review is not limited to ASME Code components and supports, but is extended to other components such as control rod drive mechanisms (CRDMs), certain reactor internals, and any safety-related piping designed to industry standards other than the ASME Code. The staff reviews such issues as load combinations, allowable stresses, methods of analysis, summary of results, and preoperational testing. The staff's review must arrive at the conclusion that there is adequate assurance of a mechanical component performing its safety-related function under all postulated combinations or normal operating conditions, system operating transients, postulated pipe breaks, and seismic events.

3.9.1 Special Topics for Mechanical Components

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

The applicant has provided a list of the design transients and the number of cycles for each design transient used for design. Five OBEs of ten cycles each and one SSE of ten cycles have been included. This is in conformance with the requirements of SRP Section 3.9.1. The staff concludes from its review of the design transients and their respective number of cycles that they are acceptable.

The applicant used computer programs to perform analyses of mechanical components. The FSAR includes a list showing all computer programs used by the applicant for static and dynamic analyses to determine the structural integrity and functional integrity of seismic Category I Code and non-Code items and the analyzes to determine stresses along with a description of the program. Design control measures to verify the adequacy of the design of safety-related components are required to meet 10 CFR 50, Appendix B.

The applicant has used inelastic analysis for two 3-in. charging line nozzles, four 3-in. high pressure safety-injection nozzles, and 12 sets of circumferential as-welded butt welds. The method of analysis is in accordance with ASME Code, Section III, Paragraph NB-3228, and Code Case N-196-1. A limit of 5% on the maximum accumulated strain is used to preclude the possibility of incremental collapse. Nonlinear stress-strain property characteristics are obtained from Code Case N-47-21. The staff review has found that the method of analysis is in accordance with the provisions of the ASME Code and is, thus, acceptable. However, the staff review did find that the loadings evaluated in the inelastic analysis did not include LOCA loads. This issue is addressed in Section 3.9.3.1 of this SER.

On the basis of an acceptable resolution of the above-identified item, the staff concludes that the design transients and resulting loads and load combinations with appropriate specified design and service limits for mechanical components and supports are acceptable and meet the relevant requirements of GDC 1, 2, 14, and 15; 10 CFR 50, Appendix B; and 10 CFR 100, Appendix A. This is based on the following:

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

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

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

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

The staff has reviewed the methodology, testing procedures, and dynamic analyses used by the applicant to ensure the structural integrity and functionality of piping systems, mechanical equipment, and their supports under vibratory loadings according to SRP Section 3.9.2 (NUREG-0800). The staff's review included (1) the piping vibration, thermal expansion, and dynamic effect testing, (2) the seismic system analysis methods, (3) the dynamic responses of structural components within the reactor caused by steady-state and operational flow transient conditions for nonprototype reactors, (4) flow-induced-vibration testing of reactor internals to be conducted during the preoperational and startup test program, and (5) the dynamic analysis methods used to confirm the structural design adequacy and functional capability of the reactor internals and piping attached to the reactor vessel when subjected to loads from a LOCA in combination with an SSE.

3.9.2.1 Piping Preoperational Vibration and Dynamic Effects Testing

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

In a letter from W. G. Counsil to B. J. Youngblood, dated June 28, 1984, the applicant provided vibration preoperational test acceptance criteria for Millstone Unit 3. For steady-state vibration, the maximum alternating stress intensity will be limited to 61.5% of the endurance limit defined as the alternating stress intensity, S_a at 10^6 cycles as given in Appendix I, Figure I-9.1, of Section III of the ASME Code for carbon steels. For austenitic pipe steels, the maximum alternating stress intensity will be limited to 60% of the endurance limit defined as the alternating stress intensity, S_a at 10^6 cycles or 100% of the endurance limit defined as the alternating stress intensity, S_a at 10^{11} cycles as given in Appendix I, Figures I-9.2.1 and I-9.2.2, of Section III of the ASME Code, respectively. When curve A, B, or C of Figure I-9.2.2 is used, the applicable ASME Code requirements for each curve shall be met.

The staff has reviewed the Millstone Unit 3 piping vibration acceptance criteria and finds that these criteria will provide an acceptable level of safety for piping vibration during the plant's 40-year life.

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

3.9.2.2 Seismic Subsystem Analysis

The staff's review performed according to SRP Section 3.9.2 included FSAR Section 3.7.3. Areas reviewed were seismic analyses methods, determination of the number of earthquake cycles, basis for selection of frequencies, the combination of modal responses and spatial components of an earthquake, criteria used for damping, torsional effects of eccentric masses, interaction of other piping with Category I piping, and Category I buried piping systems.

The scope of the review of the Millstone Unit 3 seismic system and subsystem analysis included the seismic analysis methods for all seismic Category I piping, systems, and components. The staff reviewed the manner in which the dynamic system analysis is performed, the method of selection of significant modes, whether the number of masses or degrees of freedom is adequate, and how consideration is given to maximum relative displacements. The review included design methodologies and procedures used for the evaluation of the interaction of piping that is not seismic Category I with seismic Category I piping, and the seismic methods that consider the effect of fill settlement and movement at support points, penetration, and anchors for seismic Category I buried piping systems. In addition, the staff reviewed seismic analysis procedures for reactor internals. The system and subsystem analyses are performed by the applicant on an elastic basis. Modal response spectrum, multidegree-of-freedom, and time history methods form the basis for the analyses of all major seismic Category I systems and components. When the response spectrum method is used, modal responses are combined by the square-root-sum-of-the-squares (SRSS) rule.

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

On the basis of the staff's review of FSAR Section 3.7.3, the staff concludes that the applicant has met the relevant requirements of GDC 2 with respect to demonstrating the design adequacy of all Category I piping systems, components, and their supports to withstand earthquakes by meeting the regulatory positions of RG 1.61 and 1.92 and by providing acceptable seismic analysis procedures and criteria. The scope of review of the seismic system analysis included the seismic analysis methods for all Category I piping systems, components, and their supports. It included review of procedures for modeling, and inclusion of torsional effects, seismic analysis of multiply supported equipment and components with distinct inputs, and determination of composite damping. The review included design criteria and procedures for evaluating the interaction of non-Category I piping with Category I piping. The review also included criteria and seismic analysis procedures for reactor internals.

3.9.2.3 Preoperational Flow-Induced Vibration Testing of Reactor Internals

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

The Indian Point Unit 2 reactor has been established as the prototype for the Westinghouse four-loop plant internals verification program. The only significant differences between the Millstone Unit 3 internals and the Indian Point Unit 2 internals are the replacement of the annular thermal shield with neutron shield panels, the substitution of 15 x 15 fuel assemblies for 17 x 17 assemblies, and the change to the upper-head-injection (UHI)-style inverted top hat support structure configuration.

The change to the neutron shield panels and 17 x 17 fuel assemblies has been tested at the Trojan plant. The change to the UHI-style inverted top hat support structure configuration has been tested at the Sequoyah Unit 1 plant. The four-loop internals assurance program conducted on Indian Point Unit 2 supplemented by the Trojan and Sequoyah Unit 1 data satisfies RG 1.20.

The applicant has committed to test the reactor internals in accordance with the provisions of RG 1.20, Revision 2, for nonprototype Category I plants. The applicant will conduct a visual inspection before hot functional testing; after hot functional testing the applicant has committed to inspect all major load-bearing surfaces; torsional, lateral, and vertical restraints; locking and bolting devices whose failure could adversely affect the structural integrity of the internals; and all other locations examined on the prototype design. The inside of the vessel will be inspected with all the internals removed both before and after hot functional testing to verify that no loose parts or foreign material is present.

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

The staff finds the inspection program to be sufficient and the hot functional test to be of adequate length. On the basis of the staff's review of FSAR Section 3.9.2.4, the staff concludes that the applicant has met the relevant requirements of GDC 1 and 4 with respect to the reactor internals being designed and tested to quality standards commensurate with the importance of the safety functions being performed and being appropriately protected against dynamic effects by meeting the regulatory positions of RG 1.20 for the conduct of preoperational vibration tests and by having a preoperational vibration program planned for the reactor internals which provides an acceptable basis for verifying the design adequacy of these internals under test loading conditions comparable to those that will be experienced during operation. The combination of tests, predictive analysis, and post-test inspection provides adequate assurance that the reactor internals will, during their service lifetime, withstand the flow-induced vibrations of the reactor without loss of structural integrity. The integrity of the reactor internals in service is essential to ensure the proper positioning of reactor fuel assemblies and unimpaired operation of the control rod assemblies to permit safe reactor operation and shutdown.

3.9.2.4 Dynamic System Analysis of Reactor Internals Under Faulted Conditions

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

On the basis of the staff's review of FSAR Section 3.9N.2.5 and the load combinations and stress limits as presented in tables contained in FSAR Section 3.9.3, the staff concludes that the applicant has met the relevant requirements of GDC 2 and 4 with respect to the design of systems and components important to safety to withstand the effects of earthquakes and the appropriate combinations of the effects of normal and postulated accident conditions with the effects of the SSE by performing a dynamic system analysis that provides an acceptable basis for confirming the structural design adequacy of the reactor internals and unbroken piping loops to withstand the combined dynamic loads of a postulated LOCA and the SSE. The analysis provides adequate assurance that the combined stresses and strains in the components of the reactor coolant system and reactor internals will not exceed the allowable design stress and strain limits for the materials of construction and that the resulting deflections or displacements at any structural element of the reactor internals will not distort the reactor internals geometry to the extent that core cooling may be impaired. The methods used for component analysis have been found compatible with those used for the system analysis. The proposed combination of component and system analyses is, therefore, acceptable. The assurance of structural integrity of the reactor internals under LOCA conditions for the most adverse postulated loading event provides added confidence that the design will withstand a spectrum of lesser pipe breaks and seismic loading events.

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

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

3.9.3.1 Loading Combinations, Design Transients, and Stress Limits

The staff has reviewed the methodology used for load combinations and allowable stress limits in FSAR Section 3.9.3. The applicant's load combinations do not conform to the acceptance criteria in SRP Section 3.9.3. Specifically, the applicant has not included the LOCA loads in the evaluation of the faulted condition limits for ASME Code, Class 1, 2, and 3, balance-of-plant piping and their supports. Furthermore, the applicant has not yet addressed how the guidelines of NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary Systems," have been satisfied. The applicant intends to request an exemption from GDC 4 and the need to consider reactor coolant loop pipe breaks for Millstone Unit 3. The staff had required that current primary loop heavy component support design margins be maintained (i.e., LOCA loads included) even though the "leak-before-break" concept has been proposed. The applicant has clarified in a letter dated July 20, 1984, that LOCA loads are included in the reactor coolant loop heavy component support design. When the applicant submits his request for exemption from GDC 4, the staff will review that submittal in order to determine the extent and suitability of the exemption. This is an open item and will be addressed in a supplement to this SER.

On the basis of its review of FSAR Sections 3.9B.3.1 and 3.9N.3.1 and contingent on the satisfactory resolution of the open items, the staff's findings are as follows.

The applicant has met the requirements of 10 CFR 50.55a and GDC 1, 2, and 4 with respect to the design and service load combinations and associated stress and deformation limits specified for ASME Code Class 1, 2, and 3 components by ensuring that systems and components important to safety are designed to quality standards commensurate with their importance to safety and that these systems can accommodate the effects of normal operation as well as postulated events such as LOCAs and the dynamic effects resulting from earthquakes. The specified design and service combinations of loading as applied to ASME Code, Class 1, 2, and 3 pressure-retaining components in systems designed to meet seismic Category I standards are such as to provide assurance that in the event of an earthquake affecting the site for other service loading caused by postulated events or system operating transients, the resulting combined stresses imposed on system components will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under such loading combinations provides a conservative basis for the design of system components to withstand the most adverse combination of loading events without loss of structural integrity.

3.9.3.2 Design and Installation of Pressure Relief Device

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

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

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

Testing of valves in the EPRI program was completed by December 31, 1981.

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

(1) Valves Selection/Justification Report

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

(3) Test Condition Justification Report

(4) Safety and Relief Valve Test Report

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

In Section 5.4.13.2 of the FSAR, the applicant has stated that valves identical to the power-operated relief valves and safety valves at Millstone Unit 3 were tested as a part of the EPRI/PWR Owners Group Program. The applicant has committed to submit a Millstone Unit 2 plant-specific evaluation of this program prior to fuel load.

Additionally, by letter dated June 1, 1982, from R. C. Youngdahl to H. Denton, reports documenting block valve testing performed by EPRI were transmitted to NRC. In a letter dated July 11, 1984, from W. G. Counsil to B. J. Youngblood, the applicant has stated that Section 5.4.13.2 of the FSAR will be revised to include a commitment to prepare Millstone Unit 3 plant-specific information on block valve operability based on these reports. The applicant will also submit an analysis of the effect of as-built relief and safety valve discharge piping on valve operability. All of the above information will be submitted prior to fuel load.

On the basis of a preliminary review of the EPRI generic reports, the staff has concluded that they contain data that can be used by the applicant to prepare an Item II.D.1 plant-specific response for the valves and associated piping for Millstone Unit 3.

The staff requires that these plant-specific submittals be made before fuel loading in accordance with the schedule of NUREG-0737 and the September 29, 1981 (Generic Letter 81-36), clarification letter on this matter. Once the staff has received this information, it will report its findings in a supplement to this SER.

On the basis of its review of FSAR Section 3.9B.3.3 and contingent on the satisfactory resolution of the confirmatory item, the staff's findings are as follows.

The applicant has met the requirements of 10 CFR 50.55a and GDC 1, 2, and 3 with respect to the criteria used for the design and installation of ASME Code, Class 1, 2, and 3 overpressure relief devices by ensuring that safety and relief valves and their installations are designed to standards that are commensurate with their safety functions, and that they can accommodate the effects of discharge caused by normal operation as well as postulated events such as LOCAs and the dynamic effects resulting from the SSE. The relevant requirements of GDC 14 and 15 are also met with respect to ensuring that the reactor coolant pressure boundary design limits for normal operation, including anticipated operational occurrences, are not exceeded. The criteria used by the applicant in the design and installation of ASME Code, Class 1, 2, and 3 safety and relief valves provide adequate assurance that, under discharging conditions, the resulting stresses will not exceed allowable stress and strain limits for the materials of construction. Limiting the stresses under the loading combinations associated with the actuation of these pressure relief devices provides a conservative basis for the design and installation of the devices to withstand these loads without loss of structural integrity or impairment of the overpressure protection function.

3.9.3.3 Component Supports

The staff's review of FSAR Sections 3.9B.3.4 and 3.9N.3.4 relates to the methodology used by the applicant in the design of ASME Code, Class 1, 2, and 3 component supports. The review includes assessment of design and structural integrity of the supports. The review addresses three types of supports: plate and shell, linear, and component standard types. More information regarding the design and construction of ASME Code, Class 1, 2, and 3 component supports is required. This is an open item and will be addressed in a supplement to this SER.

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

3.9.4 Control Rod Drive Systems

The staff's review under SRP Section 3.9.4 covers the design of the control rod drive system up to its interface with the control rods. The rods and drive mechanism shall be capable of reliably controlling reactivity changes either under conditions of anticipated normal plant operational occurrences or under postulated accident conditions. The staff reviewed the information in FSAR Section 3.9N.4 relative to the analyses and tests performed to ensure the structural integrity and functionality of this system during normal operation and under accident conditions. The staff also reviewed the life-cycle testing performed to demonstrate the reliability of the control rod drive system over its 40-year life.

A detailed review of the design of the control rod drive system with respect to its capability of controlling reactivity and cooling the reactor core with appropriate margin in conjunction with either the emergency core cooling system or the reactor protection system was not performed because of the system's similarity with those at other Westinghouse plants that were found acceptable. On the basis of its review of the FSAR, the staff has not found any significant design changes in the control rod drive system for Millstone Unit 3.

On the basis of its review of the above information, the staff concludes that the design of the control rod drive system is acceptable and meets the requirements of GDC 1, 2, 14, 26, 27, and 29 and 10 CFR 50.55a. This conclusion is based on the following:

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

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

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

3.9.5 Reactor Pressure Vessel Internals

The staff's review under SRP Section 3.9.5 is concerned with the load combinations, allowable stress limits, and other criteria used in the design of the Millstone Unit 3 reactor internals. The staff has limited its review of SRP Section 3.9N.5 to include the design and analysis of the reactor internals and the deformation limits specified for those components. A detailed review of the configuration and general arrangement of the mechanical and structural internal elements was not performed because of the similarity with other Westinghouse plants which were found acceptable. On the basis of its review of the FSAR, the staff has not found any significant design changes in the reactor internals for Millstone Unit 3.

On the basis of its review of FSAR Section 3.9.5, the staff concludes that the design of reactor internals is acceptable and meets the requirements of GDC 1, 2, 4, and 10 and 10 CFR 50.55a. This conclusion is based on the following.

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

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

The specified design transients, design and service loadings, and combination of loadings as applied to the design of the reactor internals structures and components provides reasonable assurance that in the event of an earthquake or of a system transient during normal plant operation, the resulting deflections and associated stresses imposed on these structures and components would not exceed allowable stresses and deformations under such loading combinations. This provides an acceptable basis for the design of these structures and components to withstand the most adverse loading events that have been postulated to occur during service lifetime without loss of structural integrity or impairment of function.

3.9.6 *Inservice Testing of Pumps and Valves*

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

In Sections 3.9.2 and 3.9.3 of this SER, the staff discussed the design of safety-related pumps and valves in Millstone Unit 3. The load combinations and stress limits used in the design of pumps and valves ensure that the component pressure boundary integrity is maintained. In addition, the applicant will periodically test and perform periodic measurements of all the safety-related pumps and valves in accordance with Section XI of the ASME Code. The tests verify that these pumps and valves will operate successfully when called on to do so. Various parameters are periodically measured and compared with baseline measurements to detect long-term degradation of the pump or valve performance. The staff review covers the applicant's program for preservice and inservice testing of pumps and valves using the guidance of SRP Section 3.9.6 and with particular attention given to those areas of the test program for which the applicant requests relief from the requirements of Section XI of the ASME Code. The applicant must provide a commitment that the inservice testing of ASME Code, Class 1, 2, and 3 components will be in accordance with the revised rules of 10 CFR 50.55a(g).

The applicant has not submitted his program for the preservice and inservice testing of pumps and valves; therefore, the staff has not completed its review. The staff will report the resolution of these issues in a supplement to this SER.

There are several safety systems connected to the reactor coolant pressure boundary that have design pressures below the rated reactor coolant system (RCS) pressure. There are some systems also which are rated at full reactor pressure on the discharge side of pumps but have pump suction rated below RCS pressure. To protect these systems from RCS pressure, two or more isolation valves are placed in series to form the interface between the high-pressure RCS and the low-pressure system. The leaktight integrity of these valves must be ensured by periodic leak testing so that the design pressure of the low-pressure systems will not be exceeded.

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

Limiting conditions for operation (LCO) are required to be added to the Technical Specifications. These will require corrective action (that is, shutdown or system isolation) when the final approved leakage limits are not met. Also, surveillance requirements, which will give the acceptable leak rate testing frequency, shall be provided in the Technical Specifications.

Periodic leak testing of each pressure isolation valve is required to be performed (1) at least once every refueling outage, (2) after valve maintenance before return to service, and (3) for systems rated at less than 50% of RCS design pressure each time the valve has moved from its fully closed position. The applicant should provide justification for any exceptions to the above testing requirements. The testing interval should on an average be approximately 1 year. Leak testing should also be performed after all disturbances to the valves are complete, before reaching power operation following a refueling outage, and following maintenance performed on the valve.

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

Leak rates greater than 1 gpm will be considered if the leak rate changes are below 1 gpm from the previous test leak rate or if system design precludes measuring 1 gpm with sufficient accuracy. These items will be reviewed on a case-by-case basis.

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

The applicant has provided a list of Millstone Unit 3 pressure isolation valves to be included in the leak rate testing program. However, the applicant has not committed to the staff's position on acceptable leak rates. This is an open item and will be addressed in a supplement to this SER.

3.10 Seismic and Dynamic Qualification of Seismic Category I Mechanical and Electrical Equipment

3.10.1 Seismic and Dynamic Qualification

The staff's evaluation of the adequacy of the applicant's program for qualification of electrical and mechanical equipment important to safety for seismic and dynamic loads consists of (1) a determination of the acceptability of the procedures used, standards followed, and completeness of the program in general and (2) an onsite audit of selected equipment items to develop the basis for the staff's judgment on the completeness and adequacy of the implementation of the entire seismic and dynamic qualification program.

The Seismic Qualification Review Team (SQRT), which consists of reviewers from the NRC staff and consultants from Brookhaven National Laboratory (BNL), has reviewed the methodology and procedures of the seismic and dynamic qualification program contained in FSAR Sections 3.9.2, 3.9.3, 3.10, and Appendix 3a to FSAR Section 3.10. The SQRT has concluded that, except for the areas discussed below, the information provided in the FSAR does meet the intent of the staff's acceptance criteria as specified in SRP Section 3.10, which requires the applicant's qualification program to meet the requirements and recommendations of IEEE Std. 344-1975 and the regulatory positions of RGs 1.61, 1.89, 1.92, and 1.100 and to provide adequate assurance that such equipment will function properly under all imposed design and service loads including the loadings imposed by the safe shutdown earthquake, postulated accidents, and loss-of-coolant accidents.

The following areas need further clarification or resolution:

(1) The applicant should describe his seismic qualification program for nuclear steam supply system (NSSS) safety-related mechanical equipment in the FSAR.

(2) The applicant needs to clarify how the as-built mounting condition is determined to be equivalent to that used in qualification and how the required response spectra at the mounting location are determined to equal or exceed that used in qualification.

(3) The applicant needs to clarify how the conservative restrictions placed on allowable piping loads transmitted to the pump and valve bodies for NSSS-supplied items have been demonstrated not to cause detrimental deflections of the active components. The applicant should also clarify how this issue is resolved for balance-of-plant equipment.

(4) Although the applicant has committed to follow the requirements and recommendations of IEEE Std. 344-1975 and RG 1.100, the methods for handling aging and sequential testing in the seismic qualification of both electrical and mechanical equipment should be clarified. In addition, the applicant should commit to establish a maintenance and surveillance program to maintain equipment in a qualified status throughout the life of the plant.

(5) The applicant should clarify how Westinghouse generically qualified equipment is verified as being applicable to Millstone Unit 3.

(6) In cases where equipment was qualified by using single axis and/or single frequency testing, the equipment should be identified and in each case the justification for the use of these procedures should be given.

(7) There should be a list of types of equipment that clearly shows the methods used for qualification. This list also should address which standards are met, particularly those cited in SRP Section 3.10.

The applicant provided information concerning Items 5 and 6 in a letter from W. G. Counsil to B. J. Youngblood dated May 15, 1984. Therefore, the staff considers these two items resolved.

The applicant should submit FSAR amendments to resolve the identified FSAR deficiencies. In addition, the SQRT will follow the applicant's effort closely and will confirm its implementation during the onsite audit. During the plant site audit, the staff will review in detail the applicant's implementation of the qualification program to confirm that all applicable loads and combinations of loads have been defined, operability has been verified through appropriate tests and analyses, assemblies rather than individual components have been verified operable, and that for all safety-related equipment operability can be ensured throughout the plant's life. A substantial portion (85%-90%) of the equipment must be qualified, documented in an auditable manner, and installed on site before an onsite audit by the SQRT can be performed. Whenever the applicant will indicate that his work is substantially complete, the SQRT will conduct an onsite audit shortly thereafter. The staff shall report the results of its audit and the followup and resolution of its concerns described above in the final SER.

3.10.2 Pump and Valve Operability Assurance

The staff's evaluation of the adequacy of the applicant's pump and valve operability assurance program consists of (1) a determination of the acceptability of the procedures used, standards followed, and completeness of the program in general, and (2) an onsite audit of selected equipment items to develop the basis for the staff judgment on the completeness and adequacy of the implementation of the entire pump and valve operability assurance program.

The Pump and Valve Operability Review Team (PVORT), which consists of reviewers from the NRC staff and consultants from BNL, has reviewed the methodology and procedures of the pump and valve operability assurance program contained in FSAR Section 3.9.3.2. The PVORT has concluded that, except for the areas discussed below, the information provided in the FSAR meets the intent of the staff's acceptance criteria as specified in SRP Section 3.10, which requires that the applicant's qualification program (1) meet the requirements and recommendations of IEEE Std. 323-1974, the regulatory positions of RG 1.148, and the draft standards ANSI/ASME N551.1, N551.2, and N551.4 and ANSI B.16.41 and N41.6 and (2) provide adequate assurance that the equipment will function properly under all imposed design and service loads including the loadings imposed by the safe shutdown earthquake, postulated accidents, and loss-of-coolant accidents.

The following areas need further clarification or resolution:

(1) The applicant did not provide the design criteria for pump and valve internal parts, such as valve discs and pump shafts. A review of qualification documents is necessary to determine whether the pump and valve internals are adequately qualified.

(2) SRP Section 3.10, Paragraph II.1.a(2), indicates that equipment should be tested in the operational condition; that is, normal plant loadings should be superimposed on seismic and dynamic loads, including thermal, flow-induced loads and degraded flow conditions. The FSAR should clearly indicate how this requirement is met.

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

(4) There should be a list of types of equipment that clearly shows the methods used for qualification. This list should also address which standards are met, in particular those cited in SRP Section 3.10.

(5) Clarification of how aging was incorporated in the qualification process should be contained in the FSAR. In addition, the applicant should commit to establish a maintenance and surveillance program to maintain equipment in a qualified status throughout the life of the plant.

(6) Further justification of the independent qualification of pumps, valves, prime movers, and actuators versus their assembly qualification is also required.

The applicant should submit FSAR amendments to resolve the identified FSAR deficiencies. In addition, the PVORT will follow the applicant's effort closely and will confirm its implementation during the onsite audit. During the plant site audit the staff will review in detail the applicant's implementation of the qualification program to confirm that all applicable loads and combinations of loads have been defined, operability has been verified through appropriate tests and analyses, assemblies rather than individual components have been verified operable, and that for all safety-related equipment operability can be ensured throughout the plant's life. A substantial portion (85%-90%) of the equipment must be qualified, documented in an auditable manner, and installed on site before an onsite audit by the PVORT can be performed. Whenever the applicant will indicate that his work is substantially complete, the PVORT will conduct an onsite audit shortly thereafter. The staff shall report the results of its audit and the followup and resolution of its concerns described above in a supplement to this SER.

3.11 Environmental Qualification of Electric Equipment Important to Safety and Safety-Related Mechanical Equipment

3.11.1 Introduction

Equipment that is used to perform a necessary safety function must be demonstrated capable of maintaining functional operability under all service conditions postulated to occur during its installed life for the time it is required to operate. This requirement, which is embodied in GDC 1 and 4 and in Sections III, XI, and XVII of Appendix B to 10 CFR 50, is applicable to equipment located inside as well as outside containment. More detailed requirements and guidance relating to the methods and procedures for demonstrating this capability for electrical equipment have been set forth in 10 CFR 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," and NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment." NUREG-0588 supplements IEEE Std. 323 and various RGs and industry standards.

3.11.2 Background

NUREG-0588 was issued in December 1979 to promote a more orderly and systematic implementation of electrical equipment qualification programs by industry and to provide guidance to the NRC staff for use in ongoing licensing reviews. The positions contained in this document provide guidance on (1) how to establish environmental service conditions, (2) how to select methods that are considered appropriate for qualifying equipment in different areas of the plant, and (3) other factors such as margin, aging, and documentation. In February 1980, the NRC requested certain near-term operating license (OL) applicants to review and evaluate the environmental qualification documentation for each item of safety-related electric equipment that could be exposed to a harsh environment and to identify the degree to which their qualification program complies with the staff positions described in NUREG-0588.

IE Bulletin 79-01B, "Environmental Qualification of Class 1E Equipment," issued January 14, 1980, and its supplements dated February 29, September 30, and October 14, 1980, established environmental qualification requirements for operating reactors. This bulletin and its supplements were provided to OL applicants for consideration in their reviews.

A final rule on environmental qualification of electric equipment important to safety for nuclear power plants became effective on February 22, 1983. This rule, 10 CFR 50.49, specifies the requirements to be met for demonstrating the environmental qualification of electrical equipment important to safety located in a harsh environment. In accordance with this rule, equipment for Millstone Unit 3 may be qualified to the criteria specified in Category II of NUREG-0588.

The qualification requirements for mechanical equipment are principally contained in Appendices A and B of 10 CFR 50. The qualification methods defined in NUREG-0588 can also be applied to mechanical equipment.

The applicant has provided some equipment qualification information in FSAR Section 3.11.

3.11.3 Completeness of the Environmental Qualification Program

The staff has reviewed the information submitted by the applicant in Section 3.11 of the Millstone Unit 3 FSAR. However, before the staff can complete its review of the license application, it is necessary that the applicant comply with the Commission's requirements applicable to environmental qualification contained in 10 CFR 50.49 for electrical equipment important to safety; GDC 4; and Appendix B, 10 CFR 50, Sections III, XI, XVII.

As a result of the issuance of 10 CFR 50.49, some of the information requested in SRP Section 3.11 and RG 1.70, Section 3.11, is no longer required for staff review. Other new information is required, however, and is defined in this guidance. By using these guidelines to demonstrate compliance with the Commission's regulations, applicants can significantly reduce the need for requests for additional information from the NRC staff. The information required may be submitted in FSAR Section 3.11 or in a separate submittal. If the latter approach is chosen, Section 3.11 should reference the information in the environmental qualification program submittal. The following guidelines summarize the information to be furnished to the staff:

(1) The applicable criteria should be identified and shown to have been incorporated into the environmental qualification program.

(2) The systems and components selected for harsh environment qualification should be identified and demonstrated to be complete. Correlation with FSAR Table 3.2.1 should be provided for identification of safety-related equipment. Safety-related equipment exempt from the requirements for harsh environment qualification should be justified.

The scope of safety-related electrical equipment that should be identified is defined in 10 CFR 50.49(b)(1).

To demonstrate compliance with 10 CFR 50.49(b)(2) concerning nonsafety-related electrical equipment whose failure could prevent the satisfactory accomplishment of safety functions, and 10 CFR 50.49(b)(3), postaccident monitoring equipment, the following information should be provided:

(a) A list of all nonsafety-related electrical equipment located in a harsh environment whose failure under postulated environmental conditions could prevent the satisfactory accomplishment of safety functions by the safety-related equipment. A description of the method used to identify this equipment also must be included. The nonsafety-related equipment identified must be included in the environmental qualification program.

(b) A list of all postaccident monitoring equipment currently installed, or that will be installed before plant operation, that is specified as Category 1 and 2 in Revision 2 of RG 1.97 and is located in a harsh environment. The equipment identified must be included in the environmental qualification program. In addition, any TMI Action Plan equipment previously committed to installation before fuel loading should be identified and qualified in accordance with the applicable criteria.

(3) The normal, abnormal, and accident environments should be provided for each plant zone. References should be made to other FSAR sections, where appropriate, for methodologies used to determine accident environments. The requirement for calculation of the radiation doses to equipment in

close proximity to recirculating fluid systems inside and outside containment for LOCA events in which the primary system does not depressurize should be incorporated into the program (Item II.B.2 of TMI Action Plan, NUREG-0737). The time-dependent environments should be defined for accident conditions.

(4) The qualification methodology should be summarized by reference to appropriate criteria (RGs, industry standards, etc.) and should address the following areas:

(a) margin
(b) aging
(c) dose rate and synergistic effects
(d) use of analysis for qualification
(e) the maintenance/surveillance program, in particular its conformance with RG 1.33 and the industry standard it endorses, and its use in the aging program for equipment qualification

(5) All equipment located in a harsh environment should be identified by its tag number, and its location and operability time provided. Equipment located in a mild environment need not be included in this list. For electrical equipment, the information requested in Appendix E of NUREG-0588, and SRP Section 3.11 concerning test results should be submitted. An acceptable format for this information was provided with IE Bulletin 79-01B in the form of "SCEW sheets." Other formats providing the same information may be submitted however.

The information requested in Item 4 of Appendix E, NUREG-0588, need not be submitted but should be available for audit by the staff.

(6) For mechanical equipment, the staff review will concentrate on materials that are sensitive to environmental effects (e.g., seals, gaskets, lubricants, fluids for hydraulic systems, and diaphragms). A review and evaluation should be performed by the applicant that includes the following:

(a) identification of safety-related mechanical equipment located in a harsh environment, including required operating time

(b) identification of nonmetallic subcomponents of this equipment

(c) identification of the environmental conditions this equipment must be qualified for; the environments defined in the electrical equipment program are also applicable to mechanical equipment

(d) identification of nonmetallic material capabilities

(e) evaluation of environmental effects

The list of equipment identified should be submitted. From this list the staff will select approximately three items of mechanical equipment for which documentation of their environmental qualification should be provided for review. Also, the results of the review should be provided for all mechanical equipment in a harsh environment and corrective actions identified. Justification for interim operation must be submitted before fuel loading for any mechanical equipment whose qualification cannot be established.

Once the above information is received, the staff will review the applicant's environmental qualification program for compliance with 10 CFR 50.49 and Appendix B, 10 CFR 50, and request any additional information needed to establish its acceptability. The staff will then perform an audit review of the electrical equipment environmental qualification files and associated installed equipment. Following this audit the results of the staff's review and evaluation will be reported in a supplement to this SER. The staff must be able to conclude that the applicant is in compliance with 10 CFR 50.49 and Appendix B, 10 CFR 50, before an operating license can be issued.

Table 3.1 Reliability criteria

	Probability, yr^{-1}		
	Favorably oriented	Unfavorably oriented	Required licensee action
(1)	$P_1 < 10^{-4}$	$P_1 < 10^{-5}$	This is the general, minimum reliability requirement for loading the turbine and bringing the system on line.
(2)	$10^{-4} < P_1 < 10^{-3}$	$10^{-5} < P_1 < 10^{-4}$	If during operation this condition is reached, the turbine may be kept in service until the next scheduled outage, at which time the licensee is to take action to reduce P_1 to meet the appropriate criterion, (1) above, before returning the turbine to service.
(3)	$10^{-3} < P_1 < 10^{-2}$	$10^{-4} < P_1 < 10^{-3}$	If during operation this condition is reached, the turbine is to be isolated from the steam supply within 60 days, at which time the licensee is to take action to reduce P_1 to meet the appropriate criterion, (1) above, before returning the turbine to service.
(4)	$10^{-2} < P_1$	$10^{-3} < P_1$	If at any time during operation this condition is reached, the turbine is to be isolated from the steam supply within 6 days, at which time the licensee is to take action to reduce P_1 to meet the appropriate criterion, (1) above, before returning the turbine to service.

4 REACTOR

4.1 Introduction

The Millstone Unit 3 nuclear steam supply system (NSSS) is supplied by Westinghouse Electric Corporation and is designed to operate at a core thermal power of 3,411 MWt. Sufficient margin exists to ensure that fuel damage will not occur during steady-state operation or anticipated operational occurrences.

The NSSS, a four-loop design, has a primary coolant flow rate of 140.8×10^6 lb per hour. The reactor coolant and moderator is light water at a nominal system pressure of 2,250 psia. The reactor core consists of 193 fuel assemblies of similar mechanical design, but different fuel pellet enrichments. Each assembly contains a 17 x 17 array of 264 fuel rods. The center position in each assembly is used for incore instrumentation. The remaining 24 positions in the fuel assembly have guide thimbles for the rod cluster control assemblies. There are 24 absorber rods per cluster.

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

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

4.2 Fuel Design

The Millstone Unit 3 fuel assembly described in the FSAR is a 17 x 17 array of fuel rods having a diameter of 0.374 in. This design will be referred to as the standard fuel assembly (SFA) in the following paragraphs.

Section 4.2 of the Final Safety Analysis Report (FSAR) presents the design bases for the SFA. For the Westinghouse (W) analysis, plant design conditions are divided into four categories of operation that are consistent with traditional industry classfication (American National Standards Institute (ANSI) Stds. N18.2-1973 and N-212-1974): Condition I is normal operation, Condition II is incidents of moderate frequency, Condition III is infrequent incidents, and Condition IV is limiting faults. Fuel damage is then related to these conditions of operation, which are coupled to the fuel design bases and design limits. The subsections of the design bases section address topics such as (1) cladding, (2) fuel material, (3) fuel rod performance, (4) spacer grids, (5) fuel assembly, (6) in-core control components, and (7) surveillance program. As part of the discussion of the cladding design bases, material and mechanical properties, stress-strain limits, vibration and fatigue, and chemical properties are also presented. A similar approach is taken for the other major subtopics.

The staff review and safety evaluation follow SRP Section 4.2 (NUREG-0800). The objectives of this fuel system safety review are to provide assurance that

(1) the fuel system is not damaged as a result of normal operation and anticipated operational occurrences, (2) fuel system damage is never so severe as to prevent control rod insertion when it is required, (3) the number of fuel rod failures is not underestimated for postulated accidents, and (4) coolability is always maintained. "Not damaged" is defined as meaning that fuel rods do not fail, that fuel system dimensions remain within operational tolerances, and that functional capabilities are not reduced below those assumed in the safety analysis. This objective implements General Design Criterion (GDC) 10, and the design limits that accomplish this are called specified acceptable fuel design limits (SAFDLs). "Fuel rod failure" means that the fuel rod leaks and that the first fission product barrier (the cladding) has, therefore, been breached. Fuel rod failures must be accounted for in the dose analysis required by 10 CFR 100 for postulated accidents. "Coolability," which is sometimes called "coolable geometry," means, in general, that the fuel assembly retains its rod-bundle geometrical configuration with adequate coolant channels to permit removal of residual heat after a severe accident. The general requirements to maintain control rod insertability and core coolability appear repeatedly in the GDC (GDC 27 and 35). Specific coolability requirements for the loss-of-coolant accidents are given in 10 CFR 50.46.

To meet the above-stated objectives of the fuel system review, the following specific areas are critically examined: (1) design bases, (2) description and design drawings, (3) design evaluation, and (4) testing, inspection, and surveillance plans. In assessing the adequacy of the design, several items involving operating experience, prototype testing, and analytical predictions are weighed in terms of specific acceptance criteria for fuel system damage, fuel rod failure, and fuel coolability. Recently, Westinghouse developed the optimized fuel assembly (OFA), which is described in WCAP-9500. The staff has approved WCAP-9500 (Rubenstein, May 15, 1981, and Tedesco, May 22, 1981). The OFA design also consists of a 17 x 17 array of fuel rods having a diameter of 0.360 in., which is somewhat smaller than the rod diameter in the SFA. Because the format of WCAP-9500 followed RG 1.70, some of the fuel design bases and design limits for the OFA were not presented in WCAP-9500 in a form that permitted cross-checking by the NRC with the acceptable criteria provided in SRP Section 4.2. Therefore, several questions were issued (Rubenstein, August 8, 1980) to clarify the design bases and limits. Responses to those questions are contained in letters from Westinghouse (Anderson, January 12, 1981, and April 21, 1981). These responses are applicable to the SFA to be used in Millstone Unit 3 as well (Petrick, September 9, 1981). Reference to these questions and answers will be made at several places in the review that follows.

4.2.1 Design Bases

Design bases for the safety analysis address fuel system damage mechanisms and suggest limiting values for important parameters so that damage will be limited to acceptable levels. For convenience, acceptance criteria for these design limits are grouped into three categories in the SRP: (1) fuel system damage criteria, which are most applicable to normal operation (W plant Condition I), including anticipated operational occurrences (W plant Condition II); (2) fuel rod failure criteria, which apply to normal operation (W plant Condition I), anticipated operational occurrences (W plant Condition II), and postulated accidents (W plant Conditions III and IV); and (3) fuel coolability criteria, which apply to postulated accidents (W plant Conditions III and IV).

4.2.1.1 Fuel System Damage Criteria

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

(1) Cladding Design Stress

In FSAR Section 4.2.1.1, it is indicated that the cladding stresses under Conditions I and II are less than the Zircaloy yield stress, with due consideration of temperature and irradiation effects. The design basis for fuel rod cladding stress as given in the responses to Q231.2* is that the fuel system will not be damaged as a result of excessive fuel rod cladding stresses. The design limit for fuel rod cladding stress under the Condition I and II modes of operation is that the volume-averaged effective stress calculated with the von Mises equation, considering interference resulting from uniform cylindrical pellet-to-cladding contact (caused by pellet thermal expansion and swelling, uniform cladding creep, and fuel rod/coolant system pressure differences), is less than the Zircaloy 0.2% offset yield stress as affected by temperature and irradiation. This is a traditional limit consistent with previous Westinghouse design practice, but with credit being taken by Westinghouse for irradiation-induced strengthening. The staff has approved (Rubenstein, June 6, 1983) WCAP-9179, Revision 1, which includes approval for taking such credit.

(2) Cladding Design Strain

With regard to cladding strain, a design limit for fuel rod cladding plastic tensile creep (resulting from uniform cladding creep and uniform cylindrical fuel pellet swelling and thermal expansion) of less than 1% from the unirradiated condition is given in the response to Q231.2 and in Section 4.2.1.1 of the FSAR. Furthermore, the total tensile strain transient limit (resulting from uniform cylindrical pellet thermal expansion during the transient) is stated to be less than 1% from the pretransient value. Although the staff has not explicitly reviewed the supporting data for normal operation (Condition I), that value appears to be consistent with past practice (no numerical value for normal operation cladding strain is provided as an acceptance criterion in the SRP), and thus there is reasonable assurance that 1% total plastic creep strain is an acceptable design limit for normal operation, including Condition I power changes (load following). For transient-induced deformation, the SRP indicates that 1% uniform cladding strain is an acceptable damage limit that should preclude some types of pellet/cladding interaction (PCI) failures. Such a limit, however, although consistent with past practice, should not be construed to be a broadly applicable PCI damage limit because there is ample evidence (Tokar, November 14, 1979) that PCI failures can occur at less than 1% uniform cladding

*All questions and responses referred to in this manner were part of the review of WCAP-9500, and the first application of the SFA, on the Shearon Harris docket and will be found in the correspondence previously cited. References to the FSAR refer to the Millstone Unit 3 FSAR.

strain. Westinghouse has indicated in response to Q231.24 that 1% plastic strain from the pretransient value is not meant to serve as a broadly applicable PCI criterion. Nevertheless, the 1% cladding transient plastic strain criterion appears to be an acceptable design limit for the type of application indicated in SRP Section 4.2. For fuel assembly structural design, Westinghouse set design limits on stresses and deformations resulting from various non-operational, operational, and accident loads. As indicated in Section 4.2.1.5 of the FSAR, the stress categories and strength theory presented in Section III of the ASME Code are used as a general guide. This is consistent with acceptance criterion II.A.1(a) of SRP Section 4.2 and is acceptable.

(3) Strain Fatigue

The strain fatigue criteria given in the response to Q231.2 and in Section 4.2.3.3 of the FSAR are the same as those described in SRP Section 4.2 (a safety factor of 2 on stress amplitude or of 20 on the number of cycles) and are, therefore, acceptable.

(4) Fretting Wear

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

In Section 4.2.1.1, it is indicated that potential fretting wear resulting from vibration is prevented, ensuring that the stress-strain limits are not exceeded during the design life. From the response to Q231.5 it can also be seen that the Westinghouse design basis for fretting wear is that fuel rods shall not fail during Condition I and II events. Furthermore, Westinghouse does not use an explicit fretting wear limit in its stress and fatigue analysis for fuel rods. However, Westinghouse does use a value (proprietary) of wall thickness as a general guide in evaluating cladding imperfections, including fretting wear. Cladding imperfections including fretting wear are thus considered in the stress and fatigue analysis, albeit in a qualitative manner. In view of the apparently small effects of these defects and large stress and fatigue margins (see Section 4.2.3.1(4) of this SER), this design method is acceptable.

The design basis for guide thimble tubes (see response to Q231.41) is that the thinning of the guide thimble tube walls should not result in the failure of the fuel assembly structural integrity or functionability of the guide thimble tubes. The staff finds this to be an acceptable design basis.

With regard to a design limit for guide thimble tube wear, Westinghouse has determined from stress analyses that the most limiting load on the fuel assembly structure is that which might occur during a fuel-handling accident. For the analysis of this accident, Westinghouse uses a design criterion of 6 g, as noted in Section 4.2.1.1 of the FSAR. This design limit is therefore used for degraded guide thimble tubes and has been previously accepted for Westinghouse fuels.

(5) Oxidation and Crud Buildup

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

SRP Section 4.2 identifies cladding oxidation, hydriding, and crud buildup as potential fuel system damage mechanisms. Hydriding is discussed in Section 4.2.1.2(1). Because of the increased thermal resistance of these layers, there is an increased potential for elevated temperature within the fuel as well as the cladding. Because the effect of oxidation and crud layers on fuel and cladding temperature is a function of several different parameters (such as heat flux and thermal-hydraulic boundary conditions), a design limit on oxide or crud layer thickness does not, per se, preclude fuel damage as a result of these layers. Rather, it is necessary that these layers be appropriately considered in other temperature-related fuel system damage and failure analyses. The staff finds this approach (e.g., see FSAR Section 4.4.5.2) taken by Westinghouse in the design of the SFA acceptable.

(6) Rod Bowing

Fuel rod bowing is a phenomenon that alters the pitch dimensions between adjacent fuel rods. Bowing affects local nuclear power peaking and the local heat transfer to the coolant. Rather than placing design limits on the amount of bowing that is permitted, the effects of bowing are included in the safety analysis (see FSAR Sections 4.2.3.1 and 4.2.3.3). This is consistent with the SRP and is acceptable. The methods used for predicting the degree of rod bowing are evaluated in Section 4.2.3.1(6), and the impact of the resulting bow magnitude is evaluated in Sections 4.3 and 4.4.

(7) Axial Growth

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

The Westinghouse design bases for core component rods are that (a) dimensional stability and cladding integrity are maintained during Condition I and II events and (b) these components do not interfere with shutdown during Condition III and IV events.

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

For the Zircaloy cladding and fuel assembly components, the axial-dimensional behavior is governed by creep (resulting from mechanical or hydraulic loading) and irradiation growth. The critical tolerances that require controlling are (a) the spacing between the fuel rods and the fuel assembly (shoulder gap) and (b) the spacing between the fuel assemblies and the core internals. Failure to adequately design for the former may result in fuel rod bowing, and for the latter may result in collapse of the holddown springs. With regard to inadequately designed shoulder gaps, problems have been reported (Schenk, 1973; Kuffer and Lutz, 1973; FSAR of R. E. Ginna Unit 1 (Rochester Gas and Electric Corporation, 1972); and Rubenstein, June 17, 1983) in foreign (Obrigheim and Beznau) and domestic (Ginna and Arkansas Unit 2) plants that have necessitated predischarge modifications to fuel assemblies.

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

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

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

(8) Fuel Rod and Nonfuel Rod Pressures

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

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

To simplify the analysis of fuel system damage resultng from excessive rod internal pressure, the SRP states that rod internal gas pressure should remain below the nominal system pressure during normal operation unless otherwise justified. Westinghouse has elected to justify limits other than that provided in the SRP.

For the fuel rods, revised internal rod pressure criteria as described in WCAP-8963, an approved (Stolz, May 19, 1978) topical report, were used in the FSAR. Briefly stated, these criteria (FSAR Section 4.2.1.3) allow the fuel rod internal pressure to exceed the system pressure under certain conditions:

(a) The internal pressure is limited so that the fuel-to-cladding gap does not increase during steady-state operation.

(b) Extensive departure from nucleate boiling (DNB) propagation does not occur for postulated transients and accidents.

These criteria have been previously approved and remain acceptable.

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

(9) Assembly Liftoff

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

(10) Control Material Leaching

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

4.2.1.2 Fuel Rod Failure Criteria

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

(1) Internal Hydriding

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

(2) Cladding Collapse

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

(3) Overheating of Cladding

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

(4) Overheating of Fuel Pellets

As a second method of avoiding cladding failure resulting from overheating, Westinghouse avoids centerline fuel pellet melting as a design basis. This design basis is the same as that in the SRP and is therefore acceptable.

The design limit (FSAR Section 4.4.1.5.2) corresponding to the design basis given above is that, during modes of operation associated with Condition I and Condition II events, there is at least a 95% probability that the peak kW/ft fuel rod will not exceed the uranium dioxide melting temperature. This design limit is an acceptable representation of the design basis given previously.

(5) Pellet/Cladding Interaction

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

(6) Cladding Rupture

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

4.2.1.3 Fuel Coolability Criteria

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

(1) Fragmentation of Embrittled Cladding

For LOCA analysis (FSAR Section 15.6.5.1), Westinghouse uses the acceptance criteria of 2200°F on peak cladding temperature (PCT) and 17% on maximum cladding oxidation as prescribed by 10 CFR 50.46.

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

The Westinghouse 2700°F PCT limit was selected taking into consideration the short time (a few seconds) that the fuel is calculated to be in DNB for a locked-rotor-type event and the fact that the PCT and total metal-water reaction at the fuel hot spot would not be expected to impact fuel coolable geometry. Although this limit has been used by Westinghouse for several years, the basis for the limit has only recently been reviewed. However, an assessment by the staff (Van Houten, February 23, 1981) of the available experimental information indicates that fuel rod cladding will, indeed, retain its rod-like geometry after exposure to the short-term (a few seconds) PCT of 2700°F. That conclusion is based on four Japanese reports (Shiozawa, 1979; Hoshi, 1980; Japanese Atomic Energy Research Institute, 1980; and Fukishiro, 1980) that describe experimental results for reactor test programs reported since 1979. The staff, therefore, concludes that there is reasonable assurance that the 2700°F PCT limit for short-term events such as locked rotor is an acceptable coolability limit for the Westinghouse SFA design.

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

(2) <u>Violent Expulsion of Fuel Material</u>

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

The design limits given in the FSAR are:

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

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

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

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

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

(3) <u>Cladding Ballooning and Flow Blockage</u>

In the LOCA analyses for SFA-designed plants, empirical models are used to predict the degree of cladding circumferential strain and assembly flow blockage at the time of hot-rod and hot-assembly burst. These models are each expressed as functions of differential pressure across the cladding wall. There are no specific design limits associated with ballooning and blockage, and the ballooning and blockage models are portions of the ECCS evaluation model, which is documented in WCAP-8301 and WCAP-8302.

(4) Structural Damage From External Forces

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

4.2.2 Description and Design Drawings

The description of fuel system components, including fuel rods, bottom and top nozzles, guide and instrument thimbles, grid assemblies, rod cluster control assemblies, burnable poison rods, neutron sources, and thimble plugs, is contained in FSAR Section 4.2.2. In addition, FSAR Table 4.3.-1 provides numerical values for various core component parameters. Although each parameter listed in SRP Section 4.2.2 is not provided in the FSAR, enough information is provided in sufficient detail to provide a reasonably accurate representation of the SFA design and this information is thus acceptable.

4.2.3 Design Evaluation

Design bases and limits were presented and discussed in SER Section 4.2.1. In this section, Westinghouse methods of demonstrating that the SFA fuel design meets the design criteria that have been established are reviewed.

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

4.2.3.1 Fuel System Damage Evaluation

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

(1) Cladding Design Stress

As indicated in the response to Q231.2, Westinghouse used its performance analysis and design (PAD) code, WCAP-8720, to analyze cladding stress. That code has been reviewed and found acceptable (Stolz, February 9, 1979). Typical calculated design values for cladding effective stress provided in the response to Q231.2 are stated to be considerably below the 0.2% offset yield stress design limit.

(2) Cladding Design Strain

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

(3) <u>Strain Fatigue</u>

As indicated in the response to Q231.2, Westinghouse used its approved PAD code for the strain range and strain fatigue life usage analysis. Experimental data obtained from Westinghouse testing programs (see FSAR Section 4.2.3.3) were used by Westinghouse to derive the Zircaloy fatigue design curve, according to the response to Q231.4. For a given strain range, the number of fatigue cycles is less than that required for failure, considering a minimum safety factor of 2 on stress amplitude or a minimum safety factor of 20 on the number of cycles (the fatigue usage factor is less than 1.0). The computations were performed with an approved code. The staff, therefore, concludes that the SFA fatigue design basis has been met.

(4) <u>Fretting Wear</u>

With regard to the Westinghouse fretting analysis of the fuel cladding, the staff concludes:

(a) Cladding fretting and fuel vibration have been experimentally investigated, as shown in WCAP-8278 (and nonproprietary version WCAP-8279) and noted in FSAR Section 4.2.3.1. The staff has approved WCAP-8278 (and WCAP-8279) (Rubenstein, March 19, 1981).

(b) The out-of-pile flow tests and analyses (WCAP-9401) to determine the magnitude of fretting wear that is anticipated for the OFA design have been previously reviewed and found acceptable (Rubenstein, April 23, 1981). These analyses are also acceptably conservative for SFA applications.

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

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

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

Therefore, the staff concludes that the SFA fuel rods will perform adequately with respect to fretting wear.

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

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

(5) Oxidation and Crud Buildup

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

(6) Rod Bowing

In FSAR Section 4.2.3.1(4) (Amendment 5, November 1983), it is indicated that the model in WCAP-8691, Revision 1 (nonproprietary version WCAP-8692, Revision 1), was used for evaluation of fuel rod bowing (letter from applicant dated March 16, 1984). That report (WCAP-8691, Revision 1) was approved (Rubenstein, October 25, 1982) by the staff. Furthermore, FSAR Section 4.4.2.2.5 indicates that the Millstone Unit 3 core maintains sufficient margin to accommodate full- and low-flow DNBR penalties using the approved model. Consequently, the staff concludes that rod bow has been adequately addressed for the SFA design.

(7) Axial Growth

Relative to the discussion in Section 4.2.1 on stainless steel growth, the staff is aware of supporting information (Appleby, 1972, and Bloom, 1972) that was not cited by Westinghouse, but which also implies that irradiation growth of stainless steel should not be significant at the temperatures and fluences that are associated with PWR operation. Furthermore, because it is unaware of any operating experience that indicates axial-growth-related problems in Westinghouse NSSS plants, the staff concludes that Westinghouse has made sufficient accommodations for control, source, and burnable poison rod axial rod growth in its NSSS designs.

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

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

(8) Fuel Rod and Nonfuel Rod Pressures

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

The analysis of nonfueled rod internal pressure for the SFA is generally based on Section III, Article NG-3000, of the ASME Code (see FSAR Section 4.2.1.6). Control rod, neutron source rod, and burnable poison rod cladding is 10% cold-worked Type 304 stainless steel, which is not covered by the ASME Code. Westinghouse therefore defines as the stress limit an intensity value Sm equal to 2/3 of the material yield stress. The yield stress for this material is approximately 62,000 psi. A strain limit of 1% also applies to the cladding. Predicted maximum values of rod internal pressure have been provided in the response to Q231.2, and they are well below those imposed by the cladding stress and strain limits.

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

(9) Assembly Liftoff

In response to the staff's question on this topic, Westinghouse has confirmed that momentary liftoff will occur only during a turbine overspeed transient (this is also stated in FSAR Section 4.4.6.2). Westinghouse has further found that (a) proper reseating will occur after momentary liftoff, (b) damage to adjacent assemblies will not occur even if one assembly is fully lifted and the adjacent ones remain seated, and (c) no ill consequences of momentary liftoff are expected. The staff concludes, therefore, that fuel assembly liftoff has been adequately addressed for the SFA design.

(10) Control Material Leaching

Although the design basis for the SFA control rods is to maintain cladding integrity and the probability of control rod cladding failures appears to be quite low, the staff has considered the corrosion behavior of the Millstone Unit 3 control material and burnable poison and concludes that a breach in the cladding should not result in serious consequences because the Ag-In-Cd or hafnium absorber material and the poison material (borosilicate glass) are relatively inert.

4.2.3.2 Fuel Rod Failure Evaluation

The following paragraphs discuss the evaluation of (1) the ability of the SFA fuel to operate without failure during normal operation and anticipated transients and (2) the accounting for fuel rod failures in the applicant's accident analysis. The fuel rod failure criteria described in Section 4.2.1.2 were used for this evaluation.

(1) Internal Hydriding

Westinghouse has used moisture and hydrogen control limits in the manufacture of earlier fuel types and has found that typical end-of-life cladding hydrogen levels are less than 100 ppm - a level below which hydride blister formation is not anticipated in fuel cladding.

The staff therefore concludes that reasonable evidence has been provided that hydriding as a fuel failure mechanism will not be significant in the SFA.

(2) Cladding Collapse

In calculating the time at which cladding collapse will occur, Westinghouse uses the generic methods described in WCAP-8377, which is approved (Stello, January 14, 1975) for licensing applications. Inputs to the analysis include cladding ovality, helium prepressurization, free volume of the fuel rod, and limiting power histories.

Cladding collapse evaluations using the approved methods have been performed on the Millstone Unit 3 fuel for Regions 1, 2, and 3 and confirm that cladding collapse times are in excess of the projected lifetime of the fuel (letter from applicant dated May 11, 1984).

(3) Overheating of Cladding

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

(4) Overheating of Fuel Pellets

The design evaluation of the fuel centerline melt limit is performed with the Westinghouse fuel performance code, PAD-3.3 (WCAP-8720). This code, which the staff has approved (Stolz, February 9, 1979), is also used to calculate initial conditions for transients and accidents described in SRP Chapter 15 (see Section 4.2.3.3(1) below for further comments on PAD-3.3).

In applying the PAD-3.3 code to the centerline melting analysis, the melting temperature of the uranium dioxide is assumed to be 5081°F unirradiated and is decreased by 58F° per 10,000 MWd/t. This relation has been almost universally adopted by the industry and has been accepted by the staff in the past. The expressions for thermal conductivity and gap conductance, described in FSAR Section 4.4.2.11, are unchanged from that originally described in the PAD code. The staff considers it unnecessary to further review these models.

The peak linear heat rating resulting from overpower transients/operator errors (assuming a maximum overpower of 118%) for Millstone Unit 3 is 18.0 kW/ft. As noted in FSAR Section 4.4.2.11.6, the centerline temperature at this peak linear heat rating is below that required to produce fuel melting.

Consequently, the staff concludes that the criterion for the prevention of fuel centerline melting is satisfied.

(5) Pellet/Cladding Interaction

The only two PCI criteria in current use in licensing (1% cladding strain and no fuel melting), although not broadly applicable, are easily satisfied. As noted in the discussion of the cladding stress and strain evaluation, Westinghouse uses an approved code (PAD) to calculate creep strain, and the values calculated by that code are found to be below the 1% strain criterion. And, as indicated in the discussion on overheating failures, the no-centerline-melt criterion is satisfied on the basis of an analysis (described in Section 15.4.6) of the boron dilution event, which is analyzed with an approved code. Therefore, the two existing licensing criteria for PCI have been satisfied.

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

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

(6) Cladding Rupture

In the LOCA analysis, an empirical model (FSAR Section 15.6.5.3) is used to predict the occurrence of cladding rupture. The rupture model used for the large-break analysis is the December 1981 version of the LOCA evaluation model, which includes the modifications delineated in the ECCS evaluation model (WCAP-9220-P-A and WCAP-9221-P-A) that has been approved by the staff and, therefore, is acceptable. The rupture model used for the small-break analysis was the October 1975 version of the ECCS evaluation model (see FSAR Section 15.6.5.3(1,b)). This version has been found acceptable (NUREG-0390 and letter from applicant dated April 13, 1984) for the small-break analysis.

The overall impact of cladding rupture on the response of the SFA design to the LOCA is evaluated in Section 15.6.5 and is not reviewed further in this section.

4.2.3.3 Fuel Coolability Evaluation

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

(1) Fragmentation of Embrittled Cladding

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

One of the most significant analytical methods that is used to provide input to the analysis in Section 15.6.5 is the steady-state fuel performance code, which is reviewed in Section 4.2. This code provides fuel pellet temperatures (stored energy) and fuel rod gas inventories for the ECCS evaluation model as prescribed by Appendix K to 10 CFR 50. The code accounts for fuel thermal conductivity, fuel densification, gap conductance, fuel swelling, cladding creep, and other phenomena that affect the initial stored energy.

Westinghouse uses a relatively new fuel performance code called PAD-3.3 (WCAP-8720). This new Westinghouse code was approved with four restrictions as described in the staff's safety evaluation (Stolz, February 9, 1979). Three of those restrictions dealt with numerical limits and have been met. The fourth restriction related to the use of the PAD-3.3 code for the analysis of fission gas release from uranium dioxide for power-increasing conditions during normal operation. This restriction applied to the SFA. However, Westinghouse prepared and submitted a detailed analysis (Anderson, October 22, 1979) of this restriction in an addendum to WCAP-8720. The staff reviewed and issued (Rubenstein, June 30, 1982) a safety evaluation of the addendum. In that evaluation, the staff concluded that the fourth restriction on the use of the PAD-3.3 code is unnecessary. As a result, the analysis described for the SFA is acceptable as docketed for all cycles of operation.

For the first-cycle operation at full power, the restriction for the PAD-3.3 is not significant and the analyses presented in the FSAR are acceptable. The staff anticipates completion of its review of the Westinghouse evaluation before the attainment of extended burnup at Millstone Unit 3.

For non-LOCA events, the locked-rotor accident (one-pump seizure with four loops operating) is the most severe undercooling event that is analyzed. This event is analyzed in FSAR Section 15.3.3, where it is found that the peak cladding temperature is 1762°F, which is well below the 2700°F design limit. The analysis of this event is reviewed in Section 15.3.3 of this report, but it is clear that the SFA meets the non-LOCA peak cladding temperature design limit.

(2) Violent Expulsion of Fuel Material

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

(3) Cladding Ballooning and Flow Blockage

The Millstone Unit 3 cladding ballooning and flow blockage analysis for the large-break LOCA was performed with correlations approved by the staff (Miller, December 1, 1981) as integral parts of the 1981 ECCS evaluation model (WCAP-9220-P-A and WCAP-9221-P-A). The staff, therefore, finds this analysis for the large-break LOCA acceptable.

The cladding ballooning and flow blockage analysis for the small-break LOCA was performed with correlations from the October 1975 ECCS evaluation model (see FSAR Section 15.6.5.3(1,b)). The staff has found this version acceptable (NUREG-0390 and Counsil, April 13, 1984) for the small-break analysis.

The overall impact of cladding ballooning and assembly flow blockage models on the response of the SFA design to the LOCA is evaluated in Section 15.6.5 and is not reviewed further in this section.

(4) Structural Damage From External Forces

In a response to a staff question on combined seismic and LOCA loading on fuel assemblies, the applicant in a letter dated May 3, 1984, referenced the approved report WCAP-9401-P-A to comply with the requirements of SRP Section 4.2, Appendix A. A plant-specific seismic response spectrum shows that the result is within the approved bounding analysis described in WCAP-9401-P-A except at a small band of second-mode frequencies. The applicant stated that the maximum seismic impact on fuel assemblies is typically dominated by the forcing function at the fundamental mode. Although this may be true in general, the applicant has not demonstrated that the extent of the exception in the second-mode frequencies has any effect on the final analysis in which seismic and LOCA loads (including an asymmetric blowdown load) are combined.

Inasmuch as Westinghouse reactors of similar design have shown acceptable results for combined seismic and LOCA loads, and the Millstone Unit 3 deviation from the bounding Westinghouse seismic response curve appears to be a secondary effect, the staff considers this to be a confirmatory item. The staff will complete its review of this issue pending the applicant's further analysis, including the effect of the asymmetric blowdown load.

4.2.4 Testing, Inspection, and Surveillance Plans

4.2.4.1 Testing and Inspection of New Fuel

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

The Westinghouse quality control program that will be applied to Millstone Unit 3 fuel is discussed in FSAR Section 4.2.4 and addresses fuel system components and parts, pellets, rod inspection, assemblies, process control, and so forth. Inspection of fuel system components depends on the component parts and includes dimensions, visual appearance, audits of test reports, material certification, and nondestructive examinations. Inspections of pellets, for example, are performed for dimensional characteristics such as diameter, density, length, and squareness of ends. Inspections of fuel rods, control rods, burnable poison, and source rods reportedly consist of nondestructive examination techniques such as leak testing, weld inspection, and dimensional measurements. Process control procedures are described in detail. In addition, the applicant states in FSAR Section 4.2.4.4 that if any tests and inspections are to be performed by others on behalf of Westinghouse, Westinghouse will review and approve the quality control procedures, inspection plans, and so forth, to ensure that they are equivalent to the description provided in Sections 4.2.4.1 through 4.2.4.3 and are performed properly to meet all Westinghouse requirements.

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

4.2.4.2 On-Line Fuel Failure Monitoring

To meet the on-line fuel system monitoring requirement, the applicant has indicated in a letter dated April 13, 1984, that Millstone Unit 3 is to have an on-line failed fuel radiation monitor (see Section 11.5.2.3.7 and Figure 9.3-8, Sheet 1 of 4, in the FSAR) in the letdown portion of the chemical volume and control system. This radiation monitor has the capability to measure radiation levels in the reactor coolant that would be caused by failed fuel. This monitor is indicated and alarmed in the control room. Should this monitor alarm, possibly indicating failed fuel, plant personnel will take a reactor coolant sample and perform a radiochemical analysis.

The staff concludes that the applicant has satisfied the guidelines described in Paragraph II.D.2 of the SRP.

4.2.4.3 Postirradiation Surveillance

Westinghouse has extensive experience with the use of 17 x 17 standard fuel assemblies in other operating plants. This experience is summarized in WCAP-8183, which is periodically updated to provide the most recent information on operating plants.

Surveillance of fuel and reactor performance will be routinely conducted at Millstone Unit 3 (letter from applicant dated May 11, 1984). Methods will be used during operation to detect the occurrence of fuel rod failures as dis cussed in Section 4.2.4.2.

As a minimum, a binocular visual examination of a sample number of fuel elements will be conducted during each refueling (letter from applicant dated May 11, 1984). Additional fuel inspections may be conducted depending on the results of operational monitoring and the visual examinations. These inspections may be performed by one or more means available at the time through commercial contractors as judged necessary by plant management (letter from applicant dated June 15, 1984). Northeast Nuclear Energy Company has experience with sipping, ultrasonic examination, and high magnification photography. To the extent practicable, leaking fuel assemblies/rods will be excluded from the operating cores (letter from applicant dated June 15, 1984).

The staff concludes that the applicant has satisfied the guidelines described in Paragraph II.D.3 of the SRP.

4.2.5 Evaluation Findings

The staff concludes that the Millstone Unit 3 fuel has been designed so that (1) the fuel system will not be damaged as a result of normal operation and anticipated operational occurrences, (2) fuel damage during postulated accidents would not be severe enough to prevent control rod insertion when it is required, and (3) core coolability will always be maintained, even after severe postulated accidents, and thereby meets the related requirements of 10 CFR 50.46; 10 CFR 50, Appendix A; GDC 10, 27, and 35; 10 CFR 50, Appendix K; and 10 CFR 100. This conclusion is based on the following:

(1) The applicant has provided sufficient evidence that these design objectives will be met based on operating experience, prototype testing, and analytical predictions. Those analytical predictions dealing with structural response, control rod ejection, and fuel densification have been performed in accordance with (a) the guidelines of RG 1.77, and methods that the staff has reviewed and found to be acceptable alternatives to RGs 1.60 and 1.126, and (2) the guidelines for "Evaluation of Fuel Assembly Structural Response to Externally Applied Forces" in Appendix A to SRP Section 4.2.

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

The staff concludes that the applicant has described methods of adequately predicting fuel rod failures during postulated accidents so that radioactivity releases are not underestimated and thereby meets the related requirements of 10 CFR 100. In meeting these requirements, the applicant has (1) used the fission product release assumptions of RGs 1.4, 1.25, and 1.77 and (2) performed the analysis for fuel rod failures for the rod ejection accident in accordance with the guidelines of RG 1.77.

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

4.3 <u>Nuclear Design</u>

The Millstone Unit 3 power plant has a reactor core consisting of 193 fuel assemblies; each assembly contains a 17 x 17 array of 264 fuel rods of Westinghouse design. The core has a design heat output of 3,411 MWt and is similar to the W. B. McGuire reactor and other recent Westinghouse four-loop reactors. The staff has reviewed the nuclear design of the Millstone Unit 3 reactor in accordance with the guidelines provided by SRP Section 4.3 and on the basis of information contained in the FSAR, amendments thereto, and the referenced topical reports.

4.3.1 Design Bases

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

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

4.3.2 Design Description

The FSAR contains the description of the first-cycle fuel loading, which consists of three different enrichments and has a first-cycle length of approximately 1 1/2 to 2 years. The enrichment distribution, burnable poison distribution, soluble poison concentration, and higher isotope (actinide) content as a function of core exposure are presented. Values given for the delayed neutron fraction and prompt neutron lifetime at the beginning and end of cycle are consistent with those normally used and are acceptable.

Power Distribution

The design bases affecting power distribution are:

(1) The generic design peaking factor for the reactor is 2.32 during normal operation of full power to meet the initial conditions assumed in the LOCA analysis.

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

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

The 2.32 F_Q peaking factor is determined and maintained by means of calculations of extremes of allowed transient power distributions and periodically measured radial power distributions and radial peaking factors F_{xy} and F_H. These also provide maximum initial conditions for events described in Section 15 that ensure that peak full power does not cause centerline fuel melting or result in departure from nucleate boiling during anticipated operational occurrences.

The applicant has described the manner in which the core will be operated and power distribution monitored so as to ensure that these limits are met. The core will be operated in the CAOC mode, which has been shown to result in peaking factors less than 2.32 for both constant power and load following operation. The applicant has elected to use an improved load-follow package, developed by Westinghouse, in Millstone Unit 3. CAOC is described in WCAP-8385 (proprietary) and WCAP-8403 (nonproprietary). This report contains methodology for operation with and without part-length control rods. The former mode allows better return to power capability than the latter. Use of part-length rods has been withdrawn from Westinghouse reactors. The improved load-follow strategy provides a return to power capability during operation without part-length rods comparable to the level previously obtainable from operation with part-length rods.

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

The analysis used to calculate the maximum peaking factor that can occur using the improved strategy expands the set in the Westinghouse constant axial control mode (CAOC) topical report (WCAP-7811) to 18 calculational cases. However, with the redesigned control bank, maneuvers resulting in greater control rod insertion for a longer duration become operationally practical but tend to become slightly more limiting in terms of total peaking factors. Therefore, simulated load-follow maneuvers that return ΔI to the target value (and thereby reduce control rod insertion) have been replaced by load-follow strategies that maintain the deeper rod insertion. As a result of its evaluation, the staff agrees with the Westinghouse conclusion that substitution of these more conservative cases will maintain the limiting nature of the 18-case load-following analysis.

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

The staff concludes, for the reasons stated above, that the improved load-follow package will continue to prevent the 2.32 peaking factor limit from being exceeded in normal operation of the power plant, and is, therefore, acceptable.

Two types of instrumentation systems are normally provided to monitor core power distribution. Excore detectors with two axial sections are used to monitor core power, axial offset, and azimuthal tilt for the 2.32 F_Q limit; movable incore detectors permit detailed power distributions to be measured. These systems are used in operating reactors supplied by Westinghouse, and the staff finds their use acceptable for Millstone Unit 3 when a 2.32 limit is the minimum requirement (or possibly lower when cycle-specific 18-case analyses so indicate).

Reactivity Coefficients

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

Control

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

(1) moderator density and temperature changes from ambient to operating temperatures

(2) equilibrium xenon and samarium buildup

(3) fuel depletion and fission product buildup (that portion not controlled by lumped burnable poison)

(4) transient xenon resulting from load following

Control rods are used to control reactivity change as a result of

(1) moderator reactivity changes from hot zero to full power
(2) fuel temperature changes (Doppler reactivity changes)

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

The applicant has provided data to show that adequate control exists to satisfy the above requirements with enough additional control rod worth to provide a hot shutdown effective multiplication factor less than the design-basis value of 0.984 during initial and equilibrium fuel cycles with the most reactive control rod stuck out of the core. In addition, the chemical and volume control system will be capable of shutting down the reactor by adding soluble boron and maintaining it shut down in the cold, xenon-free condition at any time in core life. These two systems satisfy the requirements of GDC 26.

Comparisons have been made between calculated and measured control rod bank worth in operating reactors and in critical experiments. These comparisons lead to the conclusion that bank worths may be calculated to within approximately 10%. In addition bank worth measurements are performed as part of the startup test program to ensure that conservative values have been used in safety analyses.

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

Control Rod Patterns and Reactivity Worths

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

(1) there is sufficient negative reactivity available to permit rapid shutdown of the reactor with adequate margin

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

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

Stability

The stability of the Millstone Unit 3 core to xenon-induced spatial oscillations is discussed in the FSAR. The overall negative reactivity (power) coefficient provides assurance that the reactor will be stable against total power oscillation. The applicant also concluded that sustained radial or azimuthal xenon oscillations are not possible. This conclusion is based on measurements on an operating reactor of the same dimensions that showed stability against these oscillations. The staff concurs with this conclusion.

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

Criticality of Fuel Assemblies

Criticality of fuel assemblies outside the reactor is precluded by adequate design of fuel transfer and storage facilities. The applicant presents information on calculational techniques and assumptions used to ensure that criticality is avoided. The staff has reviewed this information and the criteria that will be employed and finds them acceptable.

Vessel Irradiation

Values are presented for the neutron flux in various energy ranges at mid-height of the pressure vessel inner boundary. Core flux shapes calculated by standard design methods are input to a transport theory calculation (Sn) which results in a neutron flux of 2.1×10^{10} neutrons per cm^2 per second having energy greater than 10^6 electron-volts at the inner vessel boundary. This results in a fluence of 2.9×10^{19} neutrons per cm^2 for a 40-year vessel life with an 80% use factor. The methods used for these calculations are state of the art, and the staff concludes that acceptable analytical procedures have been used to calculate the vessel fluence. The requirements for surveillance programs and the pressure-temperature limits for operation are presented in Section 5.3.2 of this report.

4.3.3 Analytical Methods

The applicant has described the computer programs and calculational techniques used to obtain the nuclear characteristics of the reactor design. The calculations consist of three distinct types, which are performed in sequence: determination of effective fuel temperatures, generation of macroscopic few-group parameters, and space-dependent few-group diffusion calculations. The programs used (e.g., LASER, TWINKLE, LEOPARD, TURTLE, and PANDA) have been applied as part of the applications for most earlier Westinghouse-designed nuclear plant facilities and the predicted results have been compared with measured characteristics obtained during many startup tests for first-cycle and reload cores. These results have validated the ability of these methods to predict experimental results. The staff, therefore, concludes that these methods are acceptable for use in calculating the nuclear characteristics of Millstone Unit 3.

4.3.4 Summary of Evaluation Findings

The Millstone Unit 3 nuclear design was reviewed according to SRP Section 4.3 (NUREG-0800). All areas of review and review procedures from that section have been followed either for this reactor or for previous similar reactors (e.g., McGuire) or for topical report reviews.

The applicant has described the computer programs and calculational techniques used to predict the nuclear characteristics of the reactor design and has provided examples to demonstrate the ability of the analyses to predict reactivity and physics characteristics of the Millstone Unit 3 plant.

To allow for changes of reactivity as a result of reactor heatup, changes in operating conditions, fuel burnup, and fission product buildup, a significant amount of excess reactivity is designed into the core. The applicant has provided substantial information relating to core reactivity balances for the first cycle and has shown that means have been incorporated into the design to control excess reactivity at all times. The applicant has shown that sufficient control rod worth is available to make the reactor subcritical with an effective multiplication factor no greater than 0.984 in the hot condition at any time during the cycle with the most reactive control rod stuck in the fully withdrawn position. On the basis of its review, the staff concludes that the applicant's assessment of reactivity control requirements over the first core cycle is suitably conservative, and that adequate negative worth has been provided by the control system to ensure shutdown capability. Reactivity control requirements will be reviewed for additional cycles as this information becomes available. The staff also concludes that nuclear design bases, features, and limits have been established in conformance with the requirements of GDC 10, 11, 12, 13, 20, 25, 26, 27, and 28.

This conclusion is based on the following:

(1) The applicant has met the requirements of GDC 11 with respect to prompt inherent nuclear feedback characteristics in the power operating range by calculating a negative Doppler coefficient of reactivity, and using calculational methods that have been found acceptable. The staff has reviewed the Doppler reactivity coefficients in this case and found them to be suitably conservative.

(2) The applicant has met the requirements of GDC 12 with respect to power oscillations that could result in conditions exceeding specified acceptable fuel design limits by showing that such power oscillations are not possible and/or can be easily detected and thereby remedied and by using calculational methods that have been found acceptable.

(3) The applicant has met the requirements of GDC 13 with respect to provisions of instrumentation and controls to monitor variables and systems that can affect the fission process by providing instrumentation and systems to monitor the core power distribution, control rod positions and patterns, and other process variables such as temperature and pressure, and by providing suitable alarms and/or control room indications for these monitored variables.

(4) The applicant has met the requirements of GDC 26 with respect to provision of two independent reactivity control systems of different designs by (a) having a system that can reliably control anticipated operational occurrences, (b) having a system that can hold the core subcritical under cold conditions, and (c) having a system that can control planned, normal power changes.

(5) The applicant has met the requirements of GDC 27 with respect to reactivity control systems that have a combined capability in conjunction with poison addition by the emergency core cooling system of reliably controlling reactivity changes under postulated accident conditions by providing a movable control rod system and a liquid poison system, and performing calculations

to demonstrate that the core has sufficient shutdown margin with the highest-worth stuck rod.

(6) The applicant has met the requirements of GDC 28 with respect to postulated reactivity accidents by meeting the regulatory position in RG 1.77, meeting the criteria on the capability to cool the core, and using calculational methods that have been found acceptable for reactivity insertion accidents reviewed under Section 15.4.8.

(7) The applicant has met the requirements of GDC 10, 20, and 25 with respect to specified acceptable fuel design limits by providing analyses demonstrating that normal operation. including the effects of anticipated operational occurrences, have met fuel design criteria; that the automatic initiation of the reactivity control system ensures that fuel design criteria are not exceeded as a result of anticipated operational occurrences and ensures the automatic operation of systems and components important to safety under accident conditions; and that no single malfunction of the reactivity control system causes violation of the fuel design limits.

4.4 <u>Thermal-Hydraulic Design</u>

4.4.1 Performance and Safety Criteria

The performance and safety criteria for Millstone Unit 3 are stated in FSAR Section 4.4.1. They are:

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

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

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

4.4.2 Design Bases

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

4.4.2.1 Departure From Nucleate Boiling

The margin to departure from nucleate boiling (DNB) at any point in the core is expressed in terms of the departure from nucleate boiling ratio (DNBR). The

DNBR is defined as the ratio of the heat flux required to produce DNB at the calculated local coolant conditions to the actual heat flux.

The thermal-hydraulic design basis, as stated in FSAR Section 4.4.1.1 for the prevention of DNB, is as follows:

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

4.4.2.2 Fuel Temperature

The fuel temperature design basis is given in FSAR Section 4.4.1.2, which states

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

This design basis is evaluated in Section 4.2 of this report.

4.4.2.3 Core Flow

The core flow design basis is given in FSAR Section 4.4.1.3, which states

> A minimum of 94.0 percent of the thermal flow rate will pass through the fuel rod region and be effective for fuel rod cooling.

4.4.2.4 Hydrodynamic Stability

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

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

4.4.3 Thermal-Hydraulic Design Methodology

4.4.3.1 Departure From Nucleate Boiling

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

The W-3 correlation was developed from data obtained from experiments conducted with fluid flowing inside single heated tubes. As test procedures progressed to the use of rod bundles instead of tubes, the correlation was modified to include the effects of "R" and "L" mixing vane grids, as well as the 0.374-in. outside diameter, and axially nonuniform power distributions.

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

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

A description of the THINC-IV computer code is given in WCAP-7956, "THINC-IV: An Improved Program For Thermal-Hydraulic Analysis of Rod Bundle Cores." The design application of the THINC-IV program is given in detail in WCAP-8054, "Application of the THINC-IV Program to PWR Design." Both WCAP-7956 and WCAP-8054 have been reviewed and approved by the staff.

The staff has previously reviewed under a different docket, a November 2, 1977, letter from C. Eicheldinger (Westinghouse) to J. Stolz (NRC) which described THINC-IV analyses using a cosine upper plenum radial pressure gradient with a maximum value of 5 psi at the core center and 0 psi at the periphery. The results of these analyses showed that the effects of a core pressure distribution on the minimum DNBR are negligible. The staff conducted a similar sensitivity study using COBRA-IV. The results also showed that the effects are small (NUREG-0847). On the basis of these analyses, the staff concludes that the use of a nonuniform exit pressure gradient in the Millstone Unit 3 thermal-hydraulic design is acceptable.

On the basis of its findings that the CHF correlation and the thermal-hydraulic computer code used by the applicant have been previously approved by the staff and that the use of a uniform core exit pressure gradient has been adequately justified, the staff concludes that the DNB design methodology used in the design of the Millstone Unit 3 is acceptable.

4.4.3.2 Core Flow

The core flow design basis requires that the minimum flow, which will pass through the fuel rod region and be effective for fuel rod cooling, is 94.0% of the primary coolant flow rate. The remainder of the flow, called bypass flow, will be ineffective for cooling since it will take the following bypass paths:

(1) flow through the spray nozzles into the upper head for head cooling purposes

(2) flow entering into the rod cluster control rod guide thimbles to cool the control rods

(3) leakage flow from the vessel inlet nozzle directly to the vessel outlet nozzle through the gap between the vessel and the barrel

(4) flow between the baffle and barrel

(5) flow in the gaps between the fuel assemblies on the core periphery and the adjacent baffle wall

The amount of bypass flow is determined by a series of hydraulic resistance calculations on the core and vessel internals and verified by model flow tests. Since the amount of bypass flow is consistent with approved plants of similar design, the staff concludes that the core bypass flow used in the design analysis, 6.0%, is acceptable.

4.4.3.3 Hydrodynamic Instability

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

The applicant stated that the core was stable because Westinghouse reactors will not experience any Ledinegg instability over Condition I and II operational ranges and that open channel configurations, which are a feature of Westinghouse PWRs, are more stable than closed-channel configurations. This was shown by flow stability tests that were conducted at pressures up to 2,200 psia. The results showed that for flow and power levels typical of reactor conditions, no flow oscillations could be induced above 1,200 psia.

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

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

In the interim, the staff concludes that past operating experience, flow stability experience, and the inherent thermal-hydraulic design of Westinghouse PWRs serve as a basis for issuance of an operating license.

4.4.4 Operating Abnormalities

4.4.4.1 Fuel Rod Bowing

A significant parameter that affects the thermal hydraulic design of the core is rod-to-rod bowing within fuel assemblies. The Westinghouse methods for predicting the effects of rod bow on DNB (WCAP-8691, Revision 1, "Fuel Rod Bow Evaluation,") have been approved by the staff.

The FSAR stated that there is a 9.1% margin to accommodate full- and low-flow DNBR penalties resulting from fuel rod bowing. The applicant should verify that (1) the breakdown of this margin into individual factors is consistent with WCAP-8691 and (2) this margin (in whole or part) was not used in any other analysis.

Also, the applicant should insert into the Bases of the Technical Specification any of the generic or plant-specific margins that may be used to offset the reduction in DNBR resulting from rod bowing.

4.4.4.2 Crud Deposition

Operating experience on two PWRs indicates that a significant reduction in the core flow rate can occur over a relatively short period of time as a result of crud deposition on the fuel rods. In establishing the Technical Specifications for Millstone Unit 3, the staff will require provisions to ensure that the minimum design flow rates are achieved. The applicant has provided the Westinghouse generic description of flow measurement methods and associated uncertainties. However, the applicant has not verified that the uncertainties are applicable to Millstone Unit 3. In addition, the applicant has not verified that venturi fouling will be adequately accounted for in determining the core flow rate. These issues must be acceptably addressed before the staff can approve the applicant's capability to measure core flow.

4.4.5 Loose Parts Monitoring System

The applicant has provided a description of the loose parts monitoring system (LPMS) that will be used at Millstone Unit 3. The design will consist of eight active instrumentation channels, each comprising a piezoelectric accelerometer (sensor) and signal conditioning equipment. Sensors are fastened mechanically to the reactor coolant system (RCS) at each of the following potential loose parts collection regions:

(1) reactor pressure vessel - upper head region
(2) reactor pressure vessel - lower head region
(3) each steam generator - reactor coolant inlet region

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

In response to staff Question 492.5, the applicant committed to supply a report describing operation of the system hardware and implementation of the loose parts detection program. The applicant also stated that the Millstone Unit 3 LPMS conforms to RG 1.133 with the following exceptions:

(1) Only one sensor is located on each steam generator. The staff will require that the applicant provide two sensors on each steam generator.

(2) The LPMS is not qualified to an operating basis earthquake (OBE). The staff does not require OBE qualification; however, it requires that the system should be able to perform following all seismic events that do not require plant shutdown up to and including the OBE.

(3) RG 1.133 requires that a calibration test be performed at least once every 18 months. The applicant modified that requirement to at least once per fuel cycle or every 18 months, whichever is greater. The staff finds this acceptable.

The staff will report resolution of these issues in a supplement to this SER.

4.4.6 Thermal-Hydraulic Comparison

The thermal-hydraulic design parameters for Millstone Unit 3 are listed in Table 4.1 and compared with values for the Watts Bar Units 1 and 2, Trojan, and Standardized Nuclear Unit Power Plant System (SNUPPS) plants.

Millstone Unit 3 was designed to operate at the same thermal power as the Watts Bar Units 1 and 2, Sequoyah, Trojan, and SNUPPS plants. The W-3 critical heat flux correlation and THINC-IV computer program were used in the design of all of the plants. The Watts Bar Units 1 and 2, Sequoyah, Trojan, and SNUPPS plants have been previously reviewed and approved by the staff.

The comparability of Millstone Unit 3 with the Watts Bar Units 1 and 2, Sequoyah, Trojan, and SNUPPS plants supports the conclusion that the Millstone Unit 3 thermal-hydraulic design is acceptable.

4.4.7 N-1 Loop Operation

In a letter dated April 9, 1984, the applicant expressed the intent to operate in the N-1 mode. However, the applicant has not yet provided core thermal-hydraulic analyses taking into account the effect of partial loop operation on core inlet flow distribution and minimum DNBR or Technical Specifications including the appropriate provisions to ensure that this type of operation is within acceptable limits.

Resolution of this issue will be included in a supplement to this SER.

4.4.8 Inadequate Core Cooling Instrumentation

The staff has reviewed the applicant's submittal (FSAR Section 4.4.6.5) and has found that the applicant's description of his proposed inadequate core cooling (ICC) instrumentation is incomplete with respect to the documentation required by Item II.F.2 of NUREG-0737.

On November 4, 1982, the Commission determined that an instrumentation system for detection of ICC consisting of upgraded subcooling margin monitors, core exit thermocouples, and a reactor coolant inventory tracking system is required for the operation of PWR facilities. The staff has also completed the review of several proposed generic reactor level or inventory tracking systems for the detection of ICC in PWRs and has found that the Combustion Engineering heated junction thermocouple (HJTC) system and the Westinghouse reactor vessel level instrumentation system (RVLIS) are acceptable for tracking RCS inventory. The details of the staff review of this generic system is reported in NUREG/CR-2627 and NUREG/CR-2628 for the Combustion Engineering and Westinghouse systems, respectively. Therefore, the staff will require the applicant to provide the itemized documentation of a complete ICC system including the subcooled margin monitor (SMM), the core exit thermocouple (CET), and the reactor inventory tracking system (RITS) on a schedule that will permit completion of the staff's review before fuel loading. The staff will report its findings in a future supplement to this SER. This is an open item.

4.4.9 Conclusion

The thermal-hydraulic design of Millstone Unit 3 was reviewed by the staff according to SRP Section 4.4, Section II, "Thermal and Hydraulic Design Acceptance Criteria" (NUREG-0800). The scope of the review included the design criteria, core design, and the steady-state analysis of the core thermal-hydraulic performance. The review concentrated on the differences between the proposed core design and those designs that have been previously reviewed and found acceptable by the staff. It was found that all such differences were acceptable. The applicant's thermal-hydraulic design analyses were performed using analytical methods and correlations that have been previously reviewed by the staff and found acceptable.

The staff concludes that the initial core has been designed with appropriate margin to ensure that acceptable fuel design limits are not exceeded during steady-state operation and anticipated operational occurrences. The thermal-hydraulic design of the initial core, therefore, meets the requirements of GDC 10 and 10 CFR 50 and is acceptable for preliminary design approval. This conclusion is based on the applicant's analyses of the core thermal-hydraulic performance that were reviewed by the staff and found acceptable. However, before final design approval and before an operating license is issued, the staff will require the applicant to perform the following:

(1) provide a commitment to supply a report describing the loose parts detection program and implementation of the system as described in Section 4.4.5 of this report

(2) supply the II.F.2 information previously enumerated in Section 4.4.8 of this report

(3) provide a description of the flow measurement capability and the procedure used to measure flow as described in Section 4.4.4.2 of this report

(4) address the concerns regarding the N-1 loop operation as discussed in Section 4.4.7 of this report

(5) address the concerns regarding the effect of rod bow on DNBR as described in Section 4.4.4.1 of this report

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

4.5 Reactor Materials

4.5.1 Control Rod Drive Structural Materials

The staff concludes that the control rod drive mechanism structural materials are generally acceptable and meet the requirements of GDC 1, 14, and 26 as well as 10 CFR 50.55a. A confirmatory response from the applicant is required concerning the yield strength of austenitic stainless steels in these components. This conclusion is based on the applicant having demonstrated that the properties of materials selected for the control rod drive mechanism components exposed to the reactor coolant satisfy Appendix I of Section III of the ASME Code and Parts A, B, and C of Section II of the Code. The applicant should confirm conformance with the staff position that the yield strength of cold-worked austenitic stainless steel does not exceed 90,000 psi. This is a confirmatory item. The applicant met the guidelines of RG 1.85 by using materials of construction that are approved for use to ASME Code cases.

In addition, the controls imposed on the austenitic stainless steel of the mechanisms satisfy, to the extent practical, the recommendations of RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal," and RG 1.44, "Control of the Use of Sensitized Stainless Steel." The alternative method of control of ferrite content by testing of the purchased material and the modification of testing procedures to evaluate weldments for stress corrosion cracking have been reviewed by the staff and are acceptable. The applicant has confirmed that the tempering temperatures and aging temperatures of heat treatable materials in the control rod drive mechanism are specified to eliminate the susceptibility to stress corrosion cracking in reactor coolant. The fabrication and heat treatment practices performed provide assurance that stress corrosion cracking will not occur during the design life of the components. The compatibility of all materials used in the control rod system in contact with the reactor coolant satisfies the criteria of Articles NB-2160 and NB-3120 of Section III of the Code. Cleaning and cleanliness controls are in accordance with ANSI Std. N 45.2.1-1973, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants," and RG 1.37, "Quality Assurance Requirements for Cleaning Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants."

4.5.2 Reactor Internals Materials

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

The applicant has met the requirements of GDC 1 and 10 CFR 50.55a with respect to ensuring that the design, fabrication, and testing of the materials used in the reactor internal and core support structure are of high quality standards and adequate for structural integrity. The controls imposed on components constructed of austenitic stainless steel satisfy, to the extent practical, the recommendations of RGs 1.31 and 1.44. Where the recommendations of those regulatory guides were not followed, the alternative approaches taken by the applicant have been reviewed by the staff and are acceptable (see Section 4.5.1).

The materials used for construction of components of the reactor internal and core support structure have been identified by specification and found to be in conformance with the requirements of NG-2000 of Section III and Parts A, B, and C of Section II of the ASME Code. In addition, the applicant has met the guidelines of RG 1.85, "Code Case Acceptability ASME Section III Materials," by using materials in construction that are approved for use by ASME Code cases. As proven by extensive tests and satisfactory performance, the specified materials are compatible with the expected environment and corrosion is expected to be negligible.

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

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

4.6 Functional Design of Reactivity Control Systems

The reactivity control systems were reviewed in accordance with SRP Section 4.6 (NUREG-0800).

The functional designs of the reactivity control systems have been reviewed to confirm that they meet the various reactivity control conditions for all modes of operation. These are

(1) the capability to operate in the unrodded, critical, full-power mode throughout plant life

(2) the capability to vary power level from full power to hot shutdown and ensure control of power distributions within acceptable limits at any power level

(3) the capability to shut down the reactor in a manner sufficient to mitigate the effects of postulated events discussed in Section 15 of this report.

The control rod drive system (CRDS), the safety injection system (SIS), and the chemical and volume control system (CVCS) constitute the reactivity control systems.

The CRDS is composed of control rod drive mechanisms to which the rod cluster control assemblies (RCCAs) are attached. The control rod drive mechanism (CRDM) is a magnetically operated jack. The magnetic jack is an arrangement of three electromagnets that are energized in a controlled sequence to insert or withdraw RCCAs in discrete steps. The RCCAs are divided into two categories: control and shutdown.

The control category of RCCAs may be automatically inserted or withdrawn to compensate for changes in reactivity associated with power level changes and power distribution, variations in moderator temperature, or changes in boron concentration. The shutdown category of RCCAs is fully withdrawn during power operations and is used solely to insert large amounts of negative reactivity to shut down the reactor. (See Section 4.3 of this SER for further discussion on these features.)

The RCCAs are the primary shutdown mechanisms for normal operation, accidents, and transients. They insert automatically upon a reactor trip signal. Concentrated boric acid solution is injected by the SIS in the event of a LOCA, steamline break, or loss of normal feedwater flow, thereby complying with the requirements of GDC 29.

Failure of electrical power to an RCCA will result in the insertion of that assembly, as will shearing of the connection between the RCCA and CRDM. Single failure of an RCCA is considered in transient and accident analyses that include the most reactive RCCA stuck outside the core. Analysis of accidental withdrawal of an RCCA is found to have acceptable results. This conforms to the requirements of GDC 23 and 25.

The SIS is automatically actuated to inject borated water into the reactor coolant system (RCS) when a safety injection actuation signal (SIAS) is received. The SIS pumps take suction from the refueling water storage tank (RWST). The SIS is discussed further in Section 6.3 of this report.

The CVCS is designed to accommodate slow or long-term reactivity changes such as those caused by fuel burnup or by variation in the xenon concentration resulting from changes in reactor power level. The CVCS is used to control reactivity by adjusting the dissolved boron concentration in the RCS. The boron concentration is controlled to (1) allow optimum RCCA positioning, (2) compensate for reactivity changes associated with variations in coolant temperature, core burnup, and xenon concentration, and (3) provide shutdown margin for maintenance and refueling operations or emergencies. A portion of the CVCS (the charging pumps, the boric acid transfer pump discharge, and the boric acid tanks) injects a concentrated boron solution into the RCS to help ensure plant shutdown in the event of an SIAS. The boric acid concentration in the RCS is controlled by the charging and letdown portions of the CVCS.

The CVCS can maintain the reactivity of the reactor within required bounds by means of the automatic makeup system to replace minor leakage without significantly changing the boron concentration in the RCS. Dilution of the RCS boron concentration, required for the reactivity losses occurring as a result of fuel depletion, may be accomplished by manual action. The CVCS is discussed further in Section 9.3.4 of this SER.

The concentration of boron in the RCS is changed under the following conditions:

(1) startup - boron concentration decreased to compensate for moderator temperature and power increase

(2) load follow - boron concentration increased or decreased to compensate for xenon transients following load changes

(3) fuel burnup - boron concentration decreased to compensate for burnup

(4) cold shutdown - boron concentration increased to compensate for increased moderator density as a result of cooldown

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

The reactivity control systems, including the addition of concentrated boric acid solution by the SIS, are capable of controlling all anticipated operational changes, transients, and accidents, except possibly the small-break LOCAs. (For further information on the performance of the charging and borating portions of the CVCS with respect to small-break LOCAs, refer to Sections 6.3 and 15.3 of this SER.) All acccidents are analyzed with the assumption that the most reactive RCCA is stuck out and cannot be inserted, which complies with the requirements of GDC 27.

Compliance with the requirements of GDC 28 is discussed in Sections 4.3 and 15.0 of this SER.

On the basis of this review, the staff concludes that the reactivity control system functional design meets the requirements of GDC 23, 25, 26, 27, 28, and 29 with respect to its fail-safe design, malfunction protection design, redundancy and capability, combined systems capability, reactivity limits, and protection against anticipated operational occurrences and is, therefore, acceptable. The CRDS meets the acceptance criteria of SRP Section 4.6.

Table 4.1 Reactor design comparison

Parameter	Millstone Unit 3	Watts Bar Units 1&2 SER*	Trojan SER**	Sequoyah SER***	SNUPPS SER†
Performance characteristics					
Reactor core heat output, MWt	3,411	3,411	3,411	3,411	3,411
System pressure, psia	2,250	2,250	2,250	2,250	2,250
Minimum DNBR					
Typical cell	2.10	2.05	2.04	2.22	2.09
Thimble cell	1.74	1.72	1.71	1.81	1.73
Minimum DNBR	1.30	1.30	1.30	1.30	1.30
Critical heat flux correlation	W-3	W-3	W-3	W-3	W-3
Coolant flow					
Total flow rate, 10^6 lb/hr	140.8	144.8	132.7	133.8	142.1
Effective flow rate for heat transfer, 10^6 lb/hr	132.4	133.9	126.7	127.8	133.9
Average velocity along fuel rods, fps	16.4	16.6	15.7	15.6	16.6
Effective core flow area, ft^2	51.1	51.1	51.1	51.1	51.1
Coolant temperature, °F					
Nominal reactor inlet	557.0	559.1	552.7	545.7	558.8
Average rise in core	60.2	62.7	66.9	67.8	59.4
Pressure drop across core, psi	25.8±3.9	25.8±3.9 25.8±2.6		24.3±2.4	
Heat transfer, 100% power					
Active heat transfer surface area, ft^2	59,700	59,700	59,700	59,700	59,700
Average heat flux, Btu/hr-ft^2	189,800	189,800	189,800	189,800	189,800
Maximum heat flux, Btu/hr-ft^2	440,300	440,300	574,500	474,500	440,300
Average linear heat rate, kW/ft	5.44	5.44	5.44	5.44	5.44
Maximum thermal output, kW/ft	12.6	12.6	13.6	12.2	12.6

*NUREG-0847.
**Dated October 1974.
***NUREG-0011.
†NUREG-0330.

5 REACTOR COOLANT SYSTEM AND CONNECTED SYSTEMS

5.1 Summary Description

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

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

The RCS pressure boundary provides a second barrier against the release of radioactivity generated within the reactor and is designed to ensure a high degree of integrity throughout the life of the plant.

The RCS pressure changes during normal operation are controlled by the use of the pressurizer where water and steam are maintained in thermodynamic equilibrium under operating conditions by electrical heaters and water spray. Spring-loaded safety valves and power-operated relief valves are mounted on the pressurizer and discharge to the pressurizer relief tank, when necessary, where steam is condensed and cooled by mixing with water.

5.2 Integrity of Reactor Coolant Pressure Boundary

5.2.1 Compliance With Codes and Code Cases

5.2.1.1 Compliance With 10 CFR 50.55a

The pressure-retaining components of the reactor coolant pressure boundary (RCPB) as defined by the rules of 10 CFR 50.55a, "Codes and Standards," have been properly classified in FSAR Table 3.2-1 as American Society of Mechanical Engineers (ASME) Code, Section III, Class 1, components. These components are designated Safety Class 1 (Quality Group A) in conformance with RG 1.26. The Quality Group A components were reviewed in accordance with SRP Section 5.2.1.1, and the results of this review are presented below. The review of other pressure-retaining components, such as those constructed to ASME Code, Section III, Classes 2 and 3, is in Section 3.2.2 of this report.

The ASME Code, Section III, edition and addenda used in the construction of these Quality Group A components are identified in FSAR Table 5.2-1 and are those that were required at the time of procurement of the components or are, where appropriate, to ensure compliance with 10 CFR 50.55a, later editions, or addenda.

In addition to the Quality Group A components of the RCPB, certain lines that perform a safety function and that meet the exclusion requirements of Footnote 2 of 10 CFR 50.55a are classified Quality Group B in accordance with the guidance provided in Position C.1 of RG 1.26 and are constructed as ASME Code, Section III, Class 2, components. Valve leakage monitoring system lines that do not perform a safety function and that meet the exclusion requirements of Footnote 2 of 10 CFR 50.55a are classified Quality Group D on the downstream side of the isolation valves.

The staff concludes that construction of components of the RCPB in conformance with the appropriate ASME Code editions and addenda and the Commission's regulations provides assurance that component quality is commensurate with the importance of the safety function of the RCPB. This constitutes an acceptable basis for satisfying the requirements of GDC 1.

5.2.1.2 Applicable Codes Cases

The applicant has identified specific ASME Code cases* whose requirements have been applied in the construction of pressure-retaining ASME Code, Section III, Class 1, components within the RCPB (Quality Group A). However, staff acceptance is contingent on the applicant supplying a list of ASME Code cases used in the construction of Section III, Class 1, components within the RCPB. The staff has reviewed these Code cases in accordance with SRP Section 5.2.1.2 with the following exception. The revised SRP includes a new requirement for the review of ASME Code cases that are used in the construction of Class 2 and Class 3 components. Because the current revision of 10 CFR 50.55a is applicable only to those ASME Code cases used in the construction of Class 1 components, the staff has limited its review in accordance with the regulation.

The basis for acceptance in the staff's review has been the Code cases found to be acceptable in RG 1.84, "Code Case Acceptability - ASME Section III, Design and Fabrication," and RG 1.85, "Code Case Acceptability - ASME Section III, Materials," and the Code cases previously found to be acceptable by the staff for plants similar to Millstone Unit 3 before publication of RGs 1.84 and 1.85. The staff concludes that compliance with the requirements of these Code cases will result in a component quality level that is commensurate with the importance of the safety function of the RCPB and constitutes an acceptable basis for satisfying the requirements of GDC 1.

5.2.2 Overpressure Protection

Overpressure protection for Millstone Unit 3 has been reviewed in accordance with SRP Section 5.2.2 (NUREG-0800). Conformance with the acceptance criteria, except as noted below, formed the basis for concluding that the design of the facility for overpressure protection is acceptable.

Overpressure protection for the RCPB is provided by means of three safety and two power-operated relief valves, in combination with the reactor protection

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*Staff acceptance is contingent upon the applicant supplying a list of ASME Code cases used in the construction of Section III, Class 1, components within the reactor coolant pressure boundary.

system, and operating procedures. The combination of these features provides overpressure protection as required by GDC 15; ASME Code, Section III; and Appendix G to 10 CFR 50. The above requirements ensure RCPB overpressure protection for both power operation and low temperature operation (startup and shutdown). The following is a discussion of both modes of overpressure protection.

5.2.2.1 Overpressure Protection During Power Operation

For this mode, the pressurizer power-operated relief valves are sized to limit system pressure to a value not exceeding the safety valve setpoint (2,485 psig) to minimize challenges to the safety valves. The pressurizer spray system is designed to maintain the reactor coolant system pressure below the power-operated relief valve setpoint of 2,335 psig during a step reduction in power level of up to 10%. The power-operated relief valves limit the pressurizer pressure to a value below the high-pressure reactor trip setpoint of 2,385 psig for all anticipated transients considered in the design, up to and including the design-basis 50% step load reduction with steam dump to the condensers.

Each of the pressurizer safety valves is spring loaded and has a relieving capacity of 420,000 lb per hour of saturated steam at 2,485 psig. The combined capacity of two of these three safety valves is adequate to prevent the pressurizer pressure from exceeding the ASME Code, Section III, limit of 110% design pressure following the worst reactor coolant system pressure transient, identified to be a 100% load rejection resulting from a turbine trip with concurrent loss of main feedwater. This event was analyzed with no credit taken for operation of reactor coolant system power-operated relief valves, main steamline atmospheric steam dump valves, condenser steam dump system, pressurizer level control system, and pressurizer spray system.

The evaluation is supported by a generic sensitivity study of required safety valve flow rate versus trip parameter presented in WCAP-7769, Revision 1. The study indicates that the safety valves are sized sufficiently to prevent RCS overpressurization, assuming no credit for an reactor trip. After determining the required safety valve capacity, the loss of load transient is again analyzed for the case when main feedwater flow is lost because of the loss of steam flow to the turbine. In the case, credit is taken for Doppler feedback and appropriate reactor trip such as pressurizer high pressure overtemperature ΔT, low main feedwater flow, and low-low steam generator water level - other than direct reactor trip on turbine trip. The above analyses were performed using the LOFTRAN code, which has been reviewed and approved by the staff.

The safety valves are designed in accordance with ASME Code, Section III, and periodic testing and inspection are performed in accordance with Section XI. In Chapter 14 of the FSAR the applicant has described the preoperational test program, which includes testing of the pressure-relieving devices discussed in this SER section, and has indicated that these tests would be conducted in full compliance with the intent of RG 1.68. Additionally, Items II.D.1 and II.D.3 of NUREG-0737 require performance testing of the relief and safety valves, and position indication of the valves. Conformance of these items is addressed in Sections 3.9.3 and 7.5.2 of this SER. The staff concludes that the overpressure protection provided for Millstone at power-operating conditions will comply with the guidelines of SRP 5.2.2 and the requirements of GDC 15.

5.2.2.2 Overpressure Protection During Low Temperature Operation

SRP Section 5.2.2 states that the overpressure protection system during low-temperature operation of the plant should be designed in accordance with the criteria of Branch Technical Position RSB 5-2.

The applicant states that administrative procedures are available to assist the operator in controlling RCS pressure during low-temperature operation. However, to provide a backup to the operator and to minimize the frequency of RCS overpressurization, an automatic system is provided to mitigate any inadvertent pressure excursions.

Protection against overpressurization events is provided through the use of two pressurizer power-operated relief valves (PORVs). The applicant states that during startup and cooldown operation, the RCS is always "water solid" and the mitigation system is required during these low-temperature operations.

Low-temperature protection is primarily provided by the PORV with opening setpoints that automatically adjust as a function of reactor coolant temperature. The PORVs are each supplied with actuation logic to ensure that an automatic and independent RCS pressure control feature is available. The reactor coolant temperature measurements are auctioneered to obtain the lowest value. This temperature is translated into a PORV setpoint curve that is below the maximum allowable system pressure set forth by 10 CFR 50, Appendix G. If the measured reactor coolant pressure approaches the PORV setpoint curve within a certain limit, an alarm is sounded in the control room indicating that a pressure transient is occurring. Once system pressure reaches the PORV opening setpoint, the PORVs open to relief system pressure. The staff has asked the applicant to address failures in the temperature auctioneer circuitry since both PORVs could be rendered inoperable by the failure of a single auctioneering circuit. In Revision 1 to the FSAR the applicant stated that instrumentation is not shared by the redundant control channels. Any single random failure in one train will not prevent the redundant train from performing its safety function. This response is acceptable. The consequences of the failure of the vital dc bus which causes the residual heat removal system (RHRS) to isolate as well as defeat the PORV were not addressed in Millstone's FSAR. In a letter dated May 22, 1984, the applicant indicated that the Millstone design would not have this type of common mode failure. The RHRS would not be isolated in the event of the failure of a vital dc bus. The response is acceptable.

As a backup to the low-temperature overpressure protection (LTOP) system, the RHRS has two relief valves (liquid relief valves) located at the RHR pump suction line with a capacity of 900 gpm each at a setpoint pressure of 450 psig. The applicant has provided test data to support the relief valve capacity. The relieving capacity of each valve is more than adequate to relieve the combined flow of the two centrifugal charging pumps. The relief valves at the RHR pump suction lines provide additional LTOP relieving capacity only when the RHR suction isolation valves are open.

The staff was concerned that a particular scenario had not been adequately assessed. If there were a loss of a vital dc bus, then there would be a loss of letdown, which would cause a pressurization of the RCS, and a loss of one of the two PORVs. If the staff's single-failure criterion is then applied to the remaining PORV, there could be an overpressure transient without any mitigation systems available.

The applicant, in a letter dated August 29, 1983, stated that whenever the RCS is in a condition in which the low-temperature overpressure protection system is required to be operable, all but one charging pump are required to be made inoperable and the operator is instructed to remove power to the safety injection pumps. This requirement ensures that only one charging pump would be operating at the initiation of the event. Also, one RHR loop is required to be in operation and the other RHR loop is required to be operable. This requirement ensures that at least one RHR suction line relief valve is available for overpressure protection.

The applicant indicated in a letter dated August 29, 1984, that the combined flow rate of two charging pumps is below the relieving capacity of a single RHR relief valve. Since the combined flow rate of two charging pumps is greater than the flow rate of the safety injection pump, the staff finds the design acceptable.

The staff reviewed the overpressure protection system for both normal and low-temperature operations and concludes that the system is acceptable and meets the relevant requirements of GDC 15 and 31 and Appendix G to 10 CFR 50. This conclusion is based on the following:

(1) The overpressure protection system prevents overpressurization of the reactor coolant pressure boundary under normal operation and limits the reactor pressure during anticipated operational occurrences.

(2) Overpressurization protection is provided by three safety valves. These valves discharge to the pressurizer quench tank through a common header. The safety and power-operated relief valves in the primary system, in conjunction with the steam generator safety and power-operated relief valves in the secondary system, and the reactor protection system, will protect the primary system against overpressure.

(3) The peak primary system pressure following the worst pressure transient is limited to the ASME Code-allowable value. The Millstone plant was assumed to be operating at design conditions (102% of rated power) and the reactor is shut down by a high pressurizer pressure trip signal. The calculated pressure is less than 110% of the design pressure.

(4) Overpressure protection during low-temperature operation of the plant is provided by two PORVs in conjunction with administrative controls. As a backup to the PORVs, the RHR suction line relief valves provide additional relief capacity.

(5) The applicant has met GDC 15 and 31 and Appendix G because the guidelines of BTP RSB 5-2 have been implemented. In addition, the applicant has incorporated in his design the recommendations of Task Action Plan Items II.G.1, II.D.1, and II.D.3 of NUREG-0737.

5.2.3 Reactor Coolant Pressure Boundary Materials

The staff concludes that the plant design is generally acceptable and meets the requirements of GDC 1, 4, 14, 30, and 31 of Appendix A of 10 CFR 50; the requirements of Appendices B and G of 10 CFR 50; and the requirements of 10 CFR 50.55a. Confirmation by the applicant of the staff position concerning the yield strength of austenitic stainless steels in the reactor coolant pressure boundary is required. This is a confirmatory item. This conclusion is based on the staff's review of the FSAR.

The materials used for construction of components of the RCPB have been identified by specification and found to be in conformance with the requirements of Section III of the ASME Code. Compliance with the above Code provisions for materials specifications satisfies the quality standards requirements of GDC 1 and 30, and 10 CFR 50.55a.

The materials of construction of the RCPB exposed to the reactor coolant have been identified and all of the materials are compatible with the primary coolant water, which is chemically controlled in accordance with appropriate Technical Specifications. This compatibility has been proven by extensive testing and satisfactory performance. This includes satisfying, to the extent practical, the recommendations of RG 1.44. Where the recommendations of the regulatory guide were not followed, the alternative approaches taken have been reviewed by the staff and are acceptable (see Section 4.5.1).

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

The materials of construction for the RCPB are compatible with the thermal insulation used in these areas. The thermal insulation used on the RCPB is either the reflective stainless steel type or is made of nonmetallic compounded materials that meet most of the recommendations of RG 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steels." The use of standard commercial packaging with receipt inspection for damage, as an alternative approach to the special packaging recommendations in the guide, is acceptable to the staff. Conformance with the above recommendations satisfies the requirements of GDC 14 and 31 relative to prevention of failure of the RCPB.

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

The fracture toughness tests required by the ASME Code, augmented by Appendix G, 10 CFR 50, provide reasonable assurance that adequate safety margins against nonductile behavior or rapidly propagating fracture can be established for all pressure-retaining components of the reactor coolant pressure boundary. The use of Appendix G of the ASME Code, Section III, and the results of fracture toughness tests performed in accordance with the Code and NRC regulations in establishing safe operating procedures, provide adequate safety margins during operating, testing, maintenance, and postulated accident conditions.

Compliance with these Code provisions and NRC regulations satisfies the requirements of GDC 31 and 10 CFR 50.55a regarding prevention of fracture of the RCPB.

The applicant has taken alternative approaches to the recommendations of RG 1.50, "Control of Preheat Temperature for Welding Low Alloy Steels." The alternative approaches taken by the applicant are that welding procedures are qualified within the preheat temperature range (minimum limit plus 50F°) rather than at the minimum preheat temperature, and preheat temperatures are maintained for an extended period of time rather than preheat temperatures maintained until the start of post-weld heat treatment. The staff concludes that these alternative approaches will not have a significant effect on the propensity for hydrogen cracking (the concern of RG 1.50) and will not cause other hazards. Accordingly, the staff accepts these alternative approaches. The controls used provide reasonable assurance that cracking of components made from low-alloy steels will not occur during fabrication. If cracking does occur, the required Code inspections should detect such flaws. These controls satisfy the quality standards requirements of GDC 1 and 30 and 10 CFR 50.55a.

RG 1.34, "Control of Electroslag Weld Properties," is not applicable because the electroslag welding process was not used on RCPB components.

The controls imposed on welding ferritic and austenitic steels under conditions of limited accessibility satisfy, to the extent practical, the recommendations of RG 1.71, "Welder Qualification for Areas of Limited Accessibility." The applicant's contractors maintain close supervisory control of the welders and reoccurrence of welding situations in production are adequate to ensure that the most skilled welders are used in areas of limited accessibility. The staff concludes, that as such welds are inspected, qualification of the welders making acceptable welds occurs automatically under the Code. These controls satisfy the quality standards requirements of GDC 1, GDC 50, and 10 CFR 50.55a. The controls imposed on weld cladding of low-alloy steel components by austenitic stainless steel are in accordance with the recommendations of RG 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components." These controls provide assurance that practices that could result in underclad cracking will be restricted. The controls also satisfy the quality standards requirements of GDC 1 and 30 and 10 CFR 50.55a.

The applicant has not addressed the staff position limiting RCPB components constructed of austenitic stainless steel to a maximum yield strength of 90,000 psi.

The controls to avoid stress corrosion cracking in reactor coolant pressure boundary components constructed of austenitic stainless steels satisfy, to the extent practical, the recommendations of RGs 1.44 and 1.37. The alternate approaches taken by the applicant were reviewed by the staff and are acceptable (see Section 4.5.1).

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

The controls imposed during welding of austenitic stainless steels in the RCPB satisfy, to the extent practical, the recommendations of RGs 1.31, 1.34, and

1.71. The alternate approaches taken by the applicant were reviewed by the staff and are acceptable (see Section 4.5.1).

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

5.2.4 Reactor Coolant Pressure Boundary Inservice Inspection and Testing

5.2.4.1 Compliance With the Standard Review Plans

The Millstone Unit 3 review is continuing because the applicant has not completed his preservice inspection (PSI) examinations. The staff review, to date, was conducted in accordance with SRP Section 5.2.4 (NUREG-0800) except as discussed below.

SRP Section 5.2.4, Paragraph II.4, "Acceptance Criteria, Inspection Intervals," has not been reviewed because this area applies only to inservice inspections (ISIs), not to the preservice inspection. This subject will be addressed during review of the ISI program after licensing.

SRP Section 5.2.4, Paragraph II.5, "Acceptance Criteria, Evaluation of Examination Results," has been reviewed and the applicant has incorporated ASME Code, Section XI, Article IWB-3000, "Standards for Examination Evaluation," into his PSI program. However, ongoing NRC generic activities and research projects indicate that the present specified ASME Code procedures may not always be capable of detecting the acceptable size flaws specified in the IWB-3000 standards. For example, ASME Code procedures specified for volumetric examination of reactor vessels, bolts and studs, and piping (in particular, cast austenitic piping) have not proven to be capable of detecting acceptable size flaws in all cases. The applicant may be required to use augmented procedures that exceed the minimum ASME Code requirements for the above examinations in order to ensure adequate detectability of flaws. The staff will continue to evaluate development of improved procedures and will require that these improved procedures be made a part of the inservice examination requirements.

The staff has not reviewed the applicant's repair procedures based on ASME Code, Section XI, Article IWB-4000, "Repair Procedures," because repairs are not generally necessary in the PSI program. This subject will be addressed during the staff review of the ISI program.

SRP Section 5.2.4, Paragraph II.8, "Acceptance Criteria, Relief Requests," has not been completed because the applicant has not identified all limitations to examination. Specific areas where ASME Code examination requirements cannot be met will be identified as performance of the PSI progresses. The staff's complete evaluation of the PSI program will be presented in a supplement to the SER after the applicant (1) submits the required examination information and identifies all plant-specific areas where ASME Code, Section XI, requirements cannot be met and (2) provides a supporting technical justification.

5.2.4.2 Examination Requirements

GDC 32, "Inspection of Reactor Coolant Pressure Boundary," Appendix A of 10 CFR 50 requires, in part, that components that are part of the RCPB be designed to permit periodic inspection and testing of important areas and features to assess their structural and leaktight integrity. To ensure that no deleterious defects develop during service, selected welds and weld heat-affected zones (HAZs) will be inspected periodically at Millstone Unit 3.

The design of the ASME Code, Class 1 and 2, components of the RCPB incorporates provisions for access for inservice inspection, as required by Subarticle IWA-1500 of Section XI of the ASME Code. 10 CFR 50.55a(g) defines the detailed requirements for the PSI and ISI programs for light-water-cooled nuclear power facility components. On the basis of the construction permit date of August 9, 1974, this section of the regulations requires that a PSI program be developed and implemented using at least the edition and addenda of Section XI of the ASME Code applied to the construction of the particular components. Also, the initial ISI program must comply with the requirements of the latest edition and addenda of Section XI of the ASME Code in effect 12 months before the date the operating license is issued, subject to the limitations and modifications listed in 10 CFR 50.55a(b).

5.2.4.3 Evaluation of Millstone Nuclear Power Station, Unit 3 Compliance With 10 CFR 50.55a(g)

A PSI program for Millstone Unit 3, based on the 1980 Edition through the Winter 1980 Addenda of Section XI of the ASME Code, was submitted by the applicant in letters dated June 1, 1983 and March 20, 1984. The preservice examination of ASME Code, Class 1, components is being performed in accordance with Section XI of the ASME Code, 1980 Edition, to the extent practical within the access provided for inspection and the limitations of component geometry. The PSI program describes the components subject to examination, the examination methods, the components exempt from volumetric and surface examination based on Section XI exclusion criteria, and the inspection isometric drawings.

On March 23, 1983, Generic Letter 83-15 was sent to all applicants for operating licenses to implement RG 1.150, Revision 1. In a letter dated May 9, 1984, the applicant submitted information on the PSI program for the reactor vessel. This document provided a detailed description of the equipment, calibration sequence, examination techniques, and recording requirements for the preservice examination of the reactor vessel with the remotely operated inspection tool. The staff has determined that the preservice examination procedure for the reactor vessel will meet the intent of RG 1.150.

The staff has found that certain ultrasonic techniques may not be fully adequate to consistently detect and reliably characterize service-induced flaws during the inservice inspection of thick-wall cast stainless steel components to the acceptance standards of Paragraph IWB-3500 of Section XI. The applicant will provide an onsite demonstration of the adequacy of the ultrasonic inspection techniques for the staff in August 1984. This item is confirmatory and will be addressed in a supplement to this report.

As a result of the staff's review of the PSI program the selection of the ASME Code, Class 1 components in the RCPB that are subject to examination has been determined to be acceptable. The applicant has stated that he will identify areas where ASME Code requirements cannot be met and will request relief from these requirements. The applicant should commit to identify all plant-specific areas where the Code requirements cannot be met when the preservice examinations are completed and should provide a supporting technical justification.

The staff will evaluate all requests for relief in a supplement to the SER after the required documentation has been submitted by the applicant. Therefore, the staff considers the review of the PSI program and the relief requests a confirmatory issue.

The applicant has not submitted the initial ISI program for Millstone Unit 3. The staff will evaluate the program after the operating license has been issued; at that time the applicable ASME Code edition and addenda can be determined based on 10 CFR 50.55a(b). The staff evaluation will be completed before the first refueling outage when inservice inspections will be performed.

5.2.4.4 Conclusions

Periodic inspections and hydrostatic testing of pressure-retaining components of the RCPB, in accordance with the requirements of Section XI of the ASME Code and 10 CFR 50, will provide reasonable assurance that evidence of structural degradation or loss of leaktight integrity occurring during service will be detected in time to permit corrective action before the safety functions of a component are compromised. Compliance with the PSI and ISI required by the Code and 10 CFR 50 constitutes an acceptable basis for satisfying the inspection requirements of GDC 32.

5.2.5 Reactor Coolant Pressure Boundary Leakage Detection

The RCPB leakage detection systems were reviewed in accordance with SRP Section 5.2.5 (NUREG-0800). Conformance with the acceptance criteria formed the basis for the staff evaluation of the RCPB leakage detection systems with respect to the applicable regulations of 10 CFR 50.

A limited amount of leakage is to be expected from components forming the RCPB. Means are provided for detecting and identifying this leakage in accordance with the requirements of GDC 30. Leakage is classified into two types: identified and unidentified. Components such as valve stem packing, pump shaft seals, and flanges are not completely leaktight. Because this leakage is expected, it is considered as identified leakage and is monitored, limited, and separated from all other leakage (unidentified) by directing it to closed systems as identified in the guidelines of RG 1.45, Position C.1.

In the containment building, identified leakage from valve stems, pump seals, the reactor vessel flange, and pressurizer relief valves is kept within a closed system by being piped to the containment drains transfer tank or pressurizer relief tank. Flow or temperature devices are provided in the leakoff lines to indicate the source of leakage. The containment drains transfer tank and the pressurizer relief tank are monitored for pressure, temperature, and water level. Leakage collected in these tanks is pumped to the radioactive gaseous waste system or the boron recovery system through flow-monitoring devices.

All RCPB leakage in the containment structure that is not collected in the containment drains transfer tank or in the pressurizer relief tank is collected in the unidentified leakage sump. Unidentified leakage is monitored by sump level and sump pump run time monitoring systems that are capable of detecting a 1-gpm change in the leakage rate into the sump within 1 hour. The applicant has indicated that the sump pump monitoring system is not seismic Category I, but is expected to remain operable during all seismic events that do not require a plant shutdown. After a seismic event, the operability of the sump level monitoring system will be verified. If the instrumentation is not available to detect a 1-gpm leakage rate in 1 hour, the appropriate action according to the plant Technical Specifications will be taken. Indication, alarm, and means to determine leak rate in gallons per minute is provided in the control room. The applicant is also providing a sump level monitoring system that will alarm, via the plant computer system, on an increase in level that corresponds to a 1-gpm leakage rate within 1 hour. Thus, the guidelines of RG 1.45, Position C.2, regarding collection of unidentified leakage and flow monitoring are met.

Unidentified leakage is also detected by containment airborne particulate radioactive monitors and containment gaseous radioactive monitors that are qualified to remain functional when subjected to the safe shutdown earthquake. This meets the guidelines of RG 1.29, Positions C.1 and C.2, and RG 1.45, Position C.6. These monitors respond to the increase in airborne radioactivity resulting from leakage. The time to detect reactor coolant leakage by airborne particulate and gaseous radioactive monitors depends on reactor coolant activity level, location of leakage, leak rate, and background concentration from previous leakage. With no prior leakage into the containment and with expected reactor coolant activity, a 1-gpm leakage rate can be detected in 5 min with the particulate monitoring system and in 40 min with the gaseous monitoring system.

Indicators and alarms are provided in the control room to indicate high activity in the containment. The procedures for converting various indicators to a common leakage equivalent will be available to the operator.

As a backup, unidentified leakage also is detected by pressure, temperature, and humidity monitors, which are capable of detecting a 5-gpm leak rate in less than 1 hour under normal operating conditions. Indications and alarms are provided in the control room. Thus, the guidelines of RG 1.45, Positions C.3 and C.5, regarding methods of unidentified leak detection and sensitivity, are met.

For intersystem monitoring, radiation monitors are used to detect reactor coolant leakage into the reactor plant component cooling water system, which supplies the residual heat removal (RHR) heat exchangers, letdown heat exchangers, reactor coolant seal water, and thermal barrier heat exchangers. Leakage through steam generator tubes is detected by a radiation monitor in the condenser's air ejector vent line and by using the secondary side sampling system. Accumulator leakage is detected by level and pressure instruments provided for each accumulator. Thus, the guidelines of RG 1.45, Position C.4, regarding intersystem leakage are satisfied.

The applicant has provided indicators and alarms for each leak detection system in the control room and has provisions for testing and calibration of the systems during plant operation. Thus, the guidelines of RG 1.45, Positions C.7 and C.8, regarding instruments and alarms and provisions for testing and calibration are satisfied.

The Technical Specifications are not available at this time. Therefore, conformance with the guidelines in RG 1.45, Position C.9, regarding limiting conditions for identified and unidentified leakage will be confirmed during the Technical Specification review in SER Section 16.

The leakage detection systems, provided to detect leakage from components of the RCPB, furnish reasonable assurance that structural degradation, which may develop in pressure-retaining components of the RCPB, will be detected on a timely basis so that corrective actions can be taken before such degradation could become sufficiently severe to jeopardize the safety of the system, or before the leakage could increase to a level beyond the capability of the makeup system to replenish the loss.

On the basis of this review, the staff concludes that the RCPB leakage detection system meets the requirements of GDC 2 and 30 with respect to protection against natural phenomena and provisions for RCPB leak detection and identification and the guidelines of RGs 1.29 (Positions C.1 and C.2) and 1.45 (Positions C.1 through C.9). The RCPB leakage detection system meets the acceptance criteria of SRP Section 5.2.5.

5.3 Reactor Vessel

The staff has reviewed the fracture toughness of ferritic reactor vessel and RCPB materials and the materials surveillance program for the reactor vessel beltline. The acceptance criteria and references that are the basis for this evaluation are in Paragraphs II.5, II.6, and II.7 (Appendices G and H, 10 CFR 50) of SRP Section 5.3.1 (NUREG-0800).

GDC 31 requires, in part, that the RCPB be designed with sufficient margin to ensure that, when stressed under operating, maintenance, and test conditions, the boundary behaves in a nonbrittle manner and the probability of rapidly propagating fracture is minimized. GDC 32 requires, in part, that the RCPB boundary be designed to permit an appropriate material surveillance program for the reactor pressure vessel. The fracture toughness requirements for the ferritic materials of the RCPB are defined in Appendices G and H, of 10 CFR 50.

5.3.1 Reactor Vessel Materials

Integrity of the reactor vessel studs and fasteners is ensured by conformance with most of the recommendations of RG 1.65, "Materials and Inspections for Reactor Vessel Closure Studs." The applicant's alternate approach of relying on metallurgical factors and specifying high fracture toughness as determined by Charpy V-notch testing to indirectly control ultimate tensile strength is acceptable to the staff. Compliance with these recommendations and the applicant's alternate approach satifies the quality standards requirements of GDC 1, GDC 30, and 10 CFR 50.55a; the prevention of fracture of the RCPB requirement of GDC 31; and the requirements of Appendix G, 10 CFR 50, as detailed in the provisions of the ASME Code, Sections II and III.

5.3.1.1 Compliance With 10 CFR 50.55a

The edition and addenda of the ASME Code that are applicable to the design and fabrication of the reactor vessel and RCPB components are specified in 10 CFR 50.55a. The ASME Code edition and addenda that are required depend on the date the construction permit was issued. The construction permit for Millstone Unit 3 was issued on August 9, 1974. On the basis of the construction permit date, 10 CFR 50.55a requires that ferritic materials used for the Millstone Unit 3 reactor vessel be designed and constructed to editions that are no earlier than the Summer 1972 Addenda to the 1971 ASME Code (hereinafter Code) and that ferritic materials used in piping, pumps, and valves be constructed to editions that are no earlier than the Winter 1972 Addenda to the Code. The Millstone Unit 3 ferritic materials meet all the above requirements with the exception of the loop safety valves, which were constructed to the Summer 1972 Addenda to the Code.

Branch Technical Position MTEB 5-2 requires that the fracture toughness of ferritic RCPB materials must be assessed to the requirements of the Code, as augmented by Appendix G, 10 CFR 50. The staff has assessed whether the loop safety valves are acceptable to the above requirements in its review of the applicant's compliance with Appendix G, 10 CFR 50

5.3.1.2 Materials and Fabrication

The staff concludes that the reactor vessel materials are generally acceptable and meet the requirements of GDC 1, 4, 14, 30, 31, and 32 of Appendix A of 10 CFR 50; the material testing and monitoring requirements of Appendices B, G, and H of 10 CFR 50; and the requirements of 10 CFR 50.55a.

The materials used for construction of the reactor vessel and its appurtenances have been identified by specification and found to be in conformance with Section III of the ASME Code. Special requirements of the applicant with regard to control of residual elements have been identified and are considered acceptable. Compliance with the above Code provisions for material specifications satisfies the quality standards requirements of GDC 1 and 30 and 10 CFR 50.55a.

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

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

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

(2) RG 1.34 is not applicable because this process is not used in reactor vessel fabrication.

(3) The controls imposed during weld cladding of ferritic steel components are in conformance with the recommendations of RG 1.43, "Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components." Accordingly, reasonable assurance is provided that underclad cracking did not occur during the weld cladding process. These controls satisfy the quality standards requirements of GDC 1 and 30 and 10 CFR 50.55a.

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

(1) The controls imposed on delta ferrite in austenitic stainless steel welds satisfy, to the extent practical, the recommendations of RG 1.31, "Control of Ferrite Content in Stainless Steel Weld Metal." The staff has reviewed the alternate approaches taken by the applicant and finds them acceptable (see Section 4.5.1). The controls used provide reasonable assurance that the welds do not contain microcracks. These controls also satisfy the quality standards requirements of GDC 1 and 30 and 10 CFR 50.55a and the requirement of GDC 14 regarding fabrication to prevent RCPB rapid propagating failure.

(2) RG 1.34 is not applicable because this process is not used in the fabrication of reactor vessels.

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

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

(2) The controls regarding onsite cleaning and cleanliness controls of austenitic stainless steel are in conformance with the recommendations of RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants." These controls provide assurance that austenitic stainless steel components were properly cleaned on site and satisfy Appendix B of 10 CFR 50 regarding controls for onsite cleaning of materials and components.

5.3.1.3 Compliance With Appendix G, 10 CFR 50

The staff's evaluation of the Millstone Unit 3 FSAR to determine the degree of compliance with the fracture toughness requirements of Appendix G, 10 CFR 50, indicates that the applicant has met all the requirements of this appendix.

As the staff has indicated in Section 5.3.1.1, loop safety valves were constructed to the Summer 1972 Addenda to the 1971 Edition of the Code, whereas

10 CFR 50.55a requires that they should be constructed to the Winter 1972 Addenda to the Code. However, the fracture toughness requirements of the Winter 1972 Addenda are the same as those of the Summer 1972 Addenda for the loop safety valve materials. Thus, if the loop safety valves meet the fracture toughness requirements of the Summer 1972 Addenda, they meet those of the Winter 1972 Addenda, and the requirements of Appendix G, 10 CFR 50, are satisfied.

5.3.1.4 Compliance With Appendix H, 10 CFR 50

The materials surveillance program at Millstone Unit 3 will be used to monitor changes in the fracture toughness properties of ferritic materials in the reactor vessel beltline region resulting from exposure to neutron irradiation and the thermal environment as required by GDC 32, "Inspection of Reactor Coolant Pressure Boundary." The surveillance program, which must be in compliance with Appendix H, 10 CFR 50, requires that fracture toughness data be obtained from material specimens that are representative of the limiting base, weld, and heat-affected zone materials in the beltline region. These data will permit the determination of the conditions under which the vessel can be operated with adequate margins of safety against fracture throughout its service life. As a result of the information supplied by the applicant, the staff has determined that the surveillance program has met all the requirements of Appendix H, 10 CFR 50.

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

On the basis of its evaluation of compliance with Appendices G and H, 10 CFR 50, the staff concludes that the applicant has met all the fracture toughness requirements of these appendices.

Appendix G, "Protection Against Non-Ductile Failures," Section III of the ASME Code, will be used, together with the fracture toughness test results required by Appendices G and H, 10 CFR 50, to calculate the pressure-temperature limitations for the Millstone Unit 3 reactor vessel.

The fracture toughness tests required by the ASME Code and by Appendix G of 10 CFR 50 provide reasonable asssurance that adequate safety margins against the possibility of nonductile behavior of rapidly propagating fracture can be established for all pressure-retaining components of the reactor coolant boundary. The use of Appendix G, Section III of the ASME Code, as a guide in establishing safe operating procedures, and the use of the results of the fracture toughness tests performed in accordance with the ASME Code and NRC regulations will provide adequate safety margins during operating, testing, maintenance, and anticipated transient conditions. Compliance with these Code provisions and NRC regulations constitutes an acceptable basis for satisfying the requirements of GDC 31.

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

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

5.3.2 Pressure-Temperature Limits

The staff has reviewed the applicant's pressure-temperature limits for operation of the reactor vessel. The acceptance criteria and list of references that are the basis for this evaluation are set forth in SRP Section 5.3.2 (NUREG-0800).

Appendices G and H, 10 CFR 50, describe the conditions that require pressure-temperature limits and provide the general bases for these limits. These appendices specifically require that pressure-temperature limits must provide safety margins at least as great as those recommended in ASME Code, Section III, Appendix G, "Protection Against Non-Ductile Failures." Appendix G, 10 CFR 50, requires additional safety margins for the closure flange region materials and beltline materials whenever the reactor core is critical, except for low-level physics tests.

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

(1) preservice hydrostatic tests
(2) inservice leak and hydrostatic tests
(3) heatup and cooldown operations
(4) core operation

The pressure-temperature limit curves, which were submitted for review, are in compliance with the requirements of Appendix G, 10 CFR 50.

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

5.3.3 Reactor Vessel Integrity

The staff has reviewed the FSAR sections related to the reactor vessel integrity of Millstone Unit 3. Although most areas are reviewed separately in accordance with other review plans, reactor vessel integrity is of such importance that a special summary review of all factors relating to reactor vessel integrity is warranted. The staff has reviewed the fracture toughness of ferritic reactor vessel and reactor coolant pressure boundary materials, the pressure-temperature limits for operation of the reactor vessels, and the materials surveillance program for the reactor vessel beltline. The acceptance criteria and references that are the basis for the evaluation are set forth in Paragraphs II-2, II.6, and II.7 (Appendices G and H, 10 CFR 50) of SRP Section 5.3.3 (NUREG-0800).

The staff has reviewed the above factors contributing to the structural integrity of the reactor vessel and concludes that the applicant has fully complied with the requirements of Appendices G and H, 10 CFR 50.

The staff has reviewed all factors contributing to the structural integrity of the reactor vessel and concludes there are no special considerations that make it necessary to consider potential reactor vessel failure for Millstone Unit 3.

5.4 Component and Subsystem Design

5.4.1 Reactor Coolant Pump Flywheel Integrity

The objective of this review is to ensure that the integrity of the primary reactor coolant pump flywheel is maintained to prevent failure at normal operating speeds and at speeds that might be reached under accident conditions and thus preclude the generation of missiles.

The basis for review is outlined in SRP Section 5.4.1.1 and RG 1.14, which describe and recommend a method acceptable to the staff in implementing GDC 4 of Appendix A of 10 CFR 50 with regard to minimizing the potential for failure of flywheels of the reactor coolant pump. Flywheels are fabricated from SA-533, Grade B, Class 1 steel and consist of two thick plates bolted together. The material is vacuum melted with a degassing process. The materials as well as finished flywheels are subjected to 100% volumetric ultrasonic inspection using procedures and acceptance standards specified in Section III of the ASME Code.

The nil-ductility transition temperature (NDTT) of the flywheel material is obtained by two drop-weight tests (DWTs) that exhibit "no break" performance at 20°F in accordance with ASTM E-208. The Charpy V-notch energy level is at least 50 ft-lb in the "weak" direction (WR orientation) at 70°F. Hence, the RT_{NDT} of 10°F can be assumed. The above DWTs also demonstrate that the NDTT of the material is no higher than 10°F.

The calculated stresses at the operating speed, as a result of centrifugal forces and the interference fit on the shaft, are within the regulatory guide limits. The pump runs at 1,185 rpm and may operate briefly at an overspeed of 110% during the loss of outside load. The design speed is 125% of the operating speed. The flywheels also are tested at 125% of the maximum synchronous speed of the motor. The combined stresses at the design overspeed, resulting from interference fit and centrifugal forces, are within the regulatory guide limit.

The flywheels can be inspected by removing the cover. Hence, any crack that develops can be noticed. The critical crack length at the keyways, where the stress concentration is high, is about 6 in. at the design overspeed.

Results for a double-ended guillotine break at the pump discharge, with full separation of pipe ends assumed, show the maximum overspeed was calculated to be about 280% of the normal speed for the above break with an assumed instantaneous loss of power to the pump. In comparison with the overspeed presented above, the flywheel could withstand a speed up to 2.3 times greater than the flywheel spin test speed of 125% with flaws no greater than 1.15 in.

Westinghouse uses the combined primary stress levels as defined in Revision 0 of RG 1.14 rather than those in RG 1.14, Revision 1.

The staff has reviewed the material, fabrication, design, and inspection aspects of the pump flywheels for compliance with RG 1.14. The staff has concluded that the structural integrity of the flywheels is adequate to withstand the forces imposed by overspeed transients without the loss of function. The integrity of the flywheels will be verified periodically by inspection to ensure that integrity is maintained. Exceptions to RG 1.14, Revision 1, are acceptable because the critical crack length of 1 in. from the keyways can be tolerated at 200% to 250% of the normal operating speed according to proposed Revision 2 of RG 1.14. Hence, the integrity of the pump flywheel will be maintained.

5.4.2 Steam Generators

5.4.2.1 Steam Generator Materials

The staff concludes that the steam generator materials specified are acceptable and meet the requirements of GDC 1, 14, 15, and 31 and Appendix B to 10 CFR 50. This conclusion is based on the following:

(1) The applicant has met the requirements of GDC 1 with respect to codes and standards by ensuring that the materials selected for use in Class 1 and Class 2 components were fabricated and inspected in conformance with codes, standards, and specifications acceptable to the staff. Welding qualification, fabrication, and inspection during manufacture and assembly of the steam generators were done in conformance with the requirements of Sections III and IX of the ASME Code.

(2) The requirements of GDC 14 and 15 have been met to ensure that the reactor coolant boundary and associated auxiliary systems have been designed, fabricated, erected, and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure, and of gross rupture, during normal operation and anticipated operational occurrences.

The primary side of the steam generator is designed and fabricated to comply with ASME Code, Class 1 criteria as required by the staff. The secondary side pressure boundary parts of the steam generator are designed, manufactured, and tested to ASME Code, Class 2.

The crevice between the tube sheet and the inserted tube is minimal because the tube was expanded to the full depth of insertion of the tube in the tube sheet. The tube expansion and subsequent positive contact pressure between the tube and the tube sheet preclude impurities from building up in the crevice region and reduce the probability of crevice boiling.

The tube support plates were manufactured from ferritic stainless steel material, which has been shown in laboratory tests to be corrosion resistant to the operating environment. The tube support plates were designed and manufactured with broached holes rather than drilled holes. The broached hole design promotes high velocity flow along the tube, sweeping impurities away from the support plate's locations.

(3) The requirements of GDC 31 have been met with respect to the fracture toughness of ferritic materials since the pressure boundary materials of ASME Code, Class 1 components of the steam generator comply with the fracture toughness requirements and tests of Subarticle NB-2300 of Section III of the Code. The materials of the ASME Code, Class 2 components of the

steam generator comply with the fracture toughness requirements of Subarticle NC-2300 of Section III of the Code.

(4) The requirements of Appendix B of 10 CFR 50 have been met since the onsite cleaning and cleanliness controls during fabrication conform to the recommendations of RG 1.37. The controls placed on the secondary coolant chemistry are in agreement with staff technical positions.

Reasonable assurance of the satisfactory performance of the steam generator tubing and other generator materials is provided by (a) the design provisions and the manufacturing requirements of the ASME Code, (b) rigorous secondary water monitoring and control, and (c) the limiting of condenser inleakage. The controls described above combined with conformance with applicable codes, standards, staff positions, and regulatory guides constitute an acceptable basis for meeting in part the requirements of GDC 1, 14, 15, and 31, and Appendix B, 10 CFR 50.

5.4.2.2 Steam Generator Tube Inservice Inspection

5.4.2.2.1 Compliance With the Standard Review Plans

Millstone Unit 3 was reviewed in accordance with SRP Section 5.4.2.2. However, the review is continuing because the applicant has not submitted the inspection program in accordance with the Standard Technical Specifications (STS). The results of the staff review to date are summarized below.

The SRP acceptance criteria recommend that the applicant perform inspections based on RG 1.83 and the applicable STS. The applicant has committed to perform the inspection of the steam generator tubes in accordance with RG 1.83, Revision 1, and the STS. The staff will document its conclusions regarding the examinations of the steam generator tube in a supplement to the SER.

5.4.2.2.2 Evaluation of the Inspection Program

GDC 32, Appendix A of 10 CFR 50, requires, in part, that components that are part of the RCPB be designed to permit periodic inspection and testing of important areas and features to assess their structural and leaktight integrity. The steam generators at Millstone Unit 3 have been designed to meet the ASME Code requirements for Class 1 and 2 components. Provisions also have been made to permit inservice inspection of the Class 1 and 2 components, including individual steam generator tubes. The design aspects that provide access for inspection and the proposed inspection program must follow the recommendations of RG 1.83, Revision 1, and NUREG-0452, Revision 2. The design aspects also must comply with the requirements of Section XI of the ASME Code with respect to the inspection methods to be used, provisions for a baseline inspection, selection and sampling of tubes, inspection intervals, and actions to be taken in the event that defects are identified.

The applicant has committed to perform the preservice and inservice inspection of the steam generator tubes in accordance with RG 1.83, Revision 1; the staff finds this acceptable. However, the applicant has not submitted the Technical Specification in conformance with the recommendations in Section 3/4.4.5, "Steam Generators," Paragraph 4.4.5.4a.9, "Preservice Inspection," and Paragraph 4.4.5.5, "Reports," of NUREG-0452, Revision 2.

Because the applicant has made commitments in the FSAR to meet the technical requirements of Section 3/4.4.5 of NUREG-0452, the staff considers this to be a confirmatory item. The staff will address this issue in a supplement to this SER and in its review of the final Technical Specifications.

5.4.2.2.3 Conclusions

Conformance with RG 1.83, NUREG-0452, and the inspection requirements of Section XI of the ASME Code constitutes an acceptable basis for meeting, in part, the requirements of GDC 32.

5.4.3*

5.4.4*

5.4.5*

5.4.6 Reactor Core Isolation Cooling System

The reactor core isolation cooling (RCIC) system is a system that is used in boiling water reactors (BWRs). It is not found in Millstone Unit 3, which is a pressurized water reactor (PWR). A review under SRP Section 5.4.6 is not applicable.

5.4.7 Residual Heat Removal System

The residual heat removal system (RHRS) for Millstone Unit 3 has been reviewed in accordance with SRP Section 5.4.7 (NUREG-0800). Conformance with the acceptance criteria, except as noted below, formed the basis for concluding that the design of the facility for residual heat removal is acceptable.

The RHRS is designed to remove heat from the reactor coolant system after the system temperature and pressure have been reduced to approximately 350°F and 425 psig, respectively. The RHRS is capable of reducing the reactor coolant temperature to the cold shutdown condition and maintaining this temperature until the plant is started up again.

The RHRS operates in the following modes:

(1) Emergency Core Cooling System (ECCS), Injection Mode

The RHRS functions in conjunction with the high head portion of the ECCS to provide injection of borated water from the refueling water storage tank (RWST) into the RCS cold legs during the injection phase following a loss-ofcoolant accident (LOCA).

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*The July 1981 Standard Review Plan (NUREG-0800) does not include sections addressing FSAR sections that consist of background or design data used in the review of other sections. The section numbers are retained in this SER to provide continuity and ensure a close correlation between subsequent SER sections and their associated SRP sections.

(2) Refueling

Both RHR pumps may be used during refueling to pump borated water from the RWST to the refueling cavity. Following refueling, the RHR pumps are used to drain the refueling cavity to the top of the reactor vessel flange by pumping water from the RCS to the RWST.

(3) Cold Shutdown

The RHRS removes RCS decay heat and maintains cold shutdown conditions. The relief valve on the RHRS suction line may be used for low-temperature overpressure protection backup.

(4) Startup

The RHRS is connected to the chemical and volume control system (CVCS) via the low-pressure letdown line to control reactor coolant pressure. The relief valve on the RHRS suction line may be used for low-temperature overpressure protection backup.

Design parameters for the RHRS are as follows:

(1)	design pressure (psig)	600*
(2)	design temperature (°F)	400*
(3)	pump capacity (gpm)	4,000
(4)	number of independent trains	2

The two RHR trains are independent in action and powered by separate essential power supplies to provide redundancy. With only one RHR pump and heat exchanger in service and the heat exchanger supplied with component cooling water at a design flow and temperature of 3.3×10^6 lb per hour and 92.2°F, respectively, the RHRS is capable of reducing the reactor coolant temperature from 350°F to 200°F within 30 hours.

5.4.7.1 Functional Requirements

The RHRS for Millstone must meet GDC 1 through 5. GDC 1 through 4 regarding quality standards and records, design bases for protection against natural phenomena, fire protection, and environmental and missile design bases are covered in Sections 3.4.1, 9.5.1, 3.3.1, and 3.5.1 of this report, respectively. GDC 5 regarding sharing of structures, systems and components is met for the Millstone RHRS since components are not shared between units.

During normal plant shutdown when nonsafety-related equipment and offsite power are assumed available, the decay heat removal function is performed by using the main feedwater system, the condenser steam dump system, and service water system. During the plant's emergency shutdown, assuming offsite power and nonsafety-related equipment are not available, the heat is transferred from the core by natural circulation with the steam generator as the heat sink.

To achieve this, the safety-related steam generator safety valves and power-operated atmospheric relief valves are used to vent secondary steam. Only two

*Applicable to the tube side of the residual heat exchanger.

out of four atmospheric relief valves need to be operable for plant cooldown. Secondary coolant makeup is provided via the auxiliary feedwater system (AFWS) from the seismic Category I tornado-missile-protected demineralized water storage tank (DWST) with a total capacity of 360,000 gal. The applicant states that 340,000 gal are sufficient to maintain the plant in a hot standby condition for up to 10 hours then cooling the reactor to 350°F hot-leg temperature within 6 hours, at which time the RHRS will be initiated. The entire DWST content is exclusively reserved for the AFWS. A single failure of any active component would not render all steam generators ineffective as a heat sink. Any one of the three auxiliary feedwater pumps has sufficient capacity to provide for all steam generator makeup requirements.

The reactor coolant system (RCS) depressurization is accomplished by the combination of RCS contraction resulting from the cooldown or opening of one of the two safety-related pressurizer power-operated relief valves. The discharge is directed to the pressurizer relief tank where it is condensed and cooled.

The depressurization process is integrated with the cooldown process to maintain the RCS within normal pressure-temperature limits. Just before initiating RHR cooling at 350°F, the RCS is depressurized to less than 425 psig.

The second stage of the cooldown is from 350°F to cold shutdown. During this stage, the RHRS is brought into operation. Circulation of the reactor coolant is provided by the RHR pumps, and the heat exchangers in the RHRS serve as the means of heat removal from the RCS. In the RHR heat exchangers, the residual heat is transferred to the component cooling water system which ultimately transfers the heat to the service water system and the ultimate heat sink.

The RHRS is a fully redundant system. Each RHR subsystem includes one RHR pump and one RHR heat exchanger. Each RHR pump is powered from a different emergency bus, and each RHR heat exchanger is served from a different component cooling water system loop. Portions of the component cooling water system and the service water system associated with the RHR system are designed and constructed to safety-grade standards. All systems are capable of being operated from the control room with either only onsite or only offsite power.

If any component in one of the RHR subsystems was rendered inoperable as the result of a single failure, cooldown of the plant could still be achieved by using the remaining operable subsystem of the system.

The staff has asked the applicant to address situations when the reactor coolant system has been partially drained, improper reactor coolant inventory, or operating the RHRS at an inadequate NPSH has resulted in air binding of the RHR pumps with a subsequent loss of shutdown cooling. In Revision 1 to the FSAR, the applicant indicated that Technical Specifications specify that multiple heat removal paths be available from RHRS, operable steam generators, or a combination of both. The redundancy and independence of these paths ensures a high degree of reliability in heat removal capability. Also, the reactor coolant level is monitored to ensure that the RHR system inlet lines do not become uncovered. This will minimize the possibility of air entrainment. Particular attention is placed on specific pump venting procedures to ensure that the redundant RHR pump does not become airbound. The staff finds the applicant's response acceptable.

Core reactivity is controlled during the cooldown by adding borated water to the RCS in conjunction with the cooldown. Boration is accomplished using safety-related portions of the chemical and volume control systems. During the cooldown one of the three centrifugal charging pumps would take suction from one of the two boric acid tanks (BATs) and inject borated water into the RCS. The capacity of one BAT is sufficient to make up for reactor coolant contraction as a result of RCS cooldown from normal operating temperatures to the temperature when RHR initiation can commence. The two BATs, three centrifugal charging pumps, and the associated piping and valves are designed to safety-grade standards. The backup borated water sources are provided from the RWST.

All systems are capable of being operated from the control room. Only in the event of a most limiting single failure (i.e., the failure of an RHR suction isolation valve interlock circuitry or emergency generator failure in conjunction with loss of offsite power) is limited operator action outside the control room required to open the suction isolation valve. The applicant stated that the spurious opening of those motor-operated RHR suction isolation valves (in the event of a fire) could result in overpressurization of the low-pressure RHR piping. To preclude this possibility, the power to these valves will be removed at the motor control center (MCC) breakers during power operation. The MCCs are located in the auxiliary building and are accessible to the operators without being subjected to adverse environment such as high temperature or high radioactivity. The water supply provided to the auxiliary feedwater system to enable the plant to achieve a safe shutdown condition is sufficient to hold the plant at hot standby for up to 10 hours and to provide a cooldown period of 6 hours to 350°F hot-leg temperature, at which time the RHRS can be initiated. If operator action is required to open the RHR isolation valve, the operator would have ample time to perform the task. The staff considers this justification acceptable.

Redundancy in the RHRS is provided by two independent trains for each unit. Leak detection for the RHRS is discussed in Section 5.2.5 of this SER. Isolation valve and power supply redundancy is discussed in Section 5.4.7.2 in this chapter. The staff has reviewed the description of the RHRS and the piping and instrumentation diagrams to verify that the system can be operated with or without offsite power and assuming a single failure. The two RHR pumps are connected to separate buses which can be powered by separate emergency diesel generators in the event of loss of offsite power.

SRP Section 5.4.7 states that the RHRS should be operable from the control room in accordance with GDC 19. Limited manual actions are permitted outside the control room after a single failure, if justified.

In accordance with Branch Technical Position RSB 5-1, the system(s) should be capable of bringing the reactor to a cold shutdown condition with only offsite or onsite power available within a reasonable period of time following shutdown, assuming the most limiting single failure. A reasonable period of time is considered to be 36 hours. The staff has asked the applicant to identify the most limiting single failure and provide an analysis to show that the reactor can be brought to the RHR entry condition within 36 hours. In Amendment 6, the applicant provided a failure mode and effects analysis which indicated that the most limiting single failure would be the loss of one redundant RHR train. The failure would not affect the operation of the remaining RHR train since one RHR train would be sufficient to cool down the reactor from 350°F to 200°F within 30 hours.

The applicant further stated that the RHRS would be designed according to seismic Category I and safety-grade requirements, and only safety-grade equipment would be needed for safe shutdown of the plant. The staff finds the applicant's response acceptable.

5.4.7.2 RHR System Isolation Requirements

The RHRS valving arrangement is designed to provide adequate protection to the residual heat removal system from overpressurization when the reactor coolant system is at high-pressure operation.

The RHRS is isolated from the RCS on the suction side by three normally closed, motor-operated valves in series on each suction line. They are closed during normal operation and are opened only for residual heat removal operation during a plant cooldown after the RCS pressure is reduced to 425 psig or lower and RCS temperature is reduced to approximately 350°F. Two of the motor-operated valves in each inlet line are provided with both "prevent open" and "auto closure" interlocks which are designed to prevent possible exposure of the RHRs to normal RCS operating pressure. The "prevent open" interlock will prevent the valves from opening if the RCS pressure is greater than 425 psig. The "auto closure" will close the valves automatically if the RCS pressure exceeds 750 psig.

The use of two independently powered motor-operated valves in each of the two inlet lines, along with two independent pressure interlock signals for each function, ensures a design that meets applicable single-failure criteria. The RHRS inlet isolation valves are provided with red-green position indicator lights on the main control board and the auxiliary shutdown panel. This system design is not as prone to spurious valve closure as other designs where a loss of power to the solid-state protection system (SSPS) will initiate valve closure.

Isolation on the discharge portion of the RHRS from the high-pressure RCS is provided by a normally open motor-operated valve and three check valves in series. These check valves are located in the ECCS. The staff finds the RHRS isolation design acceptable.

5.4.7.3 RHR Pressure Relief Requirements

Overpressure protection of the residual heat removal system is provided by four relief valves, one on each of the suction and discharge lines. Each suction line relief valve has a capacity of 900 gal per minute (gpm) at 450 psig which is sufficient to discharge the flow from both charging pumps at the relief valve setpoint. Each discharge line from the RHRS to the RCS is equipped with a pressure relief valve to relieve the maximum possible backleakage through the valves separating the RHRS from the RCS. Each valve has a relief flow capacity of 20 gpm at a set pressure of 600 psig. The fluid discharge through the suction side relief valves is collected in the pressurizer relief tank. The fluid discharged through the discharge side relief valves is collected in the primary drain tank of the equipment and floor drain system. These relief valves are adequate to protect the residual heat removal system from overpressurization. The staff concludes that this design is acceptable.

5.4.7.4 RHR Pump Protection

Each of the two RHR pumps has a mini-flow bypass line to prevent overheating in the event of a loss of adequate discharge flow, and to prevent pump deadheading. A valve located in each mini-flow line is regulated by a signal from the flow transmitters located in each pump discharge header. The control valves open when the RHR pump discharge flow is less than 500 gpm and close when the flow exceeds 1,000 gpm; flow indicators are provided in the control room. A pressure sensor in each pump discharge header provides a signal for an indicator in the control room. A high-pressure alarm is also actuated by the pressure sensor. Low-flow alarm has been provided to alert the operator so that appropriate action can be taken to prevent RHR pump damage from low flow or low suction pressure. The staff finds the design acceptable.

5.4.7.5 Test, Operational Procedures, and Support Systems

The plant preoperational and startup test program provides for demonstrating the operation of the residual heat removal system in conformance with RG 1.68, "Initial Test Programs for Water Cooled Reactor Power Plants," as specified in SRP Section 5.4.7.III.12.

Verification of adequate mixing of borated water added to the RCS under natural circulation conditions and confirmation of natural circulation cooldown ability will be accomplished either by reference to the results of the tests from a plant of similar design or actual testing to be conducted at Millstone. At this time, the staff understands that the applicant intends to reference the tests to be conducted at the Diablo Canyon plant. The staff will require the applicant to provide a report justifying the applicability of the results of the boron mixing and natural circulation tests to be conducted at Diablo Canyon to the Millstone design. If the Diablo Canyon tests are not completed or do not provide satisfactory results to support the Millstone design, the applicant must perform such tests at Millstone during startup after the first refueling.

The staff has reviewed the portion of the component cooling water system which is required to provide cooling water to the RHRS to ensure that sufficient coolant flow is available to the RHRS heat exchangers and concludes that it is acceptable. The acceptability of this cooling capacity for other systems and its conformance to GDC 44, 45, and 46 are discussed in Section 9.2.2.

The applicant states that the RHRS is housed within a structure that is designed to withstand tornados, floods, and seismic phenomena.

The residual heat removal system's capability to withstand pipe whip inside containment as required by GDC 4 and RG 1.46 is discussed in Section 3.6.2. Protection against piping failures outside containment in accordance with GDC 4 is discussed in Section 3.6.1.

5.4.7.6 Conclusions

The residual heat removal function is accomplished in two phases: the initial cooldown phase and the residual heat removal system operation phase. In the event of loss of offsite power, the initial phase of cooldown is accomplished by use of the auxiliary feedwater system (AFWS) and the atmospheric dump valves. The AFWS, in conjunction with the steam generators and PORVs, is used to reduce the reactor coolant system temperature and pressure to the condition permitting operation of the RHRS. The RHRS removes core decay heat and provides long-term core cooling following the initial phase of reactor cooldown. The scope of review of the RHRS for the Millstone plant included piping and instrumentation diagrams, equipment layout drawings, failure modes and effects analysis, and design performance specifications for essential components. The review has included the applicant's proposed design criteria and design bases for the RHRS and analysis of the adequacy of those criteria and bases and the conformance of the design to these criteria and bases.

The staff concludes that the design of the RHRS is acceptable and meets the requirements of GDC 2, 5, 19, and 34. This conclusion is based on the following:

(1) The applicant has met GDC 2 with respect to Position C.2 of RG 1.29 concerning the seismic design of systems, structures, and components whose failure could cause an unacceptable reduction in the capability of the residual heat removal system.

(2) The applicant has met the requirements of GDC 5 with respect to sharing of structure, systems, and components by demonstrating that such sharing does not significantly impair the ability of the RHRS to perform its safety function including, in the event of an accident to one unit, an orderly shutdown and cooldown of the remaining units.

(3) The applicant has met GDC 19 with respect to the main control room requirements for normal operations and shutdown and GDC 34 which specifies requirements for the residual heat removal system by meeting the regulatory position in Branch Technical Position RSB 5-1.

The staff reviews of the following Task Action Plan Items are addressed in other sections of this report.

(1) Task Action Plan Item II.E.3.2 of NUREG-0660 as it relates to systems capability and reliability of shutdown heat removal systems under various transients

(2) Task Action Plan Item III.D.1.1 of NUREG-0737 as it relates to primary coolant sources outside containment

5.4.8 Reactor Water Cleanup System

SRP Section 5.4.8 provides for a review of the reactor water cleanup system as a system used in BWRs. Millstone Unit 3 is a PWR. A review under the provisions of SRP Section 5.4.8 is not applicable.

5.4.9*

5.4.10*

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*The July 1981 Standard Review Plan (NUREG-0800) does not include sections addressing FSAR sections that consist of background or design data used in the review of other sections. The section numbers are retained in this SER to provide continuity and ensure a close correlation between subsequent SER sections and their associated SRP sections.

5.4.11 Pressurizer Relief Tank (Pressurizer Relief Discharge System)

The pressurizer relief discharge system was reviewed in accordance with SRP Section 5.4.11 (NUREG-0800). Conformance with the acceptance criteria formed the basis for the staff evaluation of the pressurizer relief discharge system with respect to the applicable regulations of 10 CFR 50.

The pressurizer relief discharge system consists of the pressurizer relief tank, the discharge piping from the pressurizer relief and safety valves, the relief tank internal spray header, the tank nitrogen supply, the vent to containment, and the drain to the waste processing system. The system is nonsafety related, Quality Group D, nonseismic Category I and is not part of the RCPB because all of its components are downstream of the reactor coolant system safety and relief valves. Therefore, its failure would not affect the integrity of the RCPB.

The pressurizer relief tank is sized to absorb the energy content of 110% of the full-power pressurizer steam volume through the primary relief and safety valves. Other relief valve discharges to the pressurizer relief tank include those from the residual heat removal system and from the chemical and volume control system. Releases from these sources are less than the design-basis release from the pressurizer. The internal spray and bottom drain on the pressurizer relief tank are used to cool the water within the tank. A nitrogen blanket also is provided in the tank to permit expansion of entering steam and to control the tank internal atmosphere. If a discharge exceeding the design-basis release should occur, the rupture discs on the tank would pass the discharge through the tank to the containment.

The contents of the tank can be drained to the waste-holding tank in the radiation waste processing system or the boron recovery tank in the boron recycle system through the pressurizer relief tank drain transfer pumps. The rupture discs on the pressurizer relief tank have a relieving capacity equal to or greater than the combined capacity of the pressurizer safety valves. The tank and the rupture disc holders are designed for full vacuum to prevent collapse if the contents cool following a discharge without addition of nitrogen. The pressurizer relief tank is provided with control room instrumentation to indicate and alarm high pressure, high temperature, and high- and low-water levels.

The pressurizer relief tank is separated from safety-related equipment so that its failure would not compromise the capability to safely shut down the plant. Also, possible rupture disc fragments would not present a missile hazard to any safety-related equipment when the disc ruptures. Thus, the requirements of GDC 2 and 4 and the guidelines of RG 1.29, Positions C.2 and C.3, are satisfied.

On the basis of its review, the staff concludes that the pressurizer relief discharge system meets the requirements of GDC 2 and 4 with respect to the need for protection against natural phenomena and internal missile protection as its failure does not affect safety system functions. The pressurizer relief tank meets the guidelines of RG 1.29, Positions C.2 and C.3, concerning seismic classification and is, therefore, acceptable. The pressurizer relief tank meets the acceptance criteria of SRP Section 5.4.11.

5.4.12 Reactor Coolant System High Point Vents

See Section 15.9.1 in this SER for discussion of reactor coolant system high point vents.

5.4.13*

5.4.14*

*The July 1981 Standard Review Plan (NUREG-0800) does not include sections addressing FSAR sections that consist of background or design data used in the review of other sections. The section numbers are retained in this SER to provide continuity and ensure a close correlation between subsequent SER sections and their associated SRP sections.

6 ENGINEERED SAFETY FEATURES

6.1 Engineered Safety Features Materials

The staff concludes that the engineered safety features materials specified are acceptable and meet the requirements of GDC 1, 4, 14, 31, 35, and 41 of Appendix A of 10 CFR 50; Appendix B of 10 CFR 50; and 10 CFR 50.55a. This conclusion is based on the following:

(1) GDC 1, 14, and 31, and 10 CFR 50.55a have been met with respect to ensuring an extremely low probability of leakage, of rapidly propagating failure, and of gross rupture. This compliance has been shown because the materials selected for the engineered safety features (ESFs) satisfy Appendix I of Section III of the ASME Code, and Parts A, B, and C of Section II of the Code, and the staff position that the yield strength of cold-worked stainless steels shall be less than 90,000 psi.

In this time frame, the Code allowed waiving of impact testing of Class 2 and 3 components. However, on the basis of the results of impact testing of the same specification steels by other applicants and correlations of the metallurgical characterization of these steels with the fracture toughness data presented in NUREG-0577, the staff concludes that the fracture toughness properties of the ferritic materials in the ESFs have adequate margins against the possibility of nonductile behavior and rapidly propagating fracture.

The controls on the use and fabrication of the austenitic stainless steel of the systems satisfy the requirements of RG 1.31, "Control of Ferrite Content of Stainless Steel Weld Metal," and RG 1.44, "Control of the Use of Sensitized Stainless Steel." The alternate approaches taken by the applicant have been and are acceptable to the staff (see Section 4.5.1). Fabrication and heat treatment practices performed accordingly provide assurance that the probability of stress corrosion cracking will be reduced during the postulated accident time interval.

Conformance with the industry codes and regulatory guides and with the staff positions mentioned above constitutes an acceptable basis for meeting the requirements of GDC 1, 4, 14, 35, and 41 and Appendix B to 10 CFR 50 to which the systems are to be designed, fabricated, and erected so that they can perform their function as required.

(2) GDC 1, 14, and 31 and Appendix B to 10 CFR 50 have been met with respect to ensuring that the reactor coolant pressure boundary and associated auxiliary systems have an extremely low probability of leakage, rapidly propagating failure, and gross rupture. The controls placed on concentrations of leachable impurities in nonmetallic thermal insulation used on components of the engineered safety features are in accordance with the recommendations of RG 1.36, "Nonmetallic Thermal Insulation for Austenitic Stainless Steels," or the applicant's alternative approaches are acceptable

to the staff as discussed in Section 5.2.3. Compliance with the recommendations of RG 1.36 forms a basis for meeting the requirements of GDC 1, 14, and 31.

Protective coating systems are discussed in Section 6.1.2.

(3) The requirements of GDC 4, 35, and 41 and Appendix B to 10 CFR 50 have been met with respect to the compatibility of ESF components with environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents.

The controls on the pH and chemistry of the reactor containment sprays and the emergency core cooling water following a loss-of-coolant or design-basis accident are adequate to reduce the probability of stress corrosion cracking of austenitic stainless steel components and welds of the ESF systems in containment throughout the duration of the postulated accident to completion of cleanup.

Also, the controls of the pH of the sprays and cooling water, in conjunction with controls on selection of containment materials, are in accordance with RG 1.7, "Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident," and provide assurance that the sprays and cooling water will not give rise to excessive hydrogen gas evolution resulting from corrosion of containment metal or cause serious deterioration of the materials in containment.

The controls placed on component and system cleaning are in accordance with RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," and provide a basis for the finding that the components and systems have been protected against damage or deterioration by contaminants as stated in the cleaning requirements of Appendix B, 10 CFR 50.

6.1.1 Postaccident Emergency Cooling Water Chemistry

The postaccident emergency cooling water chemistry has been reviewed in accordance with SRP Section 6.1.1 (NUREG-0800). This section of the FSAR was reviewed through Amendment 7.

This review is related to providing and maintaining the proper pH of the containment sump water and recirculated containment spray water following a design-basis accident to reduce the likelihood of stress corrosion cracking of austenitic stainless steel.

The applicant will use borated water (with a concentration of 4,000 ppm boron) from the refueling water storage tank during the initial injection phase of containment spray. The borated water will be mixed with a 1.35%-2% by weight sodium hydroxide solution from the chemical addition tank.

The resulting solution will have a pH greater than 7 and will drain to the containment sump. Mixing is achieved as the solution is continuously recirculated from the sump to the containment spray nozzles during the recirculation phase of containment spray.

The staff evaluated the pH of the water (mixture of water from the refueling water storage tank and sodium hydroxide solution) in the containment sump. The staff verified by independent calculations that sufficient sodium hydroxide is available to raise the containment sump water pH above the minimum 7.0 level to reduce the probability of stress corrosion cracking of austenitic stainless steel components. The removal effectiveness of the chemical additive for fission products in containment is reviewed in Section 6.5.2. The staff will review the surveillance requirements in the plant Technical Specifications to verify that sufficient sodium hydroxide is maintained in the containment spray additive tank.

On the basis of this evaluation, the staff concludes that the postaccident emergency cooling water chemistry meets the minimum pH acceptance criterion of SRP Section 6.1.1, the positions of BTP MTEB 6-1, and the requirements of GDC 14 and is, therefore, acceptable.

6.1.2 Organic Materials

The description of organic material and protective coating systems inside containment has been reviewed in accordance with SRP Section 6.1.2 (NUREG-0800). This section of the FSAR was reviewed through Amendment 7.

This evaluation is conducted to verify that protective coatings applied inside containment meet the testing requirements of American National Standards Institute (ANSI) N101.2, "Protective Coatings (Paints) for Light Water Nuclear Reactor Containment Facilities," 1972, and the quality assurance guidelines of RG 1.54.

Compliance with these requirements provides assurance that the protective coatings will not fail under design-basis-accident (DBA) conditions and will not generate significant quantities of solid debris or combustible gas that could adversely affect the engineered safety features.

The applicant referenced Westinghouse Electric Corporation's alternative to RG 1.54 for the protective coatings on nuclear steam supply system equipment inside containment. This information was documented in NS-CE-1352 dated February 1, 1977, and was accepted by the staff by letter dated April 27, 1977, from C. J. Heltemes to C. Eicheldinger.

For the balance of plant inside containment, the applicant has committed to use protective coating systems that meet the testing requirements of ANSI N101.2 (1972) and conform to the quality assurance guidelines of RG 1.54. The organic materials under DBA conditions are reviewed in Section 6.2.5.

On the basis of its evaluation, the staff concludes that the protective coating systems and their applications are acceptable and meet the requirements of Appendix B to 10 CFR 50. This conclusion is based on the applicant having met the quality assurance requirements of Appendix B to 10 CFR 50, since the coating systems and their applications meet the positions of RG 1.54 and the quality assurance standards of ANSI N101.2. Also, the containment coating systems have been evaluated as to their suitability to withstand a postulated DBA environment. The coating systems chosen by the applicant have been qualified under conditions that take into account the postulated DBA conditions.

The control of combustible gases that can potentially be generated from the organic materials and from qualified and unqualified paints is reviewed under Section 6.2.5. The consequences of solid debris that can potentially be formed from unqualified paints are reviewed under Section 6.2.2.

6.2 Containment Systems

The Millstone Unit 3 containment systems include the containment heat removal systems, the containment isolation system, and the containment combustible gas control system. The containment and the containment systems function to prevent or control the release of radioactive fission products that might be released into the containment atmosphere following a postulated loss-of-coolant accident (LOCA), secondary system pipe rupture, or fuel-handling accident.

The staff has reviewed the information provided in the FSAR relating to the design, design bases, and safety analyses for the containment and the containment systems. The acceptance criteria used as the basis for its evaluation are contained in SRP Sections 6.2.1, 6.2.2, 6.2.3, 6.2.4, 6.2.5, and 6.2.6, (NUREG-0800). These acceptance criteria include the applicable GDC, RGs, branch technical positions, and industry codes and standards as specified in the above-cited sections of the SRP.

6.2.1 Containment Functional Design

6.2.1.1 Containment Structure

The Millstone Unit 3 containment structure is a carbon-steel-lined, reinforced concrete structure with a net free volume of about 2,260,000 ft^3. The containment structure houses the nuclear steam supply system, including the reactor, steam generators, reactor coolant pumps, and pressurizer, as well as certain components of the auxiliary and engineered safety features systems. The containment structure is designed for a maximum and minimum internal pressure of 45 psig and 8 psia, respectively, and a temperature of 280°F.

For Millstone Unit 3 containment design, the subatmospheric containment concept is used. During normal operation, the containment structure will be maintained at a subatmospheric pressure of approximately 9 to 12 psia. In the event of a high-energy-line-break accident, the containment would be depressurized and a subatmospheric condition reestablished within 60 min; this condition would be maintained for at least 30 days following an accident.

Containment Pressure and Temperature Analyses

The applicant has performed containment response analyses for a spectrum of reactor coolant system and secondary system pipe ruptures to verify the acceptability of the containment design with regard to pressure, temperature, and depressurization criteria and to establish the pressure and temperature conditions for environmental qualification of safety-related equipment located inside containment. The containment functional analyses include the peak containment pressure analysis, the containment depressurization analysis, and the peak containment temperature analysis.

The applicant performed all containment pressure and temperature analyses using the LOCTIC computer code. A temperature flash model, which assumes the blowdown is instantaneously mixed with the containment atmosphere, was used for all LOCA analyses and for main steamline break (MSLB) analyses resulting in a pure-steam blowdown. A pressure flash model, which conservatively assumes that only the steam portion of a two-phase blowdown is added to the atmosphere, was used for MSLBs resulting in a two-phase blowdown. Initial conditions and input data, including passive and active heat removal parameters, were conservatively chosen to produce the highest containment pressures and temperatures.

The LOCAs (reactor coolant system pipe breaks) analyzed by the applicant include double-ended guillotine breaks in the hot leg, the cold leg at the reactor coolant pump suction, and the cold leg at the pump discharge, and a 6-ft^2 double-ended break and a 3-ft^2 split break in the pump suction line. For the double-ended breaks in the hot leg and in the pump suction line, both minimum and maximum emergency core cooling system (ECCS) flow cases were considered. A single failure analysis is not necessary for the peak containment pressure evaluation because the peak pressure for each break case analyzed occurs early in the transient before active ESF systems can influence the results. The design-basis break for containment pressure was determined to be the double-ended guillotine break in the hot leg. The peak containment pressure calculated by the applicant was 39.4 psig, which is below the containment design pressure of 45 psig. The initial containment conditions that yield the highest peak calculated containment pressure are the maximum air partial pressure (10.65 psia), maximum temperature (120°F), and maximum relative humidity (100%). The staff has performed a confirmatory analysis of this design-basis accident using the CONTEMPT-4, Mod 5 computer code. The results of the staff's analysis are in agreement with the applicant's results.

Pump suction breaks yield the highest energy flow rates during the post-blowdown period and, therefore, represent the worst break for long-term containment depressurization. The applicant has performed containment depressurization analyses for the double-ended guillotine break and the 6-ft^2 double-ended break in the pump suction line. A sensitivity analysis for initial containment conditions, single failure, and assumed containment depressurization time for mass and energy release data calculations was also included. For the single-failure analysis the diesel generator failure was assumed because it bounds all other single failures. The 6-ft^2 double-ended pump suction break with minimum ESFs (diesel generator failure resulting in loss of one charging pump, one safety injection pump, one residual heat removal (RHR) pump, one quench spray pump and two recirculation pumps with associated coolers) resulted in the maximum depressurization time and is considered the design-basis accident for the containment depressurization system. The applicant calculated a maximum depressurization time of 3,350 sec and a maximum pressure (following depressurization) of -0.31 psig. The initial containment conditions that result in the maximum depressurization time are the maximum containment air partial pressure (9.25 psia), minimum containment temperature (80°F), maximum relative humidity (100%), maximum service water temperature (75°F), and maximum refueling water storage tank (RWST) temperature (50°F). (The allowable containment air partial pressure will be prescribed in the Technical Specifications as a function of service water temperature.) The staff has performed a confirmatory analysis of this design-basis accident using the CONTEMPT-4, Mod 5 computer code. The results of its analysis confirm that the pressure will be reduced to below atmospheric pressure within 1 hour.

The spectrum of secondary system breaks analyzed by the applicant included various sizes of double-ended and split breaks of the main steamline at five different power levels ranging from 0 to 102%. For the double-ended breaks the forward flow area (effective break area) is limited to 1.4 ft^2 by a flow restrictor in the main steamline. Main feedwater line breaks were not included because the break effluent is of a lower specific enthalpy and thus are not as severe as MSLBs. All of the MSLB analyses conservatively assumed the availability of offsite power to maximize heat transfer from the primary coolant system by keeping the reactor coolant pumps operating, thus maximizing the mass and energy release rate to containment. In addition, auxiliary feedwater addition was assumed to continue for the duration of the analyses. Failure of a main steam isolation valve (MSIV) to close and failure of one emergency bus to energize causing loss of one ESF train were considered in the applicant's single-failure analyses, as were a range of initial containment conditions. Redundant valves are provided for automatic isolation of the main feedwater lines. The highest containment temperature was calculated for a double-ended guillotine break at 75% power, with failure of an MSIV and with maximum initial containment pressure, temperature, and humidity. The analyses resulted in a peak temperature of 336°F, assuming that 8% of the condensate formed on the heat sinks is revaporized. This same break, with use of minimum initial air partial pressure and maximum initial temperature, was also found to result in the highest containment pressure produced by an MSLB, 34.5 psig. The staff has performed a confirmatory analysis of the design-basis steamline break for peak containment pressure using the CONTEMPT-4, Mod 5 computer code. The results of its analysis are in agreement with the applicant's results.

The staff has reviewed the spectrum of reactor coolant system and secondary system pipe breaks analyzed by the applicant and the applicant's choice of initial conditions, input parameters, and assumptions and finds them acceptable. Additionally, the staff performed confirmatory analyses on the design-basis reactor coolant system breaks and MSLBs using the CONTEMPT-4 computer code and the applicant's mass and energy release data. On the basis of its review of the applicant's containment pressure and temperature functional analyses and contingent on resolution of those matters regarding mass and energy release data discussed in Sections 6.2.1.3, and 6.2.1.4 the staff concludes that the applicant has satisfactorily demonstrated the adequacy of the containment functional design following a LOCA or MSLB.

Protection Against Damage From External Pressure

The containment structure is designed to withstand the external (differential) pressure load resulting from a postulated inadvertent actuation of the containment quench spray system during normal plant operation. The maximum pressure differential is based on the difference between the maximum barometric pressure and the minimum attainable internal containment pressure. The applicant calculated a minimum internal pressure of 8.07 psia for this postulated event.

The staff has reviewed the applicant's analysis and has found that with one exception the applicant's assumptions regarding initial containment conditions and containment quench spray system operation tend to minimize the containment pressure. The exception is the use of an initial containment temperature representative of normal operation (100°F) rather than the limiting containment temperature (120°F). The staff feels that use of the limiting containment temperature is more appropriate. Furthermore, the barometric pressure assumed in the applicant's analysis was not stated. The staff will require the applicant to provide revised analyses for maximum containment external pressure and for containment structural capacity if necessary. The staff will review the applicant's analyses to confirm that the containment is adequately designed to accommodate the maximum postulated external loading. This will be a confirmatory item and will be discussed in a supplement to the SER.

6.2.1.2 Subcompartment Analyses

Subcompartment analyses are required to determine the acceptability of the design differential pressure loadings on containment internal structures from high-energy-line ruptures. The applicant has performed transient pressure response analyses for the containment interior subcompartments including the pressurizer cubicle, the steam generator cubicle, and the upper reactor cavity. The applicant analyzed a spectrum of pipe breaks to determine the break sizes and locations that result in peak differential loads on each of the walls around each subcompartment. These included a spray line double-ended rupture in the upper pressurizer cubicle, a surge line double-ended rupture in the lower pressurizer cubicle, a 5-ft^2 split break in the lower steam generator cubicle, a 1.6-ft^2 feedwater line limited displacement rupture in the upper steam generator cubicle, and a 0.7-ft^2 cold-leg limited displacement break in the upper reactor cavity. The main steamlines are not routed through any portions of the subcompartments and therefore were not considered in the analyses. The staff has reviewed the break sizes and locations considered by the applicant and concurs in the selection of the design-basis pipe breaks contingent on the acceptability of the mechanically constrained limit on pipe break size for limited displacement ruptures. (See Section 3.6 of the SER.)

The LOCA mass and energy release rate data used in the subcompartment analyses were obtained using the SATAN V code described in Westinghouse Topical Report WCAP-8312A. This version of the code was previously reviewed by the staff and was found acceptable for subcompartment analyses in a letter dated March 12, 1975. For subcompartment analyses, the applicant conservatively applied a factor of 1.1 to the SATAN V mass and energy release rates. The staff finds this acceptable. With regard to the mass and energy release data for the feedwater line break in the steam generator subcompartment, however, the method of calculation and the assumptions used have not been presented in the FSAR. The staff has asked for additional information from the applicant to complete its evaluation. Staff acceptance of the mass and energy release rate data for this pipe break will remain a confirmatory item pending receipt and review of the applicant's response.

The applicant analyzed the pressure response of each subcompartment using the THREED code, which has not been approved by the staff. Furthermore, the staff has reviewed the initial conditions and assumptions used in the subcompartment analyses and found that although the applicant generally assumed initial conditions to maximize differential pressures, no justification has been provided to support the use of an initial relative humidity of 50%. The use of an initial relative humidity of 50% rather than 0% results in a reduction in peak differential pressures of about 0.5 psi. Separate discussions of the pressurizer cubicle, steam generator cubicle, and reactor cavity analyses are presented below.

Pressurizer Cubicle

The pressurizer cubicle is an irregularly shaped subcompartment that encloses the pressurizer. The subcompartment is vented at the top to the upper containment and at the bottom to the lower containment. The only high-energy lines in the subcompartment are the spray line, located in the upper cubicle, and the surge line, located in the lower cubicle.

The applicant used an 8-node model of the containment for the pressurizer cubicle analyses. Using this model, the applicant calculated a maximum pressure differential across the cubicle wall of 7.4 psid for the upper cubicle (spray-line break) and 20.1 psid for the lower cubicle (surge-line break). The applicant has stated that the design pressures for the upper and lower cubicles are 7.7 and 20.5 psid, respectively.

The staff performed a confirmatory subcompartment analysis of the pressurizer cubicle using the COMPARE-MOD1A computer code and the applicant's nodalization scheme, including reported values for flow areas, K-factors, and inertia. A relative humidity of 0% was assumed in the staff analysis. The COMPARE results indicate a maximum pressure differential of 8.0 psid for the upper pressurizer cubicle and 24.7 psid for the lower pressurizer cubicle.

At the request of the staff, the applicant has committed to provide a nodalization sensitivity study for the pressurizer cubicle. The staff will review and confirm the applicant's revised analysis and if neccessary will require the applicant to verify the structural capability of the pressurizer cubicle on the basis of the staff's calculated pressure response or provide additional analyses and/or justification to support the pressurizer cubicle design.

Steam Generator Cubicle

The steam generator cubicle is that area inside the crane wall and below the operating floor that contains the reactor coolant piping, the reactor coolant pump, and the lower portion of the steam generator. A portion of the steam generator cubicle extends above the operating floor. The Millstone plant has four steam generator cubicles which are similar in design. Cubicle B was used for the steam generator subcompartment analysis because the flow areas are smaller and the K-factor and inertia values are larger.

The applicant used a 35-node model of the containment for the steam generator cubicle analysis. Although no sensitivity analysis was provided by the applicant, the staff reviewed the applicant's nodalization scheme and found it acceptable. Using this model, the applicant calculated a maximum pressure differential of 19.4 psid across the lower cubicle wall and 6.8 psid across the cubicle wall above the operating floor. The limiting breaks were a 5-ft^2 split break in the hot leg for the lower cubicle and a 1.6-ft^2 limited displacement break in the feedwater line for the steam generator cubicle above the operating floor. The applicant has stated that the design pressures for the lower steam generator cubicle and the steam generator cubicle above the operating floor are 21.7 and 9.2 psid, respectively.

The staff performed a confirmatory subcompartment analysis of the steam generator cubicle using the COMPARE-MOD1A computer code and the applicant's nodalization scheme, including reported values for flow areas, K-factors, and inertia.

A relative humidity of 0% was assumed in the staff analysis. The COMPARE results indicate a maximum pressure differential of 21.9 psid across the lower cubicle and 8.9 psid across the steam generator cubicle above the operating floor, which the staff concludes acceptably confirms the adequacy of the steam generator cubicle design.

Reactor Cavity

The reactor cavity is a reinforced concrete structure that serves to support the reactor vessel and provide radiation shielding. The applicant has stated that the design of the neutron shield tank and reactor vessel insulation prevents venting downward below the upper reactor cavity. Thus, the applicant considered only pressurization of the upper reactor cavity and the refueling cavity, which is directly above the upper reactor cavity. The staff has requested the applicant to provide additional design information to support the treatment of only the upper reactor cavity.

The applicant analyzed a total of six different nodal configurations inside the reactor cavity. A 16-node model was used as the limiting model because models with a finer mesh yielded equivalent differential pressures. Using this model, the applicant calculated a maximum pressure differential of 70.9 psid across the upper reactor cavity wall and 4.6 psid across the refueling cavity wall. The applicant has stated that the design pressure is 120 psid across the upper reactor cavity wall and 4.2 psid across the refueling cavity wall.

The staff performed a confirmatory subcompartment analysis for the reactor cavity using the COMPARE-MOD1A computer code and the applicant's 16-node representation of the reactor cavity, including reported values for flow areas, K-factors, and inertia. A relative humidity of 0% was assumed in the staff analysis. The COMPARE results indicate a maximum pressure differential of 71.8 psid across the upper reactor cavity wall and 4.8 psid across the refueling cavity wall. The latter differential pressure exceeds the design pressure by 0.6 psi.

The staff will require the applicant to verify the structural capability of the refueling cavity wall on the basis of the staff's calculated pressure response or provide additional analyses and/or justification to support the refueling cavity design. With regard to the matter of asymmetric blowdown loads on primary system components, it is not clear whether the applicant has performed an analysis of the forces and moments acting on major components to establish the design adequacy of the supports. The applicant will be required to provide the peak transient loadings (forces and moments) on the major primary system components resulting from the reactor coolant system break that gives the peak differential loads (see NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary Systems").

Until the applicant verifies the capability of the subcompartment structures to withstand the calculated pressure responses, justifies the effectiveness of the reactor vessel insulation and neutron shield tank in reducing blowdown into the lower reactivity cavity, and calculates the asymmetric (force and moment) loads on the reactor vessel, these issues will remain open. The staff will report the results of its continuing review of these issues in a supplement to the SER.

6.2.1.3 Mass and Energy Release Analysis for Postulated Loss-of-Coolant Accidents

The applicant has calculated the mass and energy release rate data for postulated LOCAs using the generic methodology described in the report attached to a reference letter from Westinghouse (Anderson, April 25, 1979) to NRC. This report is under staff review and has not been approved, pending the receipt of a response to the staff's request for additional information. Alternatively, the applicant can show that the containment response following a LOCA is within design limits using mass and energy release data based on the previously approved methodology described in WCAP-8312-A. Staff acceptance of the methodology used to calculate LOCA mass and energy release data will remain a confirmatory item pending the receipt of additional information.

6.2.1.4 Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures

The applicant has calculated the mass and energy release rate data for postulated MSLB accidents using the Westinghouse MARVEL/TRANFLO codes described in Westinghouse Topical Reports WCAP-8859 and WCAP-8860. The TRANFLO code predicts the breakflow quality, and using these quality data, the MARVEL code calculates the mass and energy release rates to the containment following a postulated MSLB. Several model changes were made by Westinghouse during the course of the staff's review of these two topical reports. The topical reports were subsequently approved by the staff in a letter dated August 22, 1983. The model chages concerning steam generator level and steam superheating have not been applied to the Millstone Unit 3 MSLB analysis. Also, WCAP-8859 addresses the Westinghouse Model D and Model 51 steam generators; whereas Millstone Unit 3 uses the Westinghouse Model F steam generator. The applicability of the MARVEL/TRANFLO method to the Model F steam generator has not been addressed in the FSAR.

The applicant will be required to provide revised mass and energy release analyses using the approved versions of the MARVEL/TRANFLO codes and to assess the impact of these changes on the thermal response of the Millstone Unit 3 containment. The applicant must also discuss and justify the applicability of the MARVEL/TRANFLO codes to Westinghouse Model F steam generators. This matter will remain an open item pending the receipt and review of this information.

The applicant has addressed the concerns of Office of Inspection and Enforcement (IE) Bulletin 80-04 regarding MSLBs with continued feedwater or auxiliary feedwater addition. On the receipt of a safety injection signal, the main feedwater lines automatically isolate. Auxiliary feedwater addition is limited by flow restrictors to 42 lb/sec and is assumed to last for 30 min, by which time auxiliary feedwater flow will have been manually terminated by an operator. The staff finds that the treatment of the main and auxiliary feedwater systems in the MSLB analysis is acceptable and that the concerns of IE Bulletin 80-04 are adequately addressed.

6.2.1.5 Minimum Containment Pressure Analysis for Emergency Core Cooling System Performance Capability Studies

Appendix K to 10 CFR 50 requires that the containment pressure used for evaluating core cooling effectiveness during reactor core reflood shall not exceed a pressure calculated conservatively for this purpose. The calculation must include the effect of operation of all installed containment pressure-reducing systems and processes.

The applicant has performed the containment backpressure calculation using the methods and assumptions described in Appendix A of WCAP-8339, "Westinghouse Emergency Core Cooling System Evaluation Model - Summary". A break spectrum analysis was performed that considered various break sizes, break locations, and Moody discharge coefficients for the double-ended cold-leg guillotine. Mass and energy release rates for these break were calculated using the methods described in FSAR Section 15.6.5 and are evaluated separately in Section 6.3.5 of this SER.

The staff has reviewed the applicant's input parameters used in the minimum containment pressure analysis, including initial containment conditions, containment net free volume, passive heat sinks, heat transfer to passive heat sinks, and containment active heat removal, and it has found them acceptably conservative and in conformance with BTP CSB 6-1, "Minimum Containment Pressure Model for PWR ECCS Performance Evaluation," with one exception. This exception is the nonconservative assumption of an initial containment pressure of 9.5 psia, which is greater than that which may be encountered under limiting normal operating conditions. The staff is unable to form a conclusion on the acceptability of the analysis, however, because the applicant has not provided the mass and energy release data and the calculated containment pressure response. Once this information is received, the staff will report the results of its review in a supplement to the SER. This is considered a confirmatory item.

6.2.1.6 Summary and Conclusions

The staff has evaluated the Millstone Unit 3 containment functional design with respect to the acceptance criteria in SRP Sections 6.2.1, 6.2.1.1.A, 6.2.1.2, 6.2.1.3, 6.2.1.4, and 6.2.1.5 and concludes that GDC 13, 16, 38, and 50 have been met with the following exceptions:

(1) The applicant has not adequately demonstrated the capability of the containment to withstand the maximum external differential pressure. The applicant must supply a revised analysis that conservatively represents initial containment temperature and barometric pressure. This is considered a confirmatory item.

(2) The applicant must supply additional analysis to demonstrate the structural adequacy of the containment subcompartments, including the pressurizer cubicle and the refueling cavity wall, additional justification concerning the effectiveness of the reactor cavity insulation and neutron shield tank to preclude blowdown to the lower reactor cavity, and analyses of the forces and moments on the major reactor coolant system components. This is considered an open item.

(3) The applicant has not adequately justified the methodologies used to compute the mass and energy release rates for postulated LOCAs, and MSLB and feedwater-line-break accidents. Staff acceptance of the applicant's mass and energy release rate data will remain an open item pending receipt and review of additional information requested by the staff.

(4) The applicant has not supplied sufficient information to allow the staff to form a conclusion on the acceptability of the applicant's minimum containment pressure analysis for use in assessing ECCS performance. This is considered a confirmatory item.

6.2.2 Containment Heat Removal Systems

The function of the containment heat removal system (CHRS) is to remove heat from the containment atmosphere to limit, reduce, and maintain at acceptably low levels the containment pressure and temperature following a LOCA or secondary system pipe rupture. In addition to heat removal provided by passive means such as heat transfer to containment walls, structures, and equipment located inside containment, the Millstone design includes active CHRSs. These systems consist of the quench spray system (QSS) and the recirculation spray system (RSS). The containment air coolers are not considered part of the CHRS. The CHRS is designed to depressurize the containment to a subatmospheric condition within 1 hour following a high-energy-line-break accident. For a discussion of the fission product removal function of the CHRS, see SER Section 6.5.

The QSS consists of two redundant 100% capacity trains, each containing a quench spray pump, a chemical injection system, and riser pipes leading to two common 360° quench spray headers. Rated flow to the quench spray headers is approximately 4,000 gpm with one quench spray pump operable and 6,000 gpm with both pumps operable. The QSS is actuated automatically on receipt of the containment depressurization actuation (CDA) signal, and spray flow becomes effective within 64 sec after the signal is received. The CDA signal is initiated by high containment pressure (24.7 psig). The quench spray is terminated when the refueling water storage tank (RWST) reaches a predetermined level.

Each redundant quench spray subsystem draws water independently from the 1.2 million gallon RWST. Sodium hydroxide solution with a concentration of 1.35 to 2.0% by weight is added to the quench spray by direct gravity feed from the chemical addition tank. The chemical addition tank maintains hydrostatic balance with the RWST during injection, and thus the ratio of flows from the chemical addition tank and RWST is constant. The pH of the spray from the quench spray headers into the containment is approximately 8.0. The final pH of the water in the containment pump after a design-basis accident is between 7.0 and 7.5.

The recirculation spray system (RSS) is designed to further enhance the depressurization of the containment and to maintain the containment at subatmospheric pressure in the long term. The RSS consists of two parallel, redundant 100% capacity trains, each containing two containment recirculation pumps with dedicated heat exchangers and riser pipes leading to two common 360° recirculation spray headers. The four redundant 50% capacity recirculation spray subsystems take suction from the containment sump; the recirculation spray water flows through recirculation coolers where it is cooled by the service water. The rated flow for each recirculation pump is about 3,900 gpm.

The RSS pumps are started automatically approximately 4 min after receipt of the containment depressurization actuation signal, and the spray becomes effective approximately 5 min after receipt of the signal. The RSS is switched to the cold-leg recirculation mode of operation approximately 38 to 62 min after receipt of the phase B signal, for maximum and minimum ESF, respectively.

During this mode a portion of the containment recirculation flow is diverted to the low head safety injection lines for use as core injection.

The CHRSs satisfy the provisions of RG 1.26, "Quality Group Classifications for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants," and RG 1.29, "Seismic Design Classifications," and meet or invoke the design, quality assurance, redundancy, power source, and instrumentation and control requirements of engineered safety features. The applicant has also provided single-failure analyses and other information demonstrating the ability of the CHRSs to function following postulated single active failures.

RG 1.82, "Sumps for Emergency Core Cooling and Containment Spray Systems," provides design guidelines to be met for containment sumps that are designed to serve as sources of water for ECCS and the containment spray systems following a LOCA. The guidelines address redundancy, location, and arrangement of sumps and debris screen provisions to ensure adequate pump performance. The staff has reviewed the Millstone Unit 3 sump design against this guidance.

A single containment sump has been provided and is enclosed by a protective screen assembly that has a total screen area of about 150 ft^2. Furthermore, the containment sump is divided at the centerline by fine screening (3/32-in. opening) and vertical bars so that a failure of either half would not adversely affect the other half. The redundant recirculation pump suctions are located in separate halves of the sump. Therefore, even though the single sump design is not in accordance with recommendations of RG 1.82, the staff has concluded that adequate measures have been taken to ensure that the RSS function will not be lost.

The protective screen assembly provides three stages of screening, namely, vertical trash bars, a coarse mesh screen (3/8-in. opening) and a fine mesh screen (3/32-in. opening). The fine mesh screen opening is smaller than the smallest coolant passage gap in the reactor core and smaller than a spray nozzle orifice. The screen assembly rises vertically approximately 5 ft above the containment floor and is arranged so that no single failure could result in the clogging of all suction points of the recirculation spray system. Following a LOCA, the top of the screen assembly would be under about 10 ft of water. System design allows for 50% blockage of the sump screening without loss of function.

The applicant has conducted containment sump model testing at the Alden Research Laboratory using a 1:3.25 scale model of the sump. A range of possible flow distributions, bar rack and screen blockages, water levels, and pump operation combinations were tested to identify undesirable flow patterns. As a result of these tests, a vortex suppression grating is provided to ensure acceptable sump performance at the minimum expected sump water level.

The applicant has calculated the sump water velocity at the fine mesh screens and found it to be about 0.15 ft/sec assuming no screen blockage, and about 0.3 ft/sec assuming 50% blockage. This approach velocity exceeds the value recommended in RG 1.82. In light of this deviation, the staff will require the applicant to evaluate sump screen blockage using an acceptable methodology and considering the types and quantities of insulation that are to be installed to justify the assumption of 50% blockage.

The staff has reviewed the applicant's net positive suction head (NPSH) calculations and finds that, contingent on satisfactory resolution of the matter concerning sump screen blockage, the NPSH available in either the recirculation spray mode or the ECCS cold-leg recirculation mode is adequate. The applicant has complied with the provisions of RG 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Systems," with one exception. RG 1.1 states that containment heat removal systems should be designed so that adequate NPSH is provided to system pumps assuming maximum expected temperatures of pumped fluids and no increase in containment pressure from that present before postulated LOCAs. Instead, the applicant has calculated available NPSH using a saturated sump model (i.e., the containment atmospheric pressure is conservatively assumed to be equal to the vapor pressure of the liquid in the recirculation sumps ensuring that credit is not taken for containment pressurization during the transient). The staff has previously found the saturated sump model conservative (SRP Section 6.2.2, Acceptance Criterion 2) and, therefore, acceptable.

The staff has reviewed the information in the FSAR and in responses to staff requests for additional information concerning the containment heat removal systems to ensure conformance to all of the acceptance criteria in SRP Section 6.2.2. The staff concludes that the containment heat removal systems satisfy the requirements of GDC 38, 39, and 40. The staff also concludes that, subject to resolution of the matter concerning sump screen blockage, the containment sump design satisfies the requirements of RGs 1.1 and 1.82. The staff considers this to be a confirmatory item and will report the results of its review of this issue in a supplement to the SER.

6.2.3 Secondary Containment Functional Design

The secondary containment in the Millstone design consists of the containment enclosure building and the associated supplementary leak collection and release system (SLCRS) provided to mitigate the radiological consequences of postulated accidents.

The enclosure building is a cylindrical structure that completely surrounds the primary containment above the roof of contiguous buildings and is designed as a structural steel framework with metal siding. The enclosure building is structurally supported from the primary containment and incorporates sliding joints and neoprene seals to accommodate building expansion without the loss of building integrity. This design minimizes the effect of primary containment building expansion on the enclosure building pressure following a LOCA because both buildings will expand in a similar manner. Thermal effects of a LOCA on enclosure building response are also negligible because, in the short term, the concrete primary containment structure thermally isolates the exterior surface of the primary containment from the LOCA environment.

The enclosure building is designed as a seismic Category I structure and will remain functional under all applicable loading conditions except for the design-basis tornado loads. During the postulated design-basis tornado, the metal siding will fail; the structural steel framework, however, is designed to remain intact.

The SLCRS consists of two redundant fans and filtration trains with common ductwork. Each train contains a 100% capacity exhaust fan and a 100% capacity filter bank consisting of a moisture separator, electric heater, upstream high efficiency particulate air (HEPA) filter, a charcoal adsorber, and downstream HEPA filter. The SLCRS will collect and filter radioactive fission products that may leak from the primary containment to the enclosure building and surrounding buildings (main steam valve building, engineered safety features building, hydrogen recombiner building, and auxiliary building). The SLCRS is powered from the emergency diesel generators and is designated seismic Category I and QA Category I, Safety Class 3. The SLCRS is not designed to remain functional following a high-energy-line break outside the primary containment.

The SLCRS, not normally in operation, is actuated on receipt of a safety injection signal. The capacity of each redundant train (9,500 ft^3/min) is sufficient to reduce and maintain a pressure of -0.25-in. water gauge throughout the enclosure building and contiguous buildings within 1 min after the accident, assuming wind velocity of 22 mph. The capability of the SCLRS to achieve and maintain a pressure of -0.25-in. water gauge in the enclosure building and contiguous buildings will be verified by test at the preoperational phase and at intervals not greater than 18 months. The applicant will also perform testing in accordance with Appendix J to 10 CFR 50 for each bypass leakage path and will confirm that the combined leakage for all bypass paths is less than 0.01 X La (9 scfh).

The staff has reviewed the information in the FSAR and the responses to staff requests for additional information concerning the SLCRS to ensure conformance with all of the acceptance criteria in SRP Section 6.2.3 and BTP 6-3. The staff concludes that the SLCRS satisfies the requirements of GDC 43; 10 CFR 50, Appendix J; and RGs 1.26 and 1.52. The staff also finds that the SLCRS satisfies the requirements of GDC 16, as prescribed in SRP Section 6.2.3, Section II.3.C, pending verification by the applicant that all openings in the secondary containment are under administrative control and are provided with position indicators and alarm capability in the main control room. This is considered a confirmatory item.

6.2.4 Containment Isolation System

The function of the containment isolation system is to allow the normal or emergency passage of fluids through the containment boundary while preserving the ability of the boundary to prevent or limit the escape of fission products that may result from postulated accidents. In general, for each fluid system penetration, at least two barriers are required between the containment atmosphere or the reactor coolant system and the outside atmosphere so that failure of a single barrier will not prevent isolation of the containment.

Containment isolation for Millstone Unit 3 is accomplished in two phases. The containment isolation phase A (CIA) signal, which shuts all nonessential system lines penetrating the containment, is initiated by any of the following: (1) high containment pressure (1.5 psig), (2) low compensated steamline pressure (3) pressurizer low pressure or (4) manual actuation. The containment isolation phase B (CIB) signal, which isolates the reactor plant component cooling water supply and return lines and opens the containment isolation valves for the containment depressurization systems, is initiated by high containment pressure (10.0 psig) or by manual actuation.

The CIA and CIB signals, the CDA signal, and all other actuation signals with containment isolation functions are summarized in Table 6.1. The applicant has provided documentation demonstrating that each line having automatic containment isolation valves that must be isolated immediately following an accident is isolated by one of the signals listed in Table 6.1. Although the CIB signal is not actuated by diverse parameters, it is acceptable because the only affected lines are considered important to safe shutdown of the plant and the lines can be remote-manually isolated. The staff concludes that adequate diversity has been provided with regard to the monitored parameters that actuate containment isolation.

The staff has reviewed the applicant's containment isolation system design information and has found that in general (1) there are at least two barriers between the atmosphere outside containment and the reactor coolant system or the containment atmosphere on each fluid line penetrating containment, (2) automatic isolation valves are provided in those lines that must be isolated immediately following an accident, (3) each line that must remain open for safety reasons following an accident has at least one valve capable of being remote-manually isolated, (4) each power-operated isolation valve is provided with position indication and a manual control switch in the main control room, and (5) each air- or solenoid-operated isolation valve assumes the position of greater safety in the event of power failure to the valve operator.

The staff has reviewed the applicant's containment isolation provisions to determine conformance with the requirements of GDC 54, 55, 56, and 57. The staff's review has confirmed that the containment isolation system meets all requirements except in the following cases.

GDC 55

The containment isolation system meets the requirements of GDC 55, where applicable, except in cases where remote-manual ioslation valves are used instead of automatic isolation valves and in cases where automatic valves fail "as is" rather than fail closed on loss of power to the valve operators.

The cases where remote manual isolation valves are used instead of automatic isolation valves include the reactor coolant pump seal water injection lines, the safety injection system lines discharging to the reactor, the residual heat removal (RHR) system lines discharging to the reactor, and the postaccident sample lines. The reactor coolant pump seal water lines and the safety injection system lines discharging to the reactor are important to safe shutdown of the plant, and provisions have been made to detect possible leakage from these lines outside containment, thereby allowing the use of remote-manual instead of automatic isolation valves. Each of the RHR system discharge lines contains three check valves inside containment in series with a remote manual valve outside containment. These lines are part of the ECCS and are required to be open following an accident; hence, use of remote-manual valves instead of automatic isolation valves is acceptable. Each of the postaccident sample lines includes an automatic isolation valve inside containment in series with a manual valve outside containment. Both valves in each line are shut during normal operation and under administrative control. The staff finds these isolation provisions acceptable.

The cases where automatic valves fail as is, as opposed to failing closed, include the reactor coolant charging line, the high pressure boron injection line, and the RHR pump suction line. The reactor coolant charging line and the high pressure boron injection line contain a check valve inside containment in series with the automatic isolation valve outside containment; therefore, a single active failure will not result in the loss of both containment isolation barriers. The RHR pump suction line contains an independently powered remote manual valve inside containment in series with a remote manual valve outside containment. Both valves are shut during normal operation and under accident conditions and are under administrative control.

On the basis of its review, the staff finds that the containment isolation provisions for Millstone either meet the requirements of GDC 55, or, for the specific lines discussed above, are acceptable alternatives to the requirements of GDC 55.

GDC 56

The containment isolation system meets the explicit requirements of GDC 56, where applicable, except in cases where remote-manual isolation valves are used instead of automatic isolation valves, in cases where automatic isolation valves fail as is rather than fail closed on loss of power to the valve operators, and in cases where fluid lines penetrating containment do not contain two isolation valves in series.

The cases where remote-manual isolation valves are used instead of automatic isolation valves include the containment leakage monitoring open taps. The containment isolation design for these four 3/8-in.-diameter instrument lines complies with RG 1.11 instead of GDC 56 because of the size of the lines. A normally open, remote-manual motor-operated valve is provided in each line outside containment. All instrument lines are Safety Class 2 up to and including the isolation valves and are sized to restrict leakage to a value that would not significantly affect offsite doses in the event of a line rupture. The staff has reviewed the containment isolation provisions for the containment leakage monitoring instrumentation lines and concludes that they meet the provisions of RG 1.11.

The cases where automatic isolation valves fail as is rather than fail closed include the following lines:

(1) instrument air
(2) containment atmosphere monitor discharge
(3) containment vacuum pump discharge
(4) quench spray pump discharge
(5) containment recirculation pump suction and discharge

The instrument air and containment atmosphere monitor discharge lines have independently powered automatic isolation valves inside containment in series with the automatic isolation valves outside containment; therefore, single active failure will not result in the loss of both containment isolation barriers. The containment vacuum pump discharge line has a remote-manual valve inside containment and a manual valve outside containment, with both valves shut during normal operation and under administrative control. The staff finds these isolation provisions acceptable. The fail as is design of the containment isolation valves in the quench spray pump discharge lines and containment recirculation pump suction and discharge lines is also acceptable because these sytems are part of the ESF systems and are required to be open following an accident.

The cases where system lines penetrating the containment do not contain two isolation valves in series includes the containment recirculation pump suction lines, which contain single isolation valves. The containment leakage monitoring open taps also contain a single isolation valve, but are designated to comply with RG 1.11.

The suction lines from the containment recirculation sumps must be opened following a LOCA to satisfy their postaccident functional requirement, which is to permit long-term cooling of the reactor core and the containment atmosphere. For these lines, a single isolation valve located outside containment is provided. Because these lines do not have isolation valves inside the containment, the piping between the containment wall and the valves is individually encapsulated in stainless steel. This encapsulation is an extension of the containment structure and prevents a rupture in the suction line between the containment wall and the isolation valve from causing a release of fluids to the environment. The staff finds the design of these lines with a single containment isolation valve outside containment acceptable.

The staff finds that the containment isolation provisions for Millstone Unit 3 either meet the requirements of GDC 56, or, for the specific lines described above, are acceptable alternatives to the requirements of GDC 56.

The staff has reviewed information provided by the applicant to demonstrate compliance with the provisions of NUREG-0737, Item II.E.4.2, "Containment Isolation Dependability." As previously described, the applicant has complied with the provisions regarding diversity in parameters sensed for initiation of containment isolation, has considered the functional requirements of all systems penetrating containment, and has made acceptable provisons for isolation of systems not required for mitigation of the consequences of an accident or safe shutdown of the plant. The applicant also made provisions that resetting of a containment isolation signal will not result in the automatic reopening of containment isolation valves. In addition, the applicant has designated all system lines penetrating the containment as essential or nonessential systems and has provided appropriate isolation signals for isolation valves in each line. Therefore, the staff concludes that the applicant has complied with the provisions of NUREG-0737, Item II.E.4.2.

The staff has reviewed the containment isolation provisions for postaccident sampling lines and found that, in all cases (except for the containment leakage monitoring open taps), these lines contain an isolation valve inside containment that is supplied from a Class 1E power supply, and automatically close on a containment isolation phase A signal or on auxiliary feedwater pump startup in the case of steam generator blowdown lines.

The applicant has stated that all containment isolation system components, including valves, controls, piping, and penetrations, are protected from internally or externally generated missiles, water jets, and pipe whip and jet impingement. The staff, therefore, finds that the containment isolation system meets the requirements of GDC 1, 2, and 4. The containment isolation system also meets the provisons of RGs 1.29, and 1.26.

During normal operation the containment isolation valves in the containment purge air subsystem are closed, and the containment is not purged. To permit containment access, the concentration of airborne particulates and iodine is reduced by use of the containment air filtration subsystem, in which air is drawn from the low elevations of containment, passed through a series of filters, and discharged to the upper elevation of containment. The staff has reviewed the applicant's containment isolation system design for conformance to the provisions of BTP CSB 6-4, "Containment Purging During Normal Plant Operation," and found it acceptable.

In summary, the staff has reviewed the information in the applicant's FSAR and in responses to NRC questions concerning the containment isolation system to ensure conformance to all of the acceptance criteria in SRP Section 6.2.4. The staff concludes that the Millstone Unit 3 containment isolation system meets the requirements of GDC 1, 2, 4, 16, 54, 55, 56, and 57, and is, therefore, acceptable.

6.2.5 Combustible Gas Control System

Following a LOCA, hydrogen may accumulate within containment as a result of (1) metal-water reaction between the zirconium fuel cladding and the reactor coolant, (2) radiolytic decomposition of the water in the reactor core and the containment sump, and (3) corrosion of metals by emergency core cooling and containment spray solutions. To monitor and control the buildup of hydrogen within containment, the applicant has provided a hydrogen recombiner system, a hydrogen monitoring system, and a post-LOCA purge system.

The hydrogen recombiner system consists of two redundant, 100% capacity, thermal-type hydrogen recombiners and associated control units located in the recombiner building. Each recombiner train has a capacity of 50 scfm and is designed to seismic Category I design criteria. The recombiner system is supplied from the Class 1E emergency buses and is manually started and operated from a local control panel.

A redundant containment hydrogen monitoring system is provided in the Millstone Unit 3 design. Each train contains stand-alone analyzer and control cabinets and analyzes, monitors, alarms, and trends containment hydrogen concentration. The system has analog output for display, monitoring, recording, and alarming in the main control room. Input is also provided to the plant computer.

In accordance with 10 CFR 50.44 and RG 1.7, "Control of Combustible Gas Concentration in Containment Following a Loss-of-Coolant Accident," a backup containment purge system is available to purge the containment as an aid to cleanup. This purge capability is provided by the containment vacuum system, which has the primary function of reducing and maintaining the containment at atmospheric pressure. The system consists of two vacuum pumps, piping, valves, and instrumentation. The design capacity of the vacuum system is 108 ft^3/min per pump. The containment vacuum system is not an ESF and only portions of the system are safety related.

Hydrogen mixing to prevent excessive stratification and eliminate areas of potential stagnation is provided at Millstone Unit 3 by forced and natural convective flows within the containment. Mixing is also assisted by the containment spray system when it is operating. The containment internal structures are designed to be as open as practical to allow the circulation and mixing mechanisms to function. The staff finds that these mixing mechanisms in conjunction with the relatively open design of the containment structures will ensure a well-mixed atmosphere within containment and will limit the potential for local hydrogen pocketing following a LOCA.

The applicant has analyzed the production and accumulation of hydrogen within containment using the guidelines in RG 1.7 and American Nuclear Society (ANS) 56.1, Draft 6, "Combustible Gas Control". The applicant's analysis shows that a single recombiner started 4 days after the accident is sufficient to limit the hydrogen concentration in containment to below the RG 1.7 lower flammability limit of 4.0 volume percent. Using the COGAP computer code, the staff has performed analyses that support the applicant's conclusion that a single recombiner is sufficient to maintain the hydrogen concentration below 4.0 volume percent. However, the staff analyses indicate that, for initial containment atmosphere conditions that minimize the mass of air in the containment at the time of the accident, the recombiner would need to be started within the first day following an accident.

On the basis of its review of the Millstone Unit 3 combustible gas control system, the staff concludes that the system satisfies the design and performance requirements of 10 CFR 50.44, the provisions of RG 1.7, the requirements of GDC 5, 41, 42, and 43, and the requirements of NUREG-0737, Items II.E.4.1 and II.F.I, Attachment 6. The applicant, however, will be required to provide procedures for actuating the recombiner system that will ensure that the system is actuated in a timely manner for limiting initial containment conditions. The staff will review the adequacy of these procedures and supporting justification as a confirmatory item.

6.2.6 Containment Leakage Testing Program

The Millstone Unit 3 containment design includes the provisions and features required to satisfy the testing requirements of Appendix J to 10 CFR 50. The design of the containment penetrations and isolation valves permits preoperational and periodic leakage rate testing at the pressure specified in Appendix J to 10 CFR 50.

The staff has reviewed the containment leakage testing program contained in the FSAR and in responses to NRC questions and finds that the proposed reactor containment leakage testing program complies with the requirements of Appendix J to 10 CFR 50. However, the applicant does not intend to vent the containment liner weld channels during type A testing so as to prevent the entry of moisture into the channels. The staff finds this arrangement acceptable provided the applicant demonstrates that the weld channel design is compatible with that for the steel liner. The staff will review the information provided by the applicant regarding weld channel design and, if deemed necessary, will require by plant Technical Specifications that the weld channel be vented during the type A test. The staff considers this a confirmatory item.

The applicant has proposed that the test frequency for passive, normally closed butterfly valves used as containment isolation valves in systems connected to the containment atmosphere be in accordance with Appendix J requirements. As a result of the numerous reports on unsatisfactory performance of the resilient seals for the isolation valves in containment purge and vent lines (addressed in IE Circular 77-11, dated September 6, 1977), Generic Issue B-20, "Containment Leakage Due to Seal Deterioration," was established to evaluate the matter and establish an appropriate testing frequency for the isolation valves. As a result of the staff's investigation of this issue, it was recommended that leakage integrity tests be performed on containment isolation valves with resilient material seals in passive purge systems (i.e., those that must be administratively controlled closed during reactor operating modes 1 through 4) at least once every 6 months. Consistent with this recommendation, the staff will require by plant Technical Specifications that the containment isolation valves in the Millstone Unit 3 purge/vent system be tested for leak integrity at least once every 6 months.

On the basis of the above discussion, the staff concludes that the proposed reactor containment leakage testing program is acceptable and complies with the requirements of GDC 52, 53, and 54; Appendix J to 10 CFR 50; and 10 CFR 100. Such compliance provides adequate assurance that containment leaktight integrity can be verified periodically throughout service lifetime on a timely basis to maintain such leakage within the limits of the Technical Specifications. Maintaining containment leakage rates within such limits provides reasonable assurance that, in the event of any radioactivity releases within the containment, the loss of the containment atmosphere through the leak paths will not be in excess of acceptable limits specified for the site.

6.2.7 Fracture Prevention of Containment Pressure Boundary

The staff's safety evaluation review assessed the ferritic materals in the Millstone Unit 3 containment system that constitute the containment pressure boundary to determine if the material fracture toughness is in compliance with the requirements of GDC 51.

GDC 51 requires that under operating, maintenance, testing, and postulated accident conditions (1) the ferritic materials of the containment pressure boundary behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized.

The Millstone Unit 3 containment is a reinforced concrete structure with a thin steel liner on the inside surface, which serves as a leaktight membrane. The ferritic materials of the containment pressure boundary that were considered in the staff's assessment are those that been applied in the fabrication of the equipment hatch, personnel locks, penetrations, and fluid system components, including the valves required to isolate the system. These components are the parts of the containment system that are not backed by concrete and must sustain loads during the performance of the containment function under the conditions cited by GDC 51.

The staff has determined that the fracture toughness requirements contained in ASME Code editions and addenda typical of those used in the design of the Millstone Unit 3 containment may not ensure compliance with GDC 51 for all areas of the containment pressure boundary. The staff has elected to apply in its licensing reviews of ferritic containment pressure boundary materials, the criteria for Class 2 components identified in the Summer 1977 Addenda of Section III of the ASME Code. Because the fracture toughness criteria that have been applied in construction typically differ in Code classification and Code edition and addenda, the staff has chosen the criteria in the Summer 1977 Addenda of Section III of the Code to provide a uniform review, consistent with the safety function of the containment pressure boundary materials. Therefore, the staff reviewed the materials of the components of the Millstone Unit 3 containment pressure boundary according to the fracture toughness requirements of the Summer 1977 Addenda of Section III for Class 2 components.

Considered in the staff's review were components of the containment system that are load bearing and provide a pressure boundary in the performance of the containment function under operating, maintenance, testing, and postulated accident conditions as addressed in GDC 51. These components are the equipment hatch, personnel airlocks, penetrations, and elements of specific containment penetrating systems.

The staff's assessment is based on the metallurgical characterization of these materials and fracture toughness data presented in NUREG-0577, "Potential for Low Fracture Toughness and Lamellar Tearing on PWR Steam Generator and Reactor Coolant Pump Supports," and ASME Code, Section III, Summer 1977 Addenda, Subsection NC.

The metallurgical characterization of these materials with respect to their fracture toughness was developed from a review of how these materials were fabricated and what thermal history they experienced during fabrication. The metallurgical characterization of these materials, when correlated with the data presented in NUREG-0577 and the Summer 1977 Addenda of ASME Code, Section III, provides the technical basis for the staff's evaluation of compliance with the Code requirements.

On the basis of its review of the available fracture toughness data and materials fabrication histories and the use of correlations between metallurgical characteristics and material fracture toughness, the staff concludes, contingent on the receipt of confirmatory information, that the ferritic components in the Millstone Unit 3 containment pressure boundary meet the fracture toughness requirements that are specified for Class 2 components by the 1977 Addenda of Section III of the ASME Code. Compliance with these Code requirements provides reasonable assurance that the Millstone Unit 3 reactor containment pressure boundary will behave in a nonbrittle manner, that the probability of rapidly propagating fracture will be minimized, and that the requirements of GDC 51 are satisfied.

6.3 Emergency Core Cooling System

The emergency core cooling system for Millstone Unit 3 has been reviewed in accordance with SRP Section 6.3 (NUREG-0800). Conformance with the acceptance criteria, except as noted below, formed the basis for concluding that the design of the facility for emergency core cooling is acceptable.

As specified in SRP Section 6.3, Paragraph I.2, the design of the ECCS was reviewed to determine that it is capable of performing all of the functions required by the design bases. The ECCS is designed to provide core cooling as well as additional shutdown capability following accidents that result in significant depressurization of the reactor coolant system (RCS). These accidents include mechanical failure of the RCS piping up to and including the double-ended break of the largest pipe, rupture of a control rod drive, spurious relief valve operation in the primary and secondary fluid systems, and breaks in the main steam piping.

The principal bases for the staff's acceptance of this system are conformance to 10 CFR 50.46 and Appendix K to 10 CFR 50, and GDC 2, 4, 5, 17, 27, 35, 36, and 37.

The applicant stated that the requirements will be met even with minimum engineered safeguards available, such as the loss of one emergency power bus, with offsite power unavailable.

6.3.1 System Design

As specified in SRP Section 6.3, Paragraph I.2, the design of the ECCS is reviewed to determine that it is capable of performing all of the functions required by the design bases. The ECCS design is based on the availability of a minimum of three accumulators, one charging pump, one safety injection pump, one residual heat removal (RHR) pump, and one containment recirculation pump together with associated valves and piping. The ECCS of Millstone Unit 3 is not shared by other nuclear plants. Following a postulated LOCA, passive (accumulators) and active (injection pumps and associated valves) systems will operate. After the water inventory in the refueling water storage tank (RWST) has been depleted, long-term recirculation will be provided by the containment recirculation pump taking suction from the containment sump and discharging to the RCS cold and/or hot legs. The low-pressure passive accumulator system consists of four pressure vessels partially filled with borated water and pressurized with nitrogen gas to approximately 640 psia. Fluid level, boron concentration, and nitrogen pressure can be remotely monitored and adjusted in each tank. When RCS pressure is lower than accumulator tank pressure, borated water is injected through the RCS cold legs.

The high head injection system consists of two centrifugal charging pumps which provide high pressure injection of boric acid solution into the RCS. In addition to the high head charging pump system, two intermediate head safety injection pumps deliver fluid to the RCS. Both high and intermediate head pumps are aligned to take suction from the RWST for the injection phase of their operation. Low head injection is accomplished by two RHR pump subsystems taking suction from the RWST during the short-term ECCS injection phase. For long-term recirculation, the containment recirculation pumps will take suction from the containment sump.

The RWST minimum water inventory is 1,162,800 gal of 2,000-ppm borated water. The refueling water storage tank is a vertical, seismic Category I tank mounted on and secured to a reinforced concrete foundation. The borated water in the RWST is maintained at a maximum temperature of 50°F and a minimum temperature of 40°F by circulating the RWST water through the refueling water cooler, which uses chilled water from the seismic Category I, tornado- and missile-protected chilled water system. The RWST is insulated to limit the temperature rise of the water to 1/2F° or less per 24-hour period whenever the chilled water system is inoperable.

Water temperature in the RWST is indicated in the control room. Four water level indicator channels, which indicate in the control room, are provided. The high and low level alarm are provided to initiate and stop makeup to ensure that a sufficient volume of water is always available in the RWST. The low-low level alarm stops the RHR pumps and alerts the operator to realign the ECCS from the injection to the recirculating mode following an accident. The staff asked the applicant to provide and justify the minimum time available to the operator to complete the switchover to the recirculation mode. In Amendment 8 to the FSAR, the applicant stated that minimum elasped time from a LOCA to the receipt of the RWST low-low level signal has been calculated to be 36 min. This response is acceptable.

As specified in SRP Section 6.3, Paragraph II, the ECCS system is initiated either manually or automatically on (1) low pressurizer pressure, (2) high containment pressure, or (3) low pressure in any main steamline. This meets the requirements of GDC 20.

The ECCS may also be manually actuated, monitored, and controlled from the control room as required by GDC 19. The ECCS is supplemented by instrumentation that will enable the operator to monitor and control the ECCS equipment following a LOCA so that adequate core cooling may be maintained. The acceptability of the proposed ECCS instrumentation and controls is addressed further in Section 7.3.

As specified in SRP Section 6.3, Paragraph III.3, the available net positive suction head (NPSH) for all the pumps in the ECCS (the safety injection, centrifugal charging, and RHR pumps) should be shown to provide adequate margin by calculations performed to meet the safety intent of RG 1.1, "Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps." In response to the staff request for additional information, the applicant, in Amendment 3 to the FSAR, provided data of the required NPSH for each type of ECCS pump. The staff reviewed the information and finds it acceptable.

As required in SRP Section 6.3, Paragraph III.11, the valve arrangement on the ECCS discharge lines was reviewed with respect to adequate isolation between the RCS and the low pressure ECCS.

Isolation of the low pressure portions of the ECCS at the interface with the high pressure RCS is provided by three check valves in series. This arrangement is acceptable.

Test lines are provided for periodic leakage checks of reactor coolant system pressure boundaries. This is discussed further in Section 5.2.5.

Containment isolation features for all ECCS lines including instrument lines, the requirements of GDC 56 and RG 1.11, "Instrument Lines Penetrating Primary Reactor Containment," are discussed in Section 6.2.4.

In response to the staff concern regarding the effects of water hammer that may occur in the ECCS lines, the applicant in a letter dated August 29, 1983, indicated that proper initial fill and venting of the ECCS ensures that water hammer will not occur. In addition, the head of water provided by the RWST further ensures that the lines will remain full. High point vents in the ECCS lines are provided to ensure a means for proper venting of lines and pumps.

Also, the effects of waterhammer have been considered in the design of the ECCS components.

In response to the staff concern regarding the containment sump design and its effect on long-term cooling following a LOCA, the applicant, in Amendment 2 to the FSAR, indicated that the containment sump vortex control was verified by means of a 1:3.25 scale model test. A wide range of possible approach flow distributions, bar rack and screen blockages, water levels, and pump operation combinations were tested to identify undesirable flow patterns. The applicant stated that the test results show that the containment sump hydraulic performance is adequate at water levels above el -22 ft 6 in. without a vortex suppression grating. Since the minimum LOCA water during recirculation pump operation is estimated at el -23 ft 10 in., vortex suppression is required and will be provided for Millstone Unit 3. The applicant also indicated that tests with the vortex suppression grating in place show that the sump performance is acceptable at the minimum estimated sump water level. To ensure an acceptable pressure drop across the fine mesh sump screening during the recirculation mode of operation, the applicant stated that the design velocity through the screens is limited to 0.2 ft/sec. assuming 50% of the available screen area is blocked.

With regard to debris and fallen thermal insulation which could block the trash rack or screen, the applicant indicated that the allowance for 50% plugging or blockage of the sump has been assumed in the design. The 50% blockage assumption is conservative since lighter particles will float on the water surface which will be above the screen assembly. Heavier particles will sink to the containment floor and will not be drawn into the screen because low inlet velocities were used in the design of the sump. The effects of primary coolant sources outside containment, are discussed in Section 5.2.5.

The safety-injection (SI) lines are protected from intersystem leakage by relief valves in both suction header and discharge lines. Intersystem leakage detection for the RHR and safety injection pumps is described in Section 5.2.5.

As specified in SRP Section 6.3, Paragraph II.B, no ECCS components are shared between units, which meets the requirements of GDC 5.

6.3.2 Evaluation of System Operation and Potential Single Failures

As specified in SRP Section 6.3, Paragraph II, the staff has reviewed the system description and piping and instrumentation diagrams to verify that sufficient core cooling will be provided during the initial injection phase with and without offsite power, assuming a single failure. The cold-leg accumulators have a normally open motor-operated isolation valve and two check valves in series in their discharge lines. When the RCS pressure falls below the accumulator pressure, the check valves open and borated water is forced into the RCS. One accumulator is attached to each of the cold legs of the RCS.

During plant startup, the operator is instructed, via operating procedures, to energize and open these valves when the RCS pressure reaches the SI setpoint. Monitor lights in conjunction with an audible alarm will alert the operator should any of these valves be left inadvertently closed once the RCS pressure increases beyond the safety injection unblock setpoint. Power is disconnected after valves are opened. In addition, since only three accumulators are required to mitigate the consequences of LOCA (part of the ECCS design basis), a single failure will not affect the safety function of the system.

Certain SI systems are blocked to preclude unwanted automatic actuation during normal shutdown and startup conditions. Failure to unblock these systems could seriously impair the reactor safety. The staff asked the applicant to describe the alarms available to alert the operator to accidents when certain safety-injection systems are blocked, as well as the operator actions and time frame available for the operator to mitigate such accidents, and the consequences of the accident. In Revision 1 to the FSAR, the applicant stated that manual block features are provided for low pressurizer and low compensated steamline pressure. The same interlock allows steamline isolation on high steamline negative pressure rate. In the event of a steamline rupture while both of these SI actuation signals are blocked, steamline isolation will occur on high negative steam pressure rate. An alarm for steamline isolation will alert the operator to the accident. For a large LOCA, sufficient mass and energy would be released to the containment to automatically initiate SI at high containment pressure. Therefore, a single failure (i.e., failure to unblock the SI system) will not impair the system's safety function. Additionally, a multiplicity of indications, such as rapid decrease of RCS pressure, ECCS valve and pump position indication, status lights and annunciators, are available to the operator at the control board. For a small LOCA, the time to uncover the core is relatively long (i.e., greater than 10 min), and the operator would have sufficient time to manually initiate SI. This response is acceptable.

Power lockouts are provided in the control room for each valve whose spurious movement could result in degraded ECCS performance. The applicant's proposed method for locking out power to valves is discussed Section 7.6.

Three active injection systems are available; each system has two pumps. The pumps in each system are connected to separate power buses and are powered from separate diesel generators in the event of loss of offsite power, as required by GDC 17. Thus, at least one pump in each injection system would be actuated. The high head injection systems contain parallel valves in the suction and discharge lines, thus ensuring operability of one train even in the event that one valve fails to open. The low and intermediate head injection systems are normally aligned so that valve actuation is not required during the injection phase. A single failure of the safety injection system will not preclude the safe shutdown of the reactor.

The staff has expressed concern with regard to excessive boron concentration in the reactor vessel and hot-leg recirculation flushing related to long-term cooling following a LOCA. The applicant indicated in Revision 2 to the FSAR that this concern had been addressed in a letter from C. Caso of Westinghouse to T. Novak of NRC, dated April 1, 1975. This letter presents the method, assumptions, and result of the analysis for a typical four-loop plant at a core power of 3,411 MWt. The applicant stated that the analysis shows that boric acid concentration within the reactor vessel and core region remains at acceptable levels up to the time of hot-leg recirculation. This response is acceptable.

Flooding of ECCS components inside containment following a LOCA has been evaluated. No ECCS LOCA-related instruments or valve operators will be flooded following a postulated accident. All electrically operated valves in the ECCS required to be functional during and following a LOCA are located outside containment. All other electrical equipment in the ECCS that is required after a LOCA is either located outside containment or above the maximum calculated water level inside containment.

In response to the staff's concern about the most limiting single failure assumed by the applicant to evaluate the ECCS performance and for the transients and accidents analyzed in the FSAR, the applicant provided the information in a letter dated August 29, 1983, indicating that the failure of a single ECCS train is the most limiting single failure. The staff reviewed the information and finds it acceptable.

On the basis of staff review of the design features, the staff concludes that the ECCS complies with the single-failure criterion of GDC 35.

6.3.3 Qualification of the Emergency Core Cooling System

The ECCS is designed to seismic Category I requirements. The equipment design quality classification and its compliance with RG 1.26, "Quality Group Classification and Standard for Water, Steam and Radioactive Waste Containing Components of Nuclear Power Plants," are discussed in Section 3.10.

The ECCS protection against missiles inside and outside containment by the design of suitable reinforced concrete barriers, which includes reinforced concrete walls and slabs (conformance to GDC 4), is discussed in Section 3.5.2.

The protection of the ECCS from pipe whip inside and outside of containment is discussed in Section 3.6.

The active components of the ECCS, designed to function under the most severe duty loads including safe-shutdown-earthquake, are discussed in Section 6.3.2. The ECCS design to permit periodic inspection in accordance with ASME Code, Section XI, which constitutes compliance with GDC 36, is discussed in Section 6.3.4. This meets the intent of SRP Section 6.3, Paragraph III.23.c.

The ECCS incorporates two subsystems which serve other functions. The RHRS provides for decay heat removal during reactor shutdown. At other times the RHRS is aligned for ECCS operation. The centrifugal charging pumps are utilized as part of the chemical and volume control system (CVCS) designed to maintain the required volume and water chemistry of primary fluid in the RCS. On an ECCS actuation signal, the system is aligned to ECCS operation and the CVCS function is isolated. The dual function of the RHRS and the centrifugal charging system does not affect its capability to function as an integral portion of the ECCS.

6.3.4 Testing

The applicant has committed to demonstrate the operability of the ECCS by subjecting all components to preoperational and periodic testing, in conformance with RG 1.68, "Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors," and 1.79, "Preoperational Testing of Emergency Core Cooling System for Pressurized Water Reactors," and GDC 37.

6.3.4.1 Preoperational Tests

One of the preoperational tests is to verify system actuation; namely, the operability of all ECCS valves initiated by the safety injection signal, the operability of all safeguard pump circuitry down through the pump breaker control circuits, and the proper operation of all valve interlocks.

Another preoperational test is to check the cold-leg accumulator system and injection line to verify that the lines are free of obstructions and that the accumulator check valves and isolation valves operate correctly. The applicant will perform a low pressure blowdown of each accumulator to confirm the line is clear and check the operation of the check valves.

The applicant will use the results of the preoperational tests to evaluate the hydraulic and mechanical performance of ECCS pumps delivering through the flow paths for emergency core cooling. The pumps will be operated under both miniflow (through test lines) and full-flow (through the actual piping) conditions.

The applicant has indicated his intent to comply with the criteria of RG 1.79 and GDC 37 that cover testing of the ECCS.

On the basis of the review of the test programs discussed above, the staff concludes that the ECCS test program for Millstone Unit 3 is acceptable.

6.3.4.2 Periodic Component Tests

The ECCS components and all necessary support systems will be routinely and periodically tested. Valves that actuate after a LOCA are operated through a complete cycle. Pumps are operated individually in this test on their miniflow lines, except for the charging pumps which are tested by their normal charging function. The applicant has stated that these tests will be performed in accordance with ASME Code, Section XI.

6.3.5 Performance Evaluation

The ECCS has been designed to deliver fluid to the RCS to limit the fuel cladding temperature following transients and accidents that require ECCS actuation. The ECCS is also designed to remove the decay and sensible heat during the recirculation mode. 10 CFR 50.46 lists the acceptance criteria for an ECCS. These criteria include the following:

(1) The calculated maximum fuel cladding temperature does not exceed 2200°F.

(2) The calculated total oxidation of the cladding does not exceed 0.17 times the total cladding thickness before oxidation.

(3) The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 0.01 times the hypothetical amount that would be generated if all the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

(4) Calculated changes in core geometry are such that the core remains amenable to cooling.

(5) After any calculated successful initial operation of the ECCS, the calculated core temperature is maintained at an acceptable low value and decay heat is removed for the extended period of time required by the long-lived radioactivity remaining in the core.

In addition, 10 CFR 50.46(a)(1) states: "ECCS cooling performance shall be calculated in accordance with an acceptable evaluation model, and shall be calculated for a number of postulated loss-of-coolant accidents. Appendix K [to 10 CFR 50], ECCS Evaluation Models, sets forth certain required and acceptable features of evaluation models."

6.3.5.1 Large-Break LOCA

The applicant has examined a spectrum of large breaks in RCS piping and these analyses indicate that the most limiting event is a cold-leg guillotine break with a discharge coefficient of 0.6. The applicant stated that the analysis took credit for only one train of active ECCS components and three of the four accumulators. In the large-break analysis the worst-case break was assumed, which resulted in decreasing RCS pressure. Depressurization of the RCS results in a pressure decrease in the pressurizer. The reactor trip signal subsequently occurs when the pressurizer low pressure trip setpoint is reached. A safety-injection signal is generated when the pressurizer pressure reaches the ECCS setpoint. The analysis results demonstrated that adequate core cooling is provided assuming the worst single failure with no credit taken for nonsafety-related equipment.

The large-break LOCA evaluation model utilized in this analysis is described in WCAP-9220-P-A and WCAP-9221-P-A. This model was approved by NRC (memorandum from T. P. Speis (NRC) to R. L. Tedesco dated November 2, 1981) and is used in large-break LOCA analyses for Westinghouse plants.

Containment parameters are chosen to minimize containment pressure so that core reflood calculations are conservative. Fuel rod initial conditions are chosen to maximize cladding temperature and oxidation. Calculations of core geometry are carried out past the point where temperatures are decreasing.

The most limiting break with respect to peak cladding temperature is the double-ended guillotine break in the reactor coolant pump discharge piping. The peak cladding temperature is 1960°F, which is below the 2200°F limit.

The total core metal-water reaction is less than 0.3% for all breaks, as compared with the 1.0% conclusion of 10 CFR 50.46. The maximum local oxidation is

8.67% which is well below the embrittlement limit of 17% required by 10 CFR 50.46.

6.3.5.2 Small-Break LOCA

The LOCA sensitivity studies determined the limiting small break to be less than a 10-in.-diameter rupture of the RCS cold leg. A range of small-break analyses were presented which established the limiting break size. Analysis of this break has shown that the high head portion of the ECCS, together with accumulators, provides sufficient core flooding to keep the calculated peak cladding temperature less than that calculated for a large break and below the limits of 10 CFR 50.46.

The applicant has analyzed a spectrum of small-break LOCAs (3-in., 4-in., and 6-in). With regard to peak cladding temperature and metal-water reaction, the analyses identify that the 4-in. break is the limiting small break, the calculated peak cladding temperature is 1485°F, and the total metal-water reaction is less than 3%.

The applicant has analyzed the performance of the ECCS in accordance with the criteria set forth in 10 CFR 50.46 and Appendix K to 10 CFR 50. The staff has reviewed the applicant's evaluation, and concludes that it is acceptable and meets the criteria of 10 CFR 50.46.

6.3.5.3 Conclusions

The ECCS includes the piping, valves, pumps, heat exchangers, instrumentation, and controls used to transport heat from the reactor core following a LOCA. The scope of review of the ECCS for the Millstone Unit 3 plant included piping and instrumentation diagrams, equipment layout drawings, failure modes and effects analyses, and design specifications for essential components. The staff review has included the applicant's proposed design criteria and design bases for the ECCS and the manner in which the design conforms to these criteria and bases.

The staff concludes that the design of the ECCS is acceptable and meets the requirements of GDC 2, 5, 17, 27, 35, 36, and 37. This conclusion is based on the following:

(1) The applicant has met the requirements of GDC 2 with regard to the seismic design of nonsafety portions thereof which could have an adverse effect on ECCS by meeting Position C.2 of RG 1.29.

(2) The applicant has met the requirements of GDC 5 with respect to sharing of structures, systems, and components by demonstrating that such sharing does not significantly impair the ability of the ECCS to perform its safety function, including, in the event of an accident to one unit, an orderly shutdown and cooldown of the remaining units.

(3) The applicant has met the requirements of GDC 17 with respect to providing sufficient capacity and capability to ensure that (a) specified acceptable fuel design limits and design conditions of the reactor coolant pressure

boundary are not exceeded as a result of anticipated operational occurrences and (b) the core is cooled and vital functions are maintained in the event of postulated accidents.

(4) The applicant has met the requirements of GDC 27 with regard to providing combined radioactivity control system capability to ensure that under postulated accident conditions and with appropriate margin for stuck rods, the capability to cool the core is maintained and the applicant's design meets the guidelines of RG 1.47.

(5) The applicant has met the requirements of GDC 35 to provide abundant cooling capability for ECC by providing redundant safety-grade systems that meet the recommendations of RG 1.1.

(6) The applicant has met the requirements of GDC 36 with respect to the design of the ECCS to permit appropriate periodic inspection of important components of the system.

(7) The applicant has met the requirements of GDC 37 with respect to designing the ECCS to permit testing of the operability of the system throughout the life of the plant, including the full operational sequence that brings the system into operation.

(8) The applicant has provided an analysis of the ECCS using an acceptable evaluation model to demonstrate that the criteria of 10 CFR 50.46 are met.

6.4 Control Room Habitability

The requirements for the protection of the control room personnel under accident conditions are specified in GDC 19. The applicant proposes to meet these requirements by incorporating shielding and emergency ventilation systems in the control room design and by having an adequate supply of self-contained breathing apparatus available in the control room for the emergency team. The applicant states in the FSAR that the emergency ventilation system is redundant and that the testing and maintenance of the system is in accordance with RG 1.52. The staff's review of the conformance to this RG is given in Section 6.5.1 of this report. The staff's review of the control room habitability was performed in accordance with SRP Section 6.4 (NUREG-0800) and RGs 1.78 and 1.95.

The Millstone Unit 3 control room heating, ventilation, and air conditioning system is designed to automatically isolate on detection of high radiation or chlorine in the outside air intake, a safety injection signal, or a chlorine detector failure. The control room can also be isolated manually. The control room is to be automatically pressurized 60 sec after isolation with air from the pressurization air storage tanks, which have a 1-hour supply. After 1 hour, an emergency filter train is manually started to maintain pressurization of the control room. The emergency filter train is rated at a flow of 1,000 ft^3/min of which up to 230 ft^3/min is outside air for pressurization and at least 770 ft^3/min is recirculated air. The filter efficiency is addressed in Section 6.5.1 of this report.

The staff has evaluated the habitability of the control room with respect to toxic gases. As indicated in FSAR Table 6.4-2, 55 tons (per unit) of chlorine are proposed to be stored 434.3 m from the nearest control room intake. The applicant has demonstrated in the analysis that the control room habitability systems will adequately protect the operators against an accidental chlorine release in accordance with SRP Section 6.4 and RGs 1.78 and 1.95. In view of a potential accident involving a chlorine release, the staff will require periodic testing in the Technical Specifications to ensure control room leak-tightness and operability of isolation and pressurization systems, including the following:

(1) the pressurization test (using the engineered safety features emergency filter train) to show that the control room leak rate is equal to or less than 230 ft^3/min at equal to or greater than 1/8-in. water gauge pressure differential across adjacent areas

(2) the response time of the control room isolation system is equal to or less than 6 sec

(3) the response time of the bottled air pressurization system is approximately 60 sec

(4) pressurization tests to show the pressure differential during bottled air pressurization is equal to or greater than 1/8-in. water gauge across adjacent areas

The periodic testing in the Technical Specifications would provide assurance that the design-basis parameters used in the analysis are maintained throughout the life of the plant.

The staff has evaluated the control room doses following radiation release design-basis accidents in accordance with SRP Section 6.4. The calculated whole-body and thyroid doses are within the guidelines of SRP Section 6.4.

On the basis of the foregoing, the applicant has demonstrated that the control room habitability system will adequately protect the control room operators in accordance with the requirements of NUREG-0737, Item III.D.3.4, and GDC 19. Until this matter is resolved the control room habitability remains an open item.

6.5 Engineered Safety Feature Atmosphere Cleanup Systems

6.5.1 System Description and Evaluation

Section 6.5 of the FSAR contains information pertaining to engineered safety feature (ESF) atmosphere cleanup systems, their design bases, and applicable acceptance criteria.

The staff has reviewed the applicant's design, design criteria, and design bases for the ESF atmosphere cleanup systems for Millstone Unit 3. The acceptance criteria used as the basis for its evaluation are in Section II of SRP Section 6.5.1 (NUREG-0800). These acceptance criteria include the applicable GDC, ANSI N509-1980, ANSI N510-1980, RG 1.52, and other documents identified in Section II of the SRP. Conformance to the acceptance criteria provides the bases for concluding that the ESF atmosphere cleanup systems meet the requirements of 10 CFR 50.

The ESF atmosphere cleanup system at Millstone Unit 3 consists of process equipment and instrumentation necessary to control the release of radioactive iodine and particulate material following a design-basis accident (DBA). At Millstone Unit 3, the following four filtration systems have been designed for this purpose:

(1) control room emergency ventilation system described in FSAR Section 9.4.1

(2) fuel building exhaust system described in FSAR Section 9.4.2

(3) charging pump, component cooling water pump, and heat exchangers exhaust ventilation system described in FSAR Section 9.4.3

(4) supplementary leak collection and release system described in FSAR Section 6.2.3

Each of these systems was reviewed in accordance with the SRP. The results of these reviews are discussed below.

(1) <u>Control Room Emergency Ventilation System</u>

The control room emergency ventilation system (CREVS) consists of two 100% capacity filtration systems, with each system designed to filter up to 1,000 ft^3/min of air. Each filtration system includes, in order, a demister, an electric heating coil, a high-efficiency particulate air (HEPA) filter, a 4-in.-deep charcoal adsorber, and another HEPA filter. The purpose of the CREVS is to limit the amount of radioactivity introduced into the control room following an accident and filter radioactivity already in the control room so that doses to control room operators will be within the design crit ion of GDC 19. On receipt of a safety injection signal (SIS), a toxic gas concentration signal, or high radiation signal at the outside air intakes, the outside intake and exhaust isolation valves will be automatically closed. During the first hour following closure of the outside intake and exhaust isolation valves, the control room pressure envelope is pressurized from one of two banks of air. After 1 hour, the isolation valves are opened and 1,000 ft^3/min outside air is brought into the control room through the control room emergency ventilation system (CREVS).

The staff has credited the system with 99% removal efficiency for all forms of radioiodines. Each system is provided with a dedicated fan, and, therefore, no bypass leakage around the filter bank is expected. FSAR Sections 6.5.1 and 9.4.1 contain a detailed description of the CREVS.

(2) <u>Fuel Building Exhaust System</u>

The fuel building exhaust system (FBES) consists of two 100% capacity filtration systems with each designed to filter up to 30,000 ft^3/min of air. Each filtration system includes a demister, an electric heating coil, a HEPA filter, a 4-in.-deep charcoal adsorber, and another HEPA filter. The purpose of the FBES is to maintain the fuel storage building at a negative pressure so that any radioiodines or particulates released to the building will be contained within the building and then filtered before release.

The staff has credited the system with 99% removal efficiency for all forms of radioiodine. Each train is provided with a dedicated fan, and, therefore, no bypass leakage around the filter bank is expected. FSAR Sections 6.5.1 and 9.4.2 contain a detailed description of the FBES.

(3) Charging Pump, Component Cooling Water Pump, and Heat Exchangers Exhaust Ventilation System

The charging pump, component cooling water pump, and heat exchangers exhaust ventilation system (CCHVS) consists of two 100% capacity filtration systems with each system designed to filter up to 30,000 ft^3/min of air. Each system consists of the same components as the FBES, including a 4-in.-deep charcoal adsorber. The purpose of the CCHVS is to produce an airflow direction from the general areas of the auxiliary building into component cooling pump and heat exchanger areas in the event of a DBA to prevent areas of low radiation from being affected by airflow from areas of high radiation. On the receipt of a high radiation signal from the radiation monitors at various points in the exhaust air duct stream of the auxiliary building ventilation system, or an SIS signal, exhaust air can be manually diverted through one or both CCHVSs.

The staff has credited the system with 99% removal efficiency for all forms of radioiodines. Each train is provided with a dedicated fan, and, therefore, no bypass around the filter bank is expected. FSAR Sections 6.5.1 and 9.4.3 contain a detailed description of the CCHVS.

(4) Supplementary Leak Collection and Release System

The supplementary leak collection and release system (SLCRS) consists of two 100% capacity filtration systems with each system designed to filter up to 9,500 ft^3/min of air. Each system consists of the same components as the CCHVS, including a 4-in.-deep charcoal adsorber. The SLCRS is designed to maintain a negative pressure in the containment enclosure building and associated contiguous structures during a DBA. The system automatically starts on the receipt of an SIS signal.

The staff has credited the system with 99% removal efficiency for all forms of radioiodines. Each train is provided with a dedicated fan; therefore, no bypass leakage around the filter bank is expected. FSAR Sections 6.5.1 and 6.3 contain a detailed description of the SLCRS.

The ESF filtration systems were reviewed according to SRP Section 6.5.1 (NUREG-0800) and RG 1.52, Revision 1. RG 1.52, Revision 2, was not in existence at the time the Millstone Unit 3 ESF filtration systems were designed, nor when the equipment was purchased. Therefore, the review of the ESF filtration systems was conducted using Revision 1 of RG 1.52, which more adequately reflects the criteria that were in effect at the time the Millstone Unit 3 ESF filtration systems were designed and purchased.

The applicant has provided a comparison of the design of the Millstone Unit 3 ESF filtration systems with the regulatory positions of RG 1.52, Revision 2, in FSAR Tables 1.8-1 and 6.5-1. The staff has determined that the applicant has proposed two significant exceptions to RG 1.52, Revisions 1 and 2, and that the other remaining exceptions are trivial in nature and are acceptable.

SRP Section 6.5.1, Revisions 1 and 2, and RG 1.52, Revisions 0, 1, and 2, call for each ESF atmosphere cleanup system to be instrumented to signal, alarm, and record pressure drop and flow rate at the control room. Millstone Unit 3 ESF atmosphere cleanup systems are provided with only local pressure differential indicators across each filter with a common alarm in the control room for high-pressure differential across the filter bank. The pressure differentials across each filter are transmitted to the plant computer in the main control room. The applicant stated in a letter dated July 13, 1984, that (1) the system flow rate and pressure drop will be verified at least once every 18 months and (2) the system fans are fixed-speed fans and the system flow rates against pressure drops will be verified during plant operation using the certified fan curves. The staff will require verification of the system flow rate versus pressure drop during plant operation on a routine basis in forthcoming Millstone Unit 3 Technical Specifications. With this requirement in the Technical Specification, the staff finds this exception acceptable.

In FSAR Table 6.5-1, the applicant has taken an exception to RG 1.52, Revision 2, and SRP Section 6.5.1, Revisions 1 and 2, in that the fuel building exhaust system activation is manual while the SRP and the regulatory guide require that the system should be automatically activated when a DBA occurs. The applicant states that the system will be continuously operated during refueling, fuel handlings, and storage of spent fuels that have decayed less than 60 days in accordance with NUREG-0452, Revision 4. This requirement was not a regulatory position in RG 1.52, Revision 1, and, therefore, the staff finds this deviation acceptable.

The staff concludes that the design of the ESF atmosphere cleanup systems, including the equipment and instrumentation to control the release of radioactive materials in gaseous effluents following a postulated DBA, is acceptable except as noted. This conclusion is based on the applicant having met the requirements of GDC 19, 41, and 61 by providing ESF atmosphere cleanup systems on the control room habitability, containment, and associated systems. The applicant has met the requirements of GDC 41, 43, and 64 by providing for the inspection and testing of the ESF atmosphere cleanup systems and monitoring for radioactive materials in effluents from these systems. In meeting these regulations, the applicant has demonstrated that the design of the ESF atmosphere cleanup systems meets the guidelines of RG 1.52 and the ANSI N509 and N510 industry standards, as referenced in the SRP. The staff has reviewed the applicant's system descriptions and design criteria for the ESF atmosphere cleanup systems. On the basis of its evaluation, with respect to the SRP criteria, the staff finds the proposed ESF atmosphere cleanup systems acceptable.

The filter efficiencies given in Table 2 of RG 1.52 are appropriate for use in accident analyses.

6.5.2 Fission Product Removal and Control Systems

The fission product removal and control systems used by Millstone Unit 3 to mitigate the radiological consequences for a DBA are

(1) Fuel building exhaust system (FBES) - This system is designed to maintain the fuel building at a negative pressure during fuel-handling operations and DBA to enhance fission product filtration.

(2) Charging pump, component cooling pump, and heat exchanger ventilation system - This system is designed to produce an air flow direction from the auxiliary building general areas into component cooling pump and heat exchanger areas during a LOCA to prevent the spread of contamination from areas of high contamination to areas of low contamination.

(3) Supplementary leak collection and release system (SLCRS) - This system is designed to maintain a negative 0.25-in. water gauge pressure in the containment enclosure building and associated contiguous structures (auxiliary building, ESF building, main steam valve building, and hydrogen recombiner building) during a LOCA. This is achieved by exhausting air from these areas and passing it through a charcoal filter assembly before releasing it to the atmosphere.

Redundant filtration units are provided for each of these ESF filter systems, and these systems are designed to reduce the concentration and quality of fission products released to the environment following postulated accidents. These systems provide suitable redundancy in components and features so that their safety functions can be accomplished assuming a single failure. The applicant states in the FSAR that the emergency ventilation systems described above are in compliance with RG 1.52. The staff's review of the conformance to RG 1.52 is given in Section 6.5.1 of this report. Thus, these systems conform to GDC 41. These systems are designed to permit periodic inspection and testing, and, therefore, conform to GDC 42 and 43.

6.5.3 Fission Product Control Systems and Structures

The review for SRP Section 6.5.3 was performed as part of the review for Sections 6.5.1 and 15 of this report. See those sections for a discussion of the review.

6.5.4 Ice Condenser as a Fission Product Cleanup System

Millstone Unit 3 does not use an ice condenser for containment atmosphere cleanup. An evaluation under the provisions of SRP Section 6.5.4 is not applicable.

6.6 Inservice Inspection of Class 2 and 3 Components

6.6.1 Compliance With the Standard Review Plan

The staff's review of Millstone Unit 3 is continuing because the applicant has not submitted a complete preservice inspection (PSI) program and has not completed the PSI examinations. The staff review, to date, was conducted in accordance with SRP Section 6.6, except as discussed below.

The review according to SRP Section 6.6, Paragraph II.3, will be conducted when the completed PSI program plan has been received.

The review according to SRP Section 6.6, Paragraph II.4, has not been conducted because this area applies only to inservice inspection (ISI) not to PSI. This subject will be addressed during review of the ISI program after licensing.

The review according to SRP Section 6.6, Paragraph II.5, has been conducted. The applicant committed in the FSAR to incorporate ASME Code, Sections XI, Articles IWC-3000 and IWD-3000, "Standards for Examination Evaluation," into the PSI program. However, ongoing NRC generic activities and research projects indicate that the currently specified ASME Code procedures may not always be capable of detecting the acceptable size flaws specified in these standards. For example, ASME Code procedures specified for volumetric examination of vessels, bolts and studs, and piping have not proven to be capable of detecting unacceptable size flaws in all cases. The staff will continue to evaluate the development of new or improved procedures and will require that these improved procedures be made a part of the inservice examination requirements. The applicant's repair procedures based on ASME Code, Section XI, Articles IWC-4000 and IWD-4000, "Repair Procedures," have not been reviewed. Repairs are not generally necessary in the PSI program. This subject will be addressed during review of the ISI program.

The review according to SRP Section 6.6, Paragraph II.7, has not been completed because the applicant has not submitted a complete PSI program. The applicant's augmented ISI program will be reviewed after it is submitted.

The review according to SRP Section 6.6, Paragraph II.9, has not been completed because the applicant has not identified the limitations to examination. Specific areas where ASME Code examination requirements cannot be met will be identified as performance of the PSI progresses. The complete evaluation of the PSI program will be presented in a supplement to the SER after the applicant submits the required examination information, identifies all plant-specific areas where ASME Code, Section XI, requirements cannot be met, and provides a supporting technical justification.

6.6.2 Examination Requirements

GDC 36, 39, 42, and 45, Appendix A of 10 CFR 50, require, in part, that the Class 2 and 3 components be designed to permit appropriate periodic inspection of important components to ensure system integrity and capability. 10 CFR 50.55a(g) defines the detailed requirements for the PSI and ISI programs for light-water-cooled nuclear power facility components.

On the basis of the construction permit date of August 9, 1974, this section of the regulations requires that a PSI program for Class 2 and 3 components be developed and implemented using at least the edition and addenda of Section XI of the ASME Code applied to the construction of the particular components. The components (including supports) may meet the requirements set forth in subsequent editions of this Code and addenda, which are incorporated by reference in 10 CFR 50.55a(b) subject to the limitations and modifications listed therein. It is the intent of the applicant to comply with the PSI requirements of the 1980 Edition of the Code including Addenda through Winter 1980, except where specific relief is requested. The initial ISI program must comply with the requirements of the latest edition and addenda of Section XI of the ASME Code in effect 12 months before the date the operating license is issued, subject to the limitations and modifications listed in 10 CFR 50.55a(b).

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

Review has been completed on the information related to the PSI program submitted on and before June 18, 1984 and that presented in the FSAR through

Amendment 8. The preservice examination will be performed based on the requirements of the 1980 ASME Code, Section XI, through the Winter 1980 Addenda. The PSI program plan for the Class 3 components has not been received and will not be available until the first quarter of 1985. However, the applicant has stated in the FSAR that these components will be examined in accordance with the applicable Code requirements. The staff has established technical positions that should be included in the PSI program. The staff will review these sections of the PSI program for compliance and report the results in a supplement to the SER.

10 CFR 50.55a(b)(2)(iv) requires that ASME Code, Class 2, piping welds in the residual heat removal (RHR) systems, emergency core cooling systems (ECCSs), and containment heat removal (CHR) systems shall be examined. These systems should not be completely exempted from preservice volumetric examination based on Section XI exclusion criteria contained in IWC-1220. For example, staff review of the PSI program plan revealed that the Class 2 portions of the HPCI, CVCS, and containment spray system (CSS) will receive no volumetric inspections. It is the staff's position that the preservice inspection program must include volumetric examination of a representative sample of welds in the RHR, ECCS, and CHR systems. This is an open item in the safety evaluation.

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

(1) dockets a complete and acceptable PSI program
(2) submits all relief requests with a supporting technical justification

The applicant has not submitted the initial ISI program. This program will be evaluated after the applicable ASME Code edition and addenda can be determined based on 10 CFR 50.55a(b), but before ISI commences during the first refueling outage.

6.6.4 Conclusions

Compliance with the preservice and inservice inspections required by the ASME Code and 10 CFR 50 constitutes an acceptable basis for satisfying the applicable requirements of GDC 36, 39, 42, and 45.

Table 6.1 Containment isolation signals and actuation parameters

Signal	Activation parameter
Containment isolation phase A signal	High containment pressure (Hi-1) Low compensated steamline pressure Pressurizer low pressure Manual actuation
Containment isolation phase B signal	High containment pressure (Hi-3) Manual actuation
Containment depressurization actuation signal	High containment pressure (Hi-3) Manual actuation
Safety injection signal	High containment pressure (Hi-1) Low compensated steamline pressure Pressurizer low pressure Manual actuation
Main steam isolation signal	High steam pressure rate High containment pressure (Hi-2) Low steamline pressure Manual actuation
Feedwater line isolation	Safety injection Steam generator high-high level Low T_{avg}

7 INSTRUMENTATION AND CONTROLS

7.1 Introduction

7.1.1 Acceptance Criteria

FSAR Section 7.1 contains information pertaining to safety-related instrumentation and control systems, their design bases, and applicable acceptance criteria. The staff has reviewed the applicant's design, design criteria, and design bases for the instrumentation and control systems for Millstone Unit 3. The acceptance criteria used as the basis for this evaluation are those identified in the SRP (NUREG-0800) in Table 7-1, "Acceptance Criteria for Instrumentation and Control Systems Important to Safety," and Table 7-2, "TMI Action Plan Requirements for Instrumentation and Control Systems Important to Safety." These acceptance criteria include the applicable GDC and the Institute of Electrical and Electronics Engineers (IEEE) Std. 279, "Criteria for Protection System for Nuclear Power Generating Stations" (10 CFR 50.55a(h)). Guidelines for implementation of the requirements of the acceptance criteria are provided in IEEE standards, RGs, and BTPs identified in SRP Section 7.1. Conformance to the acceptance criteria provides the bases for concluding that the instrumentation and control systems meet the requirements of 10 CFR 50.

7.1.2 Method of Review

At Millstone Unit 3 a Westinghouse nuclear steam supply system (NSSS) with balance-of-plant (BOP) design provided by Stone and Webster Engineering Corporation is used. Many safety-related instrumentation and control systems are similar to those at Comanche Peak or McGuire and have been previously reviewed and approved by the staff. The staff concentrated its review on those areas where the Millstone Unit 3 design differs from previously reviewed designs and on those areas that have remained of concern during reviews of other similar plants. Several meetings were held with the applicant and the NSSS and BOP designers to clarify the design and to discuss staff concerns. Detail drawings - including piping and instrumentation diagrams, logic diagrams, control wiring diagrams, electrical one-line diagrams, and electrical schematic diagrams - were audited during the review.

7.1.3 General Conclusion

The applicant has identified the instrumentation and control systems important to safety and the acceptance criteria that are applicable to those systems as identified in the SRP. The applicant has also identified the guidelines - including the RGs and the industry codes and standards - that are applicable to the systems as identified in FSAR Table 7.1-1.

On the basis of the review of FSAR Section 7.1, the staff concludes that the implementation of the identified acceptance criteria and guidelines satisfies the requirements of GDC 1, "Quality Standards and Records," with respect to the design fabrication, erection, and testing to quality standards commensurate with the importance of the safety functions to be performed. The staff finds that the NSSS and the BOP instrumentation and control systems important to safety, addressed in FSAR Section 7.1, satisfy the requirements of GDC 1 and, therefore, are acceptable.

7.1.4 Specific Findings

7.1.4.1 Confirmatory Items

In a number of cases, the applicant has committed to provide additional documentation to address concerns raised by the staff during its review. On the basis of the information provided during meetings and discussions with the applicant, the technical issue has been resolved in an acceptable manner. However, the applicant must formally document his commitments for resolution of these items. The sections of this report that address these items are indicated in parentheses.

(1) Cable separation in NSSS process cabinets (7.2.2.1)

(2) Design modification for automatic reactor trip using shunt coil trip attachment (7.2.2.4)

(3) Reactor coolant pump underspeed trip (7.2.2.6)

(4) Conformance with BTP ICSB-26 (7.2.2.7)

(5) Test of engineered safeguard P-4 interlock (7.3.3.2)

(6) Steam generator level control and protection (7.3.3.4)

(7) IE Bulletin 80-06 concerns (7.3.3.5)

(8) Control building isolation reset (7.3.3.8)

(9) Power lockout feature for motor-operated valves (7.3.3.9)

(10) Failure mode and effects analyses of engineered safety features actuation system (7.3.3.10)

(11) Non-Class 1E control signals to Class 1E control circuits (7.3.3.11)

(12) Sequencer deficiency report (7.3.3.13)

(13) BOP instrumentation and control system testing capability (7.3.3.14)

(14) NUREG-0737, Item II.F.1, Accident Monitoring Instrumentation, Positions (4), (5), and (6) (7.5.2.4)

7.1.4.2 Technical Specification Item

The item to be included in the plant Technical Specifications and the information to be audited as part of the review of the proposed Technical Specifications are discussed in Section 7.2.2.2.

7.1.4.3 Licensing Condition

The item to be included as a license condition is discussed in Section 7.5.2.6.

7.1.4.4 Site Visit

A site review will be performed to confirm that the physical arrangement and installation of electrical equipment are in accordance with the design criteria and descriptive information reviewed by the staff. The site review will be completed before a license is issued; any problems found will be addressed in a supplement to this report.

7.1.4.5 Fire Protection Review

The review of the auxiliary shutdown panel discussed in Section 7.4 of this report includes the compliance of this panel with GDC 19, "Control Room." The aspects of the auxiliary shutdown panel related to fire protection and the review for conformance to 10 CFR 50, Appendix R (safe shutdown analysis), are included in Section 9.5 of this report.

7.1.5 TMI Action Plan Items

Guidance on implementation of the TMI Action Plan was provided to applicants in NUREG-0737. The items related to instrumentation and control systems are listed below. The specific section of the report addressing each item is indicated in parentheses.

(1) II.D.3 - Direct Indication of PORV and Safety Valve Position (7.5.2.3)

(2) II.E.1.2 - Auxiliary Feedwater System Automatic Initiation and Flow Indication (7.3.3.1)

(3) II.F.1 - Accident Monitoring Instrumentation, Positions (4), (5), and (6) (7.5.2.4)

(4) II.F.3 - Instrumentation for Monitoring Accident Conditions (7.5.2.6)

(5) II.K.3.9 - Proportional Integral Derivative Controller Modification (7.7.2.4)

(6) II.K.3.12 - Confirm Existence of Anticipatory Reactor Trip Upon Turbine Trip (7.2.2.5).

7.2 Reactor Trip System

7.2.1 Description

The reactor trip system (RTS) is designed to automatically limit reactor operation within the limits established in the safety analysis. This function is accomplished by tripping the reactor whenever predetermined safety limits are approached or reached. The RTS monitors variables that are directly related to system limitations or calculated from process variables. Whenever a variable exceeds a setpoint, the reactor is tripped by the insertion of control rods.

The RTS initiates a turbine trip when a reactor trip occurs. The RTS consists of sensors and analog and digital circuitry arranged in coincidence logic for monitoring plant parameters. Signals from these channels are used in redundant logic trains. Each of the two trains opens a separate and independent reactor trip breaker. During normal power operation, a dc undervoltage coil in each reactor trip breaker holds the breaker closed. For a reactor trip, the removal of power to the undervoltage coils opens the breakers. Opening either of two series-connected breakers interrupts the power from the rod-drive motor generator sets, and the control rods fall by gravity into the core. The rods cannot be withdrawn until the trip breakers are manually reset, and the trip breakers cannot be manually reset until the abnormal condition that initiated the trip is corrected. Bypass breakers are provided to permit the testing of the primary breakers.

In addition to the automatic trip of the reactor described above, there is also provision for manual trip by the operator. The manual trip consists of two switches. Actuation of either switch removes power from the undervoltage coils and energizes the shunt trip coils of both reactor trip breakers. The shunt trip coils are a diverse means for tripping the reactor trip breakers. The reactor will also be tripped by actuating either of the two manual switches for safety injection.

The generic implications of the Salem anticipated transient without scram (ATWS) events are discussed in Section 7.2.2.4 of this report.

The reactor trips listed below are provided in the Millstone design. The numbers in parentheses after each trip function indicate the coincident logic, for example, two out of three (2/3).

(1) nuclear overpower trips

- (a) power range high neutron flux trip (2/4)
- (b) intermediate range high neutron flux trip (1/2)
- (c) source range high neutron flux trip (1/2)
- (d) power range high positive neutron flux rate trip (2/4)
- (e) power range high negative neutron flux rate trip (2/4)

(2) core thermal overpower trips

- (a) overtemperature ΔT trip (2/4)
- (b) overpower ΔT trip (2/4)

(3) reactor coolant system pressurizer pressure and water level trips

- (a) pressurizer low pressure trip (2/4)
- (b) pressurizer high pressure trip (2/4)
- (c) pressurizer high water level trip (2/3)

(4) reactor coolant system low flow trips

- (a) low reactor coolant flow (2/3 per loop)
- (b) reactor coolant pump underspeed trip in any two loops (2/4)

(5) steam generator low-low level trip (2/4)
(6) turbine trip (anticipatory)

(a) low auto stop oil pressure (2/3)
(b) turbine stop valves closed (4/4)

(7) safety injection logic trip (1/2)
(8) manual trip (1/2)
(9) general warning alarm (2/2)

The power range high neutron flux trip has two bistables to initiate reactor trip at separate high flux setpoints. The higher setting trip is active during all modes of operation. The low setting trip provides protection during reactor startup and shutdown when the reactor is below 10% power. The lower setting trip can be manually blocked above 10% power (P-10) and is automatically reinstated when power is reduced below the P-10 interlock setpoint.

The intermediate range trip provides protection during reactor startup and shutdown. This trip can be manually blocked above 10% power (P-10) and is automatically reinstated when power is reduced below the P-10 interlock setpoint.

The source range trip provides protection during reactor startup and shutdown when the neutron flux channel is below the P-6 interlock setpoint (6×10^{-11} amp). This trip can be manually blocked above the P-6 interlock setpoint and automatically reinstated when power is reduced below the P-6 interlock setpoint.

A power range high positive neutron flux rate trip occurs when an abnormal increase in the rate of nuclear power is detected. This trip provides departure from nucleate boiling (DNB) protection against low-worth rod ejection accidents from midpower and is active during all modes of operation.

A power range high negative neutron flux rate trip occurs when an abnormal decrease in the rate of nuclear power is detected. This trip provides protection against two or more dropped rods and is active during all modes of operation.

The overtemperature ΔT trip protects the core against a low departure from nucleate boiling ratio (DNBR). The setpoint for this trip is continuously calculated by analog circuits to compensate for the effects of temperature, pressure, and axial neutron flux difference on DNBR limits.

The overpower ΔT trip protects against excessive power (fuel rod rating protection). The setpoint for this trip is continuously calculated by analog circuits to compensate for the effects of temperature and axial neutron flux difference.

The pressurizer low pressure trip is used to protect against low pressure that could lead to DNB. The reactor is tripped when the pressurizer pressure (compensated for rate of change) falls below a preset limit. This trip may be manually blocked below approximately 10% power (P-7 interlock) to allow startup and controlled shutdown. It is automatically reinstated when power is increased above 10% power.

The pressurizer high pressure trip is used to protect the reactor coolant system against system overpressure. The reactor is tripped when pressurizer pressure exceeds a preset limit.

The pressurizer high water level trip is provided as a backup to the pressurizer high-pressure trip and serves to prevent water relief through the pressurizer safety valves. This trip is automatically blocked below approximately 10% of full power (P-7 interlock) to allow startup.

The low reactor coolant flow trip protects the core against DNB resulting from a loss of primary coolant flow. Above the P-7 setpoint (approximately 10% power), a reactor trip will occur if any two loops have low flow. Above the P-8 setpoint (approximately 48% power), a trip will occur if any one loop has low flow. These trips are automatically blocked below the respective interlock setpoints.

The reactor coolant pump (RCP) underspeed trip protects the reactor core from DNB resulting from low primary coolant flow. The RCP underspeed trip replaces the undervoltage and underfrequency reactor trips used in some Westinghouse plants. The principal reason for this change is to improve plant availability during voltage dip transients. There is one speed detector mounted on each reactor coolant pump. This trip is automatically blocked below P-7 to permit plant startup.

The steam generator low-low water level trip protects the reactor from loss of heat sink.

A reactor trip on a turbine trip is actuated by two out of three trip fluid pressure signals or by all (four out of four) closed signals from the turbine steam stop valves. A turbine trip causes a reactor trip above 50% power (P-9 interlock). Below 50% power this trip is automatically blocked.

A safety injection signal initiates a reactor trip. This trip protects the core against a loss of reactor coolant or overcooling.

The manual trip is initiated by operation of either of two switches. Each switch deenergizes the undervoltage coils in each reactor trip breaker, and shunt coils in these breakers are energized at the same time, which provides a diverse means to ensure that the trip breakers are tripped. Bypass breakers that are closed only when testing the reactor trip breakers are also tripped via their undervoltage and shunt trip coils by a manual reactor trip.

A general warning alarm in both solid-state protection system trains initiates a reactor trip. The general warning alarm is provided for each train of the solid-state protection system and is activated when the corresponding train is being tested or is otherwise inoperable. The trip resulting from the general warning alarm in both trains provides protection for conditions under which both trains of the protection system may be inoperable.

The analog portion of the RTS consists of a portion of the process instrumentation system (PIS) and the nuclear instrumentation system (NIS). The PIS includes those sensors that measure temperature, pressure, fluid flow, and level. The PIS also includes the power supplies, signal conditioning, and bistables that provide initiation of protective functions. The NIS includes the neutron flux monitoring instruments, including power supplies, signal conditioning, and bistables that provide initiation of protective functions.

The digital portion of the RTS consists of the solid-state logic protection system (SSLPS). The SSLPS takes binary inputs (voltage/no voltage) from the PIS and NIS channels corresponding to normal/trip conditions for plant parameters. The SSLPS uses these signals in the required logic combinations and generates trip signals (no voltage) to the undervoltage coils of the reactor trip circuit breakers. The system also provides annunciator, status light, and computer input signals that indicate the condition of the bistable output signals, partial and full trip conditions, and the status of various blocking, permissive, and actuation functions. In addition, the SSLPS includes the logic circuits for testing.

Analog signals derived from protection channels used for nonprotective functions such as control, remote process indication, and computer monitoring are provided by isolation amplifiers located in the protective system cabinets. The isolation amplifiers are designed so that a short circuit, open circuit, or the application of credible fault voltages from within the cabinets on the isolated output portions of the circuit (nonprotective side) will not affect the input signal. The signals obtained from the isolation amplifiers are not returned to the protective system cabinets.

7.2.2 Specific Findings

7.2.2.1 Cable Separation in NSSS Process Cabinets

The staff requested that cable separation inside NSSS cabinets be addressed in the FSAR. The applicant indicated that FSAR Section 7.2 will be revised to include a reference to WCAP-8872A and confirm that the BOP control systems comply with the NSSS interface criteria. This is a confirmatory item.

7.2.2.2 Trip Setpoint and Margins

The setpoints for the various functions in the reactor trip system are determined on the basis of the accident analysis requirements. As such, during any anticipated operational occurrence or accident, the reactor trip maintains system parameters with the following limits.

(1) minimum departure from nucleate boiling ratio of 1.30
(2) maximum system pressure of 2,750 psi (absolute)
(3) fuel rod maximum linear power of 18.0 kW per foot

The staff requested detailed information on the methodology used to establish the Technical Specification trip setpoints and allowable values for the reactor protection system (including reactor trip and engineered safety feature channels) assumed to operate in the FSAR accident and transient analyses. This includes the following information:

(1) The trip setpoint and allowable value for the Technical Specifications.

(2) The safety limits necessary to protect the integrity of the physical barriers that guard against uncontrolled release of radioactivity.

(3) The values assigned to each component of the combined channel error allowance (e.g., modeling uncertainties, analytical uncertainties, transient overshoot, response time, trip unit setting accuracy, test equipment accuracy, primary element accuracy, sensor drift, nominal and harsh environmental allowances, trip unit drift), the basis for these values, and the method used to sum the individual errors. Where zero is assumed for an error, a justification that the error is negligible should be provided.

(4) The margin (i.e., the difference between the safety limit and the setpoint less the combined channel error allowance).

The detailed trip setpoint review will be performed as part of the staff's review of the plant Technical Specifications and will be completed before the operating license is issued. The applicant was requested to provide an evaluation and/or an analysis of the effect of postaccident environmental conditions on the reactor trip system instrumentation (Technical Specification Table 2.2-1) and the engineered safety feature actuation system instrumentation (Technical Specification Table 3.3-4) and its impact on establishing setpoints. This information will be provided for the staff's review with the applicant's proposed Technical Specifications.

7.2.2.3 Response-Time Testing

By letter dated February 15, 1984, the applicant indicated that the report entitled "The Use of Process Noise Measurement To Determine Response Characteristics of Protection Sensors in U.S. Plants" as submitted by Westinghouse to the staff on August 15, 1983, provides justification for the use of this technique for response-time testing. The staff has reviewed the Westinghouse report, which describes the test method and provides the results of tests conducted at operating reactors from 1977 through 1982 using this technique. On the basis of its review, the staff finds that there is an adequate basis to conclude that the use of process noise measurements will provide an acceptable means to fulfill the requirements for response-time testing as specified in the plant Technical Specifications.

7.2.2.4 Design Modification for Automatic Reactor Trip Using Shunt Coil Trip Attachment

The Westinghouse Owners Group (WOG) has submitted a generic design modification to provide automatic reactor trip system (RTS) actuation of the breaker shunt trip attachments in response to Salem ATWS events. The staff has reviewed and accepted the generic design modification and has identified additional information required on a plant-specific basis. By a letter dated May 4, 1984, the applicant addressed the staff's concern on plant-specific questions. The staff finds that the applicant's commitments are acceptable subject to a detailed review of the electrical schematic/elementary diagrams. This is a confirmatory item.

7.2.2.5 NUREG-0737, Item II.K.3.12, Confirm Existence of Anticipatory Reactor Trip on Turbine Trip

The Millstone design includes an anticipatory reactor trip on a turbine trip above 50% of rated thermal power (P-9 interlock). The staff finds that the design is in compliance with the Action Plan guidelines.

7.2.2.6 Reactor Coolant Pump Underspeed Trip

In the Millstone 3 design, the reactor coolant pump (RCP) underspeed trip is used to protect the reactor core from DNB if there is loss of flow in more than one loop. Because this is a first-of-a-kind parameter used for the reactor trip system, and each pump only uses one speed sensor, the staff requests that an analysis be provided to address the conformance with the requirements of IEEE Std. 279. The FSAR should be updated to reflect the deletion of the P-17 interlock and the RCP shaft low-low speed trip. This is a confirmatory item.

7.2.2.7 Conformance With Branch Technical Position ICSB-26

Branch Technical Position (BTP) ICSB-26, "Requirements for Reactor Protection System Anticipatory Trip," applies to the entire reactor protection system (RPS) from the sensors to the final actuated device. For sensors located in nonseismic areas, the installation (including circuit routing) and design should be such that the effects of credible faults (i.e., grounding, shorting, application of high voltage, or electromagnetic interference) or failures in these areas could not be propagated back to the RPS and degrade the RPS performance or reliability. There are three groups of RPS-related cables that are routed in the turbine building:

(1) turbine trip cause reactor trip input cables
(2) reactor trip to trip the turbine output cables
(3) turbine first-stage pressure input to RPS interlock circuits

The staff has audited the cable routing drawings and finds that they are in conformance with the separation criteria. The applicant is requested to revise FSAR Section 7.2 to indicate that his design is in conformance with BTP ICSB-26. This is a confirmatory item.

7.2.3 Evaluation Conclusion

The staff has conducted an audit review of the RTS for conformance to applicable RGs and industry codes and standards. In Section 7.1 of this SER, the staff concluded that the applicant had adequately identified the guidelines applicable to these systems. On the basis of its audit review of the design to determine conformance to the guidelines, the staff finds that there is reasonable assurance that the systems will conform to the applicable guidelines. The scope of the review included the FSAR descriptive information; electrical, instrumentation, and control drawings; and piping and instrumentation diagrams. In addition, the staff met with the applicant, architect/engineer, and the NSSS vendor. These meetings provided a forum for exchanging information and answering staff questions.

The staff review has included the identification of those systems and components for the RTS that are designed to survive the effects of earthquakes, other natural phenomena, abnormal environments, and missiles. On the basis of its review, the staff concludes that the applicant has identified the systems and components consistent with the design bases for the RTS. Sections 3.10 and 3.11 of this SER address the qualification programs to demonstrate the capability of these systems and components to survive applicable events. Therefore, the staff finds that the identification of the systems and components satisfies this aspect of GDC 2 and 4.

On the basis of its review, the staff concludes that the RTS conforms to the design-basis requirements of IEEE Std. 279. The RTS includes the provision to sense accident conditions and anticipated operational occurrences and initiate reactor shutdown consistent with the analyses presented in Chapter 15 of the FSAR. Therefore, the staff finds that the RTS satisfies the requirements of GDC 20.

The RTS adequately conforms to the guidance for periodic testing in RG 1.22, "Periodic Testing of Protection System Actuation Functions," and IEEE Std. 338, as supplemented by RG 1.118, "Periodic Testing of Electric Power and Protection System." The bypassed and inoperable status indication adequately conforms to RG 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems." The RTS adequately conforms to the guidance on the application of the single-failure criterion in IEEE Std. 379, as supplemented by RG 1.53, "Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems." On the basis of its review, the staff concludes that the RTS satisfies the requirements of IEEE Std. 279 with regard to system reliability and testability. Therefore, the staff finds that the RTS satisfies the requirements of GDC 21.

The RTS adequately conforms to the guidance in IEEE Std. 384 as supplemented by RG 1.75 for protection system independence. On the basis of its review, the staff concludes that the RTS satisfies the requirements of IEEE Std. 279 with regard to independence of systems and hence satisfies the requirements of GDC 22.

On the basis of its review of failure modes and effects for the RTS, the staff concludes that the system is designed to fail into a safe mode if conditions such as disconnection of the system, loss of energy, or a postulated adverse environment are experienced. Therefore, the staff finds that the RTS satisfies the requirements of GDC 23.

On the basis of its review of the interface between the RTS and plant-operating control systems, the staff concludes that the system satisfies the requirements of IEEE Std. 279 with regard to control and protection system interaction. Therefore, the staff finds that the RTS satisfies the requirements of GDC 24.

On the basis of its review of the RTS, the staff concludes that the system satisfies the protection system requirements for malfunctions of the reactivity control system, such as accidental withdrawal of control rods. Chapter 15 of the FSAR addresses the capability of the system to ensure that fuel design limits are not exceeded for such events. Therefore, the staff finds that the RTS satisfies the requirements of GDC 25.

The staff's conclusions are based on the requirements of IEEE Std. 279 with respect to the design of the RTS. Therefore, the staff finds that the RTS satisfies the requirements of 10 CFR 50.55a(h) with regard to IEEE Std. 279.

The review of the RTS included the examination of the dependence of this system on the availability of essential auxiliary support (EAS) systems. On the basis of its review, the staff concludes that the design of the RTS is compatible with the functional performance requirements of EAS systems. Therefore, it finds the interfaces between the RTS design and the design of the EAS systems acceptable.

7.3 Engineered Safety Features Systems

7.3.1 Engineered Safety Features Actuation System

The engineered safety features actuation system (ESFAS) is a portion of the plant protection system that monitors selected plant parameters and, on detection of out-of-limit conditions of these parameters, will initiate actuation of appropriate engineered safety features (ESF) systems and EAS system equipment. The ESFAS includes both automatic and manual initiation of these systems. Also included with the ESF systems are the control systems that regulate operation of ESF systems following their initiation by the protection system.

The ESFAS is a functionally defined system and consists of

(1) process instrumentation and control
(2) solid-state and relay logic
(3) ESF test circuits
(4) manual actuation circuits
(5) emergency generator load sequence control logic

The ESFAS includes two distinct portions of circuitry: (1) an analog portion consisting of three to four redundant channels per parameter or variable to monitor various plant parameters such as reactor coolant and steam system pressures, temperatures, and flows and containment pressure and (2) a digital portion consisting of redundant logic trains that receive inputs from the analog protection channels and perform the logic to actuate the ESF equipment. The ESFAS is composed of the NSSS circuits designed by Westinghouse and the BOP circuits designed by Stone and Webster Engineering Corporation.

The actuation signals for each of the ESFAS functions are listed below. The numbers in parentheses after each actuation channel indicate the coincident logic, for example, two out of four (2/4).

(1) safety injection

   (a) manual (1/2)
   (b) high-1 containment pressure (2/3)
   (c) low compensated steamline pressure (2/3 in any line)
   (d) low pressurizer pressure (2/4)

(2) containment depressurization

   (a) manual (2/4)

(b) high-3 containment pressure (2/4)

(3) containment isolation

(a) phase A isolation

- safety injection (same as Item (1) above)
- manual (1/2)

(b) phase B isolation

- high-3 containment pressure (2/4)
- manual (2/4)

(4) steamline isolation

(a) low compensated steamline pressure (2/3 in any line)
(b) high-2 containment pressure (2/3)
(c) high negative steam pressure rate (2/3 in any line)
(d) manual (1/2 for a'l lines or 1/1 for each valve)

(5) feedwater line isolation

(a) safety injection (same as Item (1) above)
(b) high steam generator level (2/3 in any generator)
(c) low T_{avg} (2/4) coincident with reactor trip

(6) auxiliary feedwater system actuation

The motor-driven auxiliary feedwater pumps will be started on any of the following signals:

(a) safety injection (same as Item (1) above)
(b) low-low steam generator level (2/4) in any generator
(c) loss of power (2/4) undervoltage at 4.16-kV buses
(d) manual actuation (1/1)

The turbine-driven auxiliary feedwater pump will be started on any of the following signals:

(a) low-low level (2/4) in two steam generators
(b) loss of power (2/4) undervoltage at 4.16-kV buses
(c) manual actuation (1/1)

(7) control building isolation

(a) high-high radiation in air intake (1/2)
(b) high-1 containment pressure (2/3)
(c) high chlorine in air intake (1/2)
(d) manual safety injection (1/2)
(e) manual actuation (1/2)

7.3.2 Engineered Safety Features and Essential Auxiliary Support Systems Operation

The following systems are provided:

(1) engineered safety features systems

  (a) emergency core cooling system
  (b) containment depressurization system

    - quench spray system
    - containment recirculation system

  (c) containment isolation system including main steam and feedwater isolation
  (d) design-basis accident hydrogen recombiner system
  (e) supplementary leak collection and release system
  (f) auxiliary feedwater system
  (g) ESF filtration system

    - control room ventilation system
    - fuel building exhaust system
    - ESF equipment areas ventilation and filtration system

(2) essential auxiliary support systems

  (a) service water system
  (b) reactor component cooling system
  (c) emergency onsite power supply system
  (d) emergency diesel generator support systems

7.3.2.1 Emergency Core Cooling System

The emergency core cooling system (ECCS) cools the reactor core and provides shutdown capability for (1) pipe breaks in the reactor coolant system (RCS) that cause a loss of primary coolant greater than that which can be made up by the normal makeup system, (2) rod cluster control assembly ejection, (3) pipe breaks in the secondary coolant system, and (4) steam generator tube failure. The primary function of the ECCS is to remove the stored and fission product decay heat from the reactor core during accident conditions. The ECCS consists of the centrifugal charging safety injection pumps, residual heat removal pumps, accumulators, containment recirculation pumps, refueling water storage tank (RWST), and the associated piping, valves, and instrumentation.

The ECCS provides reactor shutdown capability for the accidents described above by injecting borated water into the RCS. The system's safety function can be performed with a single active failure (short term) or passive failure (long term). The emergency diesel generators supply power if offsite power is unavailable.

The safety injection signal will start the diesel generators and automatically initiate the following actions in the ECCS:

(1) starts charging pumps
(2) opens RWST suction valves to charging pumps

(3) opens charging pumps to RCS cold-leg injection headers isolation valves
(4) closes normal charging path valves
(5) closes charging pump miniflow valves
(6) starts safety injection pumps
(7) starts residual heat removal (RHR) pumps
(8) opens any closed accumulator isolation valves
(9) closes volume control tank outlet isolation valves

Switchover from the injection mode to the recirculation mode involves the procedures described below. The changeover from the injection mode to the recirculation mode is initiated manually by operator action from the main control room. The switchover procedures are:

(1) From injection to cold-leg recirculation

(a) The RHR pumps are stopped automatically when RWST level reaches low-low setpoint.

(b) Valves associated with RHR pumps and containment recirculation pumps for cold-leg recirculation mode are aligned.

(c) The safety injection pump miniflow valves are closed.

(d) Safety injection and charging pumps to the containment recirculation pump discharge are aligned.

(e) The refueling water storage tank is isolated.

(2) After approximately 15 hours, cold-leg recirculation is terminated and hot-leg recirculation is initiated.

(a) The containment recirculation pump is aligned to deliver directly to the RCS through the hot-leg injection header.

(b) The containment recirculation pump is aligned to deliver to the RCS via the safety injection pumps.

7.3.2.2 Containment Depressurization System

The containment depressurization system consists of the quench spray system and the containment recirculation spray system. Subsequent to a design-basis accident (DBA), the quench spray pumps are started automatically on receipt of a containment depressurization actuation (CDA) signal. The isolation valves in the quench spray discharge headers and the chemical addition tank open on receipt of a CDA signal. Each redundant quench spray subsystem draws water independently from the RWST. Sodium hydroxide solution is added to the quench spray by direct gravity feed from the chemical addition tank. The quench spray pumps are stopped automatically on receipt of an RWST low-3 signal. Before the RWST reaches the low-3 level, the RWST low-2 signal alerts the operator to take manual action for changeover from injection mode to recirculation mode. The containment recirculation pumps start automatically on a CDA signal after about a 5-min time delay. The containment recirculation pumps take suction from the containment sump. Two of the four containment recirculation pumps perform the containment spray function to replace the quench spray pumps during the recirculation mode. The other two pumps are used for cold-leg injection.

7.3.2.3 Containment Isolation System Including Main Steam and Feedwater Isolation

The safety function of the containment isolation system (CIS) is to automatically isolate the process lines penetrating the containment structure. The CIS is designed to limit the release of radioactive materials from the containment following an accident.

The CIS is automatically actuated by signals developed by the ESFAS in two phases: phase A containment isolation and phase B containment isolation. Phase A isolates all nonessential process lines penetrating the containment. Phase B isolates all other process lines not included in phase A containment isolation, except for the safety injection and containment spray lines.

Containment isolation valves, which are equipped with power operators and are automatically actuated, may also be controlled individually by manual switches in the control room. Containment isolation valves with power operators are provided with an open/closed indication, which is displayed in the control room at the main control board and the safeguard status panel. All electric power supplies and equipment necessary for containment isolation are Class 1E.

The main steamline isolation signal is generated on low steamline pressure, high-2 containment pressure, or high negative steam pressure rate. A manual bypass permissive is provided for the low steamline pressure signal for use during normal plant cooldowns and heatups. The high negative steamline pressure rate is used to initiate main steam isolation when the low steamline pressure signals are bypassed during normal plant startup and shutdown. The main steam isolation trip valves are Y-pattern-type globe valves designed to prevent main steam flow in both the forward and reverse directions. Closing forces are provided by steam pressure from the main steamline. Each main steam isolation valve is closed by redundant logic trip signals. The main steam isolation valves are capable of being tested on line by partial closure of the valve.

Feedwater line isolation is provided to terminate main feedwater following a pipe rupture or an excessive feedwater flow event. The feedwater line isolation signal is generated on safety injection, high steam generator water level, or low reactor coolant temperature coincident with reactor trip. Upon receipt of this signal, the main feedwater isolation valves and other valves associated with the main feedwater lines are closed. Redundant actuation systems are provided for each valve operator and receive closure signals from the two redundant ESFAS logic trains.

7.3.2.4 Design-Basis Accident Hydrogen Recombiner System

The DBA hydrogen recombiner system controls the building up of hydrogen gas inside the containment. The DBA hydrogen recombiner system consists of hydrogen monitors and hydrogen recombiners. Redundant safety-grade hydrogen monitoring systems are provided. Each train contains a stand-alone analyzer and control cabinet that analyzes, monitors, alarms, and trends containment hydrogen concentration. Redundant safety-grade hydrogen recombiners maintain the hydrogen in the containment atmosphere at a safe concentration following a DBA. The system provides analog output for display, recording, and alarming in the main control room.

7.3.2.5 Supplementary Leak Collection and Release System

A dual containment design is used at Millstone Unit 3. There is a containment enclosure building surrounding the containment. The supplementary leak collection and release system (SLCRS) is designed to maintain the containment enclosure building at a negative pressure of 0.25 in. wg after a design-basis accident (DBA). The SLCRS also maintains part of contiguous buildings - main steam valves building, engineering safety features building, hydrogen recombiner building, and auxiliary building - under a negative pressure following a DBA.

The SLCRS exhausts air from these areas and filters and removes particulate and gaseous iodine from the air before discharge to the atmosphere. The SLCRS consists of two exhaust fans, each supplied from a separate emergency bus, two filter banks, and the associated ductwork and dampers. The safety injection signal opens both train A and B filter bank inlet dampers and starts both train A and B exhaust fans. High differential pressure across each filter bank is alarmed in the control room. The filtered exhaust is monitored for radiation before it is discharged to the atmosphere through the Millstone Unit 1 stack.

7.3.2.6 Auxiliary Feedwater System

The function of the auxiliary feedwater system (AFWS) is to provide an adequate supply of water to the steam generators if the main feedwater system is not available. The AFWS consists of two motor-driven pumps and one turbine-driven pump with associated valves, controls, and instrumentation. Each motor-driven pump supplies water to two of the four steam generators; the turbine-driven pump supplies water to all four steam generators. The auxiliary feedwater (AFW) actuation system will automatically start the pumps and provide feedwater to the steam generators. The initiating conditions are listed in Section 7.3.1, Item (6). The AFW pump suction is normally supplied from the seismic Category I demineralized water storage tank. An additional source of water is available from a non-seismic Category I condensate storage tank. The service water is the long-term safety-grade source of auxiliary feedwater which can be manually connected by spool pieces.

The AFWS can be manually initiated and controlled from the main control board or the auxiliary shutdown panel. The AFWS control is discussed in Section 7.4 of this report.

The amount of flow to any steam generator is limited by cavitating venturis located in the auxiliary feedwater line to each steam generator. The cavitating venturis will prevent runout flow to a depressurized steam generator. Manual isolation of AFW flow to a depressurized steam generator can be performed from the main control board or the auxiliary shutdown panel.

7.3.2.7 Engineered Safety Features Filtration System

Control Room Ventilation System

The control room ventilation system includes the control room air conditioning system, the instrument room and computer room air conditioning system, control room emergency ventilation and pressurization system, and other control building ventilation systems. The control room is normally maintained at a slightly positive pressure. The pressure is maintained by redundant isolation valves or dampers on all inlet and exhaust openings. Redundant radiation monitors and chlorine gas detectors are located at the control room air intake. High radiation or high chlorine levels will automatically cause isolation of the control room. After isolation, compressed air from air storage tanks is used to maintain a positive air pressure during the first hour following an accident. After an hour, outdoor air is introduced to the control room through redundant emergency filtration trains. The control room air intake is also provided with smoke detectors to actuate smoke alarms. Smoke can be purged by the purge ventilation system.

Fuel Building Exhaust System

The fuel building filter banks are normally bypassed by the unfiltered exhaust fan. During refueling and in the event of high radiation, the fuel building exhaust is manually diverted to the fuel building filter bank. Either train A or train B is operated with the other train at standby.

Equipment Areas Ventilation and Filtration System

The ESF equipment areas ventilation and filtration system controls and minimizes the potential for spread of airborne radioactive material within the building. On receipt of a safety injection signal (SIS) or containment depressurization signal (CDS), all the nonsafety-related ventilation systems will be shut down and isolated except the areas served by the ESF filtration system. These areas include the charging pump rooms, component cooling water pump rooms, safety-related heat exchanger areas, rod control areas and safety-related motor control center areas. The ESF filtration system includes redundant trains. Each train consists of exhaust fans, filter banks, and the associated ductwork and dampers. Each train is powered from a separate emergency bus. The exhaust air can be directed through the auxiliary building filters to the atmosphere. The filter inlet dampers from the charging pump and component cooling pump areas are in parallel and fail open on loss of power or instrument air. The filter inlet dampers from other safety-related areas are in series and fail closed on loss of power or instrument air. The ESF filter banks can be manually controlled from the control room or at the switchgear. Control transfer switches are provided at the switchgear. An alarm is sounded when LOCAL control is selected. High differential pressure across a filter bank is alarmed in the control room.

7.3.2.8 Service Water System

The service water system performs both safety and nonsafety functions by providing cooling water for heat removal components during all modes of operation. The service water system consists of two trains. Each train contains two half-capacity service water pumps, two strainers, two booster pumps, and associated piping and valves. One pump in each train is operated with the other on standby. The service water system is designed to meet the single-failure criterion. Power is supplied to redundant pumps from separate emergency buses. On receipt of a safety injection signal or loss-of-power signal, the water supply lines to the nonsafety-related equipment are isolated. On receipt of a containment depressurization actuation signal, the water supply lines to the reactor plant component cooling water heat exchangers are isolated and the water supply lines to the containment recirculation coolers are opened.

7.3.2.9 Reactor Components Cooling Systems

The cooling systems for reactor components consist of the charging pumps cooling system, safety injection pumps cooling system, reactor plant component cooling water (RPCCW) system and other nonsafety-related component cooling systems. These systems are used individually or in combination to provide cooling water for heat removal from reactor plant components.

The charging pumps cooling system is a safety-related closed-loop cooling system that transfers the heat load from the charging pumps lubricating oil coolers to the service water system. This system consists of two full-capacity pumps, two coolers, a surge tank, and associated piping and valves. On failure of the operating cooling pump, the standby pump will automatically start. Either pump can supply cooling water to any charging pump oil cooler.

The safety injection pumps cooling system is a safety-related closed-loop cooling system that cools the safety injection pumps bearing oil. This system consists of two full-capacity pumps, two coolers, a surge tank, and associated piping and valves.

The RPCCW system is a closed-loop cooling system. It includes three half-capacity pumps and heat exchangers, a surge tank, a chemical addition tank, and associated piping and valves. Two redundant trains serve those components essential for safe shutdown but not required for accident mitigation. One pump and heat exchanger are provided as a spare. The pump can be manually connected to either train's emergency bus. The spare pump motor breaker has to be racked out from one train cubicle and then racked into the other train cubicle to prevent a cross tie between redundant buses. An electrical interlock prevents simultaneous operation of two pumps on the same train. Redundant pressure switches are located at the nonsafety portion water supply header to detect a drop in pressure, which indicates a rupture of nonsafety-related system piping. Low pressure automatically isolates component cooling water to the nonsafety portions of the system.

7.3.2.10 Emergency Onsite Power Supply System

The emergency onsite power supply system consists of two 4.16-kV diesel generators, two 4.16-kV ESF buses, various ESF and non-ESF 480-V buses, motor control centers, and 208/120-V power panels. There are four 120-Vac safety-related power distribution panels for safety-related vital instrumentation and control loads. Each power panel has a separate rectifier/inverter. The dc power system consists of four Class 1E dc power panels (two panels per train) and two non-safety dc power panels. Each Class 1E dc power panel consists of a battery bank and a static battery charger. One spare battery charger per train is available to replace either of the two chargers in that train.

7.3.2.11 Emergency Diesel Generator Support Systems

The diesel generator fuel oil system, the diesel engine cooling water system, the diesel generator starting air system, the diesel engine lubrication system, and the diesel generator air intake and exhaust system are essential auxiliary support systems. These systems are evaluated in Section 9.5 of this report.

7.3.3 Specific Findings

7.3.3.1 NUREG-0737, Item II.E.1.2, AFWS Automatic Initiation and Flow Indication

The automatic system used to initiate the operation of the auxiliary feedwater system is part of ESFAS. The redundant actuation channels that provide signals to the pumps and valves are physically separated and electrically independent. Redundant trains are powered from independent Class 1E power sources. The initiation signals and circuits are testable during power operation, and the test requirements are included in the plant Technical Specifications. Manual initiation and control can be performed from the main control board or the auxiliary shutdown panel. No single failure within the manual or automatic initiation system for the auxiliary feedwater system will prevent initiation of the system by manual or automatic means. The environmental qualification is addressed in Section 3.11 of this report.

Redundant auxiliary feedwater flow instrument channels are provided for each steam generator. Each channel is powered from a separate Class 1E power source. Auxiliary feedwater flow indicators are located at the main control board and the auxiliary shutdown panel. The staff concludes that the design satisfies the requirements of NUREG-0737, Item II.E.1.2.

7.3.3.2 Test of Engineered Safeguards P-4 Interlock

On November 7, 1979, Westinghouse notified the Commission of an undetectable failure that could exist in the engineered safeguards P-4 interlocks. Test procedures were developed to detect failures that might occur. The procedures require the use of voltage measurements at the terminal blocks of the reactor trip breaker cabinets.

The staff raised a concern on the possibility of accidental shorting or grounding of safety system circuits during testing of the P-4 interlocks. The applicant has committed to incorporate builtin test features to facilitate testing of the P-4 interlock. This is a confirmatory item subject to documentation of this change.

7.3.3.3 Level Measurement Errors Resulting From Environmental Temperature Effects on Level Instrument Reference Legs

The staff requested that the applicant evaluate the effects of high temperatures in reference legs of water level measurement systems resulting from high-energy-line breaks. This issue was addressed for operating reactors through IE Bulletin 79-21. In FSAR Amendment 5, the applicant committed to insulate the steam generator reference legs in response to the heatup concern addressed in IE Bulletin 79-21. The staff finds this acceptable.

7.3.3.4 Steam Generator Level Control and Protection

Three steam generator level channels are used in a two-out-of-three logic for isolation of feedwater on high steam generator level. One of the three level channels is used for control. This design for actuation of feedwater isolation does not meet the requirements of Paragraph 4.7 of IEEE Std. 279 on "Control and Protection System Interaction" in that the failure of the level channel used for control could require protective action and the remainder of the protection system channels would not satisfy the single-failure criterion. By a letter dated May 4, 1984, the applicant stated that the high steam generator level trip will be changed to two-out-of-four logic. The staff finds that the applicant's commitment for design modification is acceptable. This is a confirmatory item subject to documentation of these changes in the FSAR system description and the related drawings.

7.3.3.5 IE Bulletin 80-06 Concerns

As was done for operating reactors through IE Bulletin 80-06, the staff requested that the applicant review all safety systems to determine if any safety equipment would change state after reset. In FSAR Amendment 5, the applicant stated that the requested reviews have been performed and that safety-related equipment will remain in its associated emergency mode following reset. The conclusions of the applicant review are:

(1) All equipment receiving an ESF actuation signal directly and not through the emergency diesel sequencer will remain in the emergency mode. After the equipment receives an ESF signal, it is driven to its emergency position. The ESF signal can be reset, and the equipment will remain in the emergency mode.

(2) To change the equipment from its emergency position, the ESF signal must be reset and the equipment control switch must be operated.

(3) All equipment receiving a loss-of-offsite-power (LOP) actuation signal via the sequencer will go to its emergency position and remain there as in Items (1) and (2) above, except the quench spray and recirculation spray pump motors. The reason for this is that the SIS cannot be reset until after a time delay which ensures that load sequenced by an SIS will have started; however the CDA signal can be reset at any time. If the CDA signal is reset before the quench spray and recirculation spray pumps are actuated by the sequencer after a LOP, then the quench spray and recirculation spray pumps will not start. Resetting the CDS signal will not stop the motors after a CDA signal is received and the quench spray or recirculation spray pump motors start. The pump motors can be stopped with their control switch if the CDA signal is not present. If the CDA output signal is reset and blocked before the pump motors are actuated, then this is treated as a bypassed or inoperable status and annunciated as part of the RG 1.47 alarms.

The staff finds that the design is consistent with the intent of the bulletin. The bulletin requires a confirmatory test to verify the conclusions of this review. This is a confirmatory item subject to the applicant's commitment to perform this test.

7.3.3.6 Containment Isolation for the Main Steamlines to the Turbine of the Auxiliary Feedwater Pump

GDC 57 requires that each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary nor connected directly to the containment atmosphere shall have at least one containment isolation valve that shall be either automatic, or locked closed, or capable of remote manual operation. The main steamlines to the AFW pump turbine have a motor-operated check stop valve in parallel with an air-operated bypass valve, both of which are remote manually operated. The staff expressed a concern that the bypass valves (AOV84A, B, and D) are not supplied power from a Class 1E power source and, therefore, isolation of the bypass valves cannot be ensured. By a letter dated April 2, 1984, the applicant stated that the 1/4-in. bypass line and bypass valves (AOV84 A, B, and D) around the stop check valves will be eliminated because prewarming the steam supply piping during testing of the turbine-driven AFW pump is not required. The staff considers this matter closed.

7.3.3.7 Letdown Line Relief Valve

The staff raised a concern that the relief valve located on the letdown line would relieve primary coolant to the pressurizer relief tank in the event the isolation valve inside containment did not close on a containment isolation signal or if the outside containment isolation valve failed closed. By a letter dated April 2, 1984, the applicant addressed the staff's concern by presenting various failure mode analyses. The analyses show that the containment isolation is accomplished, there is sufficient instrumentation to detect the flow into the pressurizer relief tank, the core integrity is maintained, and 10 CFR 50, Appendix K, limits are not exceeded. The staff finds that this concern is resolved.

7.3.3.8 Control Building Isolation Reset

The staff raised a concern on the design of the reset/override features used for control building isolation signals. Because the design for this safety function is based on one-out-of-two logic for some of the initiating conditions, single failures in instrument channels associated with these functions may result in system actuation. The use of the reset/override feature was designed so that the override of one initiating signal would defeat system initiation by all other initiating signals. In response to this concern, the applicant modified the design so that the use of the reset/override features, which blocks an initiating signal for one condition, will not defeat system initiation by other initiating signals. Therefore, on the basis of this modification to the design of the control building isolation reset/override features, the staff finds that the design is acceptable. This is a confirmatory item subject to revision of Sheet 8 of FSAR Figure 7.2.1.

7.3.3.9 Power Lockout Feature for Certain Motor-Operated Valves

The design of the control circuits for some motor-operated valves includes a power lockout feature. The power lockout is used to preclude single failures that could result in an inadvertent change in valve position. The power lockout feature consists of an additional set of contactors that interrupts power to the valve motor and is controlled by manual switches located on the rear panel of the main control board. The staff raised a concern that when the power lockout feature is used, a single failure could result in the pickup and sealin of the contactors used for normal valve control and that this condition would not be detectable. Further, this condition could occur if an attempt were made to change the position of the valve by the valve control switch. Under these conditions single failures in the power lockout circuits could result in an inadvertent change in valva position. The applicant proposed a modification of the design that uses an auxiliary contact of the power lockout contactors to deenergize the normal contact circuit. The staff finds the proposed modification acceptable. This is a confirmatory item subject to documentation of the drawing changes.

The modifications of the power lockout feature will be implemented for the following motor-operated valves:

Valve No.	Function	Drawing No.
3SIH*MV 8806	SI pumps suction/RWST	ESK-GMF
3SIL*MV 8840	RHR pumps/hot leg	ESK-GNM
3SIH*MV 8802A	SI pump disch/hot leg	ESK-GMR
3SIH*MV 8802B	SI pump disch/hot leg	ESK-GMS
3SIH*MV 8835	SI pumps disch/cold leg	ESK-GML
3SIL*MV 8809A	RHR pump disch/cold leg	ESK-GME
3SIL*MV 8809B	RHR pump disch/cold leg	ESK-GNA
3SIH*MV 8813	SI pumps recir/RWST	ESK-GMN
3RHS*MV 8716A	RHR pump disch cross over/hot/cold leg	ESK-GNJ
3RHS*MV 8716B	RHR pump disch cross over/hot/cold leg	ESK-GNK
3SIH*MV 8821A	SI pump disch cross over/hot/cold leg	ESK-GMJ
3SIH*MV 8821B	SI pump disch cross over/hot/cold leg	ESK-GMK

7.3.3.10 Failure Modes and Effects Analyses of ESFAS

The applicant referred to the Westinghouse Topical Report WCAP-8584, "Failure Mode and Effects Analysis (FMEA) of the Engineered Safety Feature Actuation System," for ESF systems equipment (FMEA) within the nuclear steam supply system (NSSS) scope of supply. For balance-of-plant (BOP) equipment, fault tree analyses, based on actual wiring diagrams and components of the plant, were performed. The applicant concluded that the single-failure criterion of IEEE Std. 279 requirements was met for the Class 1E instrumentation and control portions of the safety-related systems.

Because the FMEA for the NSSS was performed using assumptions on the BOP design, the staff requested the applicant to confirm that the interface requirements of Appendices B and C of WCAP-8584 are met. The applicant confirmed that the BOP design complies with the interface requirements of Appendices B and C of WCAP-8584. This is a confirmatory item subject to documentation in the FSAR.

7.3.3.11 Non-Class 1E Control Signals to Class 1E Control Circuits

The staff requested the applicant to provide a list of non-Class 1E control signals that are used as inputs to Class 1E control circuits and assess their effects on the safety systems. By a letter dated May 4, 1984, the applicant provided a list of non-Class 1E signals to Class 1E circuits. The applicant stated that these non-Class 1E signals are either bypassed by the ESF actuation signal or the non-Class 1E signal can only act to the safe direction and therefore will not degrade safety systems. This is a confirmatory item subject to staff's review of all the related electrical drawings, which are not available at the present time.

7.3.3.12 Isolators Used in the BOP Design for Isolation Between Safety- and Nonsafety-Related Systems

At Millstone Unit 3, multiplexers are used for information processing. Portions of the radiation monitoring system are safety related and use safety-related microprocessors that interface with the nonsafety-related radiation monitoring computer via qualified isolators. The staff requested additional information on the qualification of the isolators used for the radiation monitoring system. By a letter dated May 4, 1984, the applicant provided a Kaman Instrumentation Company test report entitled "Qualification of the Safety-Related Monitoring System (SRMS) Isolation Module to IEEE Std. 323-1974 and IEEE Std. 334-1975." The test results indicate that the SRMS isolation module performed satisfactorily when tested before and after the simulated aging and design-basis event conditions. The staff finds that the qualification of the isolators used for the radiation monitoring system is acceptable.

7.3.3.13 Sequencer Deficiency Report

On August 19, 1983, the Vitro Laboratories, the manufacturer of the Millstone emergency power loading sequencer, filed a 10 CFR 21 deficiency report. The report indicated that the design of the auto test circuitry does not permit the proper output in response to loss-of-coolant accident (LOCA) events in some circumstances. Specifically, one reset function was omitted from the input LOCA time delay. The result is that LOCA events occurring during the portion of the auto test cycle will not actuate some output relays. The applicant has also filed a 10 CFR 21 report to note this deficiency. This item is a confirmatory item subject to implementation of the required corrective action.

7.3.3.14 BOP Instrumentation and Control System Testing Capability

FSAR Sections 7.2.2.2.3 and 7.3.2.2.5 describe the capability for testing the reactor trip system and the engineered safety features (ESF) system. Most of the descriptions are based on NSSS scope of supply equipment. It is not clear whether all the BOP instrumentation and control systems satisfy the same criteria. The staff cited an example of the refueling water storage tank (RWST) level measurement, which is a BOP design. The low-low loop signal from one out of two level switches will automatically stop the residual heat removal pump. The empty tank signal from one out of two level switches will automatically stop the quench spray pumps. The testing of these actuation logic circuits is not discussed in the FSAR, and they are not tested by the same method as NSSS ESF instrument systems. The staff requested that the applicant perform a thorough evaluation of the BOP safety-related instrumentation and control systems with respect to testing capabilities, identify any instrument channels that cannot be tested as described in Sections 7.2.2.2.3 and 7.3.2.2.5, and justify that the design is in conformance with the testing requirements of GDC 21. By a letter dated April 2, 1984, the applicant provided a draft response to address each BOP safety-related instrumentation and control system with respect to testing capabilities and its conformance with the testing requirements of GDC 21. On the basis of its audit review, the staff finds that there is reasonable assurance that the BOP designs are in conformance with GDC 21. This is a confirmatory item subject to documentation in the FSAR.

7.3.4 Evaluation Conclusion

The review of the instrumentation and control aspects of the ESF systems included the ESFAS and the ESF control systems. The ESFAS detects a plant condition requiring the operation of an ESF system and/or EAS system and initiates operation of these systems. The ESF control systems regulate the operation of the ESF systems following automatic initiation by the protection system or manual initiation by the plant operator.

The staff concludes that the ESFAS and the ESF control systems are acceptable and meet the relevant requirements of GDC 2, 4, 20 through 24, 34, 35, 38, and 41 and 10 CFR 50.55a(h).

On the basis of its audit review of the system design for conformance to the SRP guidelines, the staff finds that there is reasonable assurance that systems conform fully to the guidelines applicable to these systems.

The staff's review has included the identification of those systems and components for the ESFAS and ESF control systems that are designed to survive the effects of earthquakes, other natural phenomena, abnormal environments, and missiles. On the basis of its review, the staff concludes that the applicant has identified those systems and components consistent with the design bases for those systems. Sections 3.10 and 3.11 of this SER discuss the qualification programs to demonstrate the capability of these systems and components to survive these events. Therefore, the staff finds that the identification of these systems and components satisfies this aspect of GDC 2 and 4.

On the basis of its review, the staff concludes that the ESFAS conforms to the design-basis requirements of IEEE Std. 279 and that the system includes the provision to sense accident conditions and anticipated operational occurrences to initiate the operation of ESF and EAS systems consistent with the accident analysis presented in Chapter 15 of the FSAR. Therefore, the staff finds that the ESFAS satisfies the requirements of GDC 20.

The ESFAS conforms to the guidelines for periodic testing in RG 1.22 and IEEE Std. 338, as supplemented by RG 1.118. The bypassed and inoperable status indication conforms to the guidelines of RG 1.47. The ESFAS conforms to the guidelines on the application of the single-failure criterion in IEEE Std. 379, as supplemented by RG 1.53. On the basis of its review, the staff concludes that the ESFAS meets the criteria of IEEE Std. 279 with regard to the system's reliability and testability. Therefore, the staff finds that the ESFAS satisfies the requirement of GDC 21.

The ESFAS conforms to the guidelines in IEEE Std. 384, as supplemented by RG 1.75, for the protection system independence. On the basis of its review, the staff concludes that the ESFAS satisfies the requirement of IEEE Std. 279 with regard to the system's independence. Therefore, the staff finds that the ESFAS satisfies the requirement of GDC 22.

On the basis of its review of the analysis for the ESFAS, the staff concludes that the system is designed with due consideration of safe failure modes if conditions such as disconnection of the system, loss of energy, or postulated adverse environment are experienced. Therefore, the staff finds that the ESFAS satisfies the requirements of GDC 23.

On the basis of its review of the interfaces between the ESFAS and plant operating control systems, the staff concludes that the system satisfies the requirements of IEEE Std. 279 with regard to control and protection system interactions. Therefore, the staff finds that the ESFAS satisfies the requirement of GDC 24.

The staff's conclusions noted above are based on the requirements of IEEE Std. 279 with respect to the design of the ESFAS. Therefore, it finds that the ESFAS satisfies the requirement of 10 CFR 50.55a(h) with regard to IEEE Std. 279.

The staff's review of the ESF control systems included conformance to the requirements for testability, operability with onsite and offsite electrical power, and single failures consistent with the GDC applicable to these ESF systems. The staff concludes that the ESF control systems are testable and are operable on either onsite or offsite power (assuming only one source is available) and that the controls associated with redundant ESF systems are independent and satisfy the single-failure criterion. Therefore, they meet the relevant requirements of GDC 34, 35, 38, and 41.

The staff in its review of the ESFAS and ESF control systems examined the dependence of these systems on the availability of essential auxiliary support (EAS) systems. On the basis of its review and coordination with those having primary review responsibility of the EAS systems, the staff concludes that the design of the ESFAS and ESF control systems is compatible with the functional performance requirements of EAS systems. Therefore, the staff finds the interfaces between the design of the ESFAS and ESF control systems and the design of the EAS systems acceptable.

7.4 Systems Required for Safe Shutdown

7.4.1 Description

This section describes the equipment and associated controls and instrumentation of systems required for safe shutdown. It also includes controls and instrumentation located outside the main control room that enable safe shutdown of the plant if the main control room is evacuated.

7.4.1.1 Safe Shutdown System

The systems required for safe shutdown are those required to (1) control the reactor coolant system temperature and pressure, (2) borate the reactor coolant, and (3) provide adequate residual heat removal. There are two kinds of shutdown conditions: hot standby and cold shutdown. Hot standby is a stable condition of the plant achieved shortly after a programmed or emergency shutdown of the plant. Cold shutdown is a stable condition of the plant achieved after the residual heat removal process has brought the primary coolant temperature below 200°F. For either case, the following systems are required for achieving and maintaining the safe shutdown condition.

(1) emergency Class 1E electrical power supply systems
(2) auxiliary feedwater system
(3) residual heat removal system
(4) boration and reactor coolant inventory control system
(5) reactor coolant pressure relief system
(6) steam generator power-operated relief valves (PORVs) and bypass valves
(7) component cooling water system
(8) service water system
(9) safety-related heating, ventilation, and air conditioning systems

To achieve and maintain safe shutdown, the reactor and the turbine are tripped. Automatic protection and control system functions are discussed in Sections 7.2 and 7.3. The controls and the indicators for all of the equipment listed above are provided in the main control room. In addition, an auxiliary shutdown panel is provided that allows the plant to be maintained in a hot standby condition or taken to cold shutdown should the main control room become uninhabitable.

The safe shutdown design basis for Millstone Unit 3 is cold shutdown. The plant can be taken from no-load temperature and pressure to residual heat removal (RHR) system initiation within 36 hours following any condition II, III, or IV events using only safety-grade systems, with or without offsite power, with a single failure and with limited operator action outside the control room. Safe shutdown includes boration and depressurization of the primary coolant system. During the first phase of cooldown, heat removal is accomplished by means of the steam generator power-operated relief valves (PORVs) and the auxiliary feedwater system. Boration is accomplished by the charging pumps injecting borated water into the reactor coolant system. Gravity drain lines are connected from the boric acid tanks to the charging pumps header. Control of the boration rate is accomplished by throttling valves in the flow paths from the charging pumps to the high head safety injection lines. A parallel and series arrangement of Class 1E solenoid valves is provided for the reactor vessel head letdown path. Depressurization of the reactor coolant system is accomplished by the solenoid operated pressurizer PORVs. When the reactor coolant system temperature and pressure are reduced to about 350°F and 425 psig, RHR is initiated and cooldown proceeds to the normal plant cold shutdown condition.

7.4.2 Specific Findings

7.4.2.1 Turbine-Driven Auxiliary Feedwater Pump Control Transfer

During its drawing review, the staff raised a concern on turbine-driven auxiliary feedwater pump control transfer design. Whenever the control for the turbine-driven auxiliary feedwater pump is transferred from the main control room to the auxiliary shutdown panel, the turbine-driven feedwater pump starts automatically. The applicant has proposed a design modification for the control circuitry to prevent inadvertent starting of the pump during the transfer. The staff finds the modified design acceptable.

7.4.2.2 Auxiliary Feedwater Control

The staff's review of the auxiliary feedwater system (AFWS) included the following:

(1) automatic initiation (discussed in Section 7.3)

(2) capability of controlling flows to establish and maintain steam generator level

(3) capability of controlling the steam generator pressure

(4) capability of isolating a faulted steam generator resulting from feedwater or steamline breaks

(5) capability for post-trip control from auxiliary shutdown panel

The auxiliary feedwater flow to each steam generator is through the normally opened control valves. Each control valve can be manually adjusted from the control room as dictated by the steam generator water level and auxiliary feedwater flow rate. The control valves also can be manually adjusted from the auxiliary shutdown panel. The auxiliary feedwater is fed to the steam generators through a connection downstream of the main feedwater stop-check valves. The auxiliary feedwater has sufficient water supply to hold the unit at hot standby for up to 10 hours. The reactor coolant temperature can be reduced to 350°F in 6 hours, at which time the residual heat removal system will be initiated.

During plant cooldown, the main steam PORVs are automatically controlled by steamline pressure. Manual control of the PORVs is provided to control the steam generator pressure to permit cooldown from the main control board or the auxiliary shutdown panel. Auxiliary feedwater flow to the steam generators is limited by flow venturis located in each auxiliary feedwater line. These venturis are sized to restrict the flow to a depressurized steam generator. Two isolation valves are provided in each of the auxiliary feedwater supply lines. One valve is powered by the train A power source; the other valve is powered by the train B power source. The isolation valves can be operated either from the main control board or the auxiliary shutdown panel.

Indications are provided at the auxiliary shutdown panel for steam generator level and pressure, auxiliary feedwater flow, and demineralized water tank level. The capability is provided to control the auxiliary feedwater pumps and to isolate a depressurized loop as well as for post-trip control of the auxiliary feedwater system at the auxiliary shutdown panel. On the basis of its review, the staff finds that the auxiliary feedwater control system design is acceptable.

7.4.2.3 Remote Shutdown Capability

GDC 19 requires that equipment at appropriate locations outside the control room be provided to achieve a safe shutdown of the reactor. SRP Section 7.4 provides guidance on conformance to the GDC 19 requirements. The design should provide redundant safety-grade capability to achieve and maintain safe shutdown from a location or locations remote from the control room, assuming no fire damage to any required systems and equipment and assuming no accident has occurred. The remote shutdown station equipment should be capable of maintaining functional operability under all service conditions postulated to occur, including the seismic event. The remote shutdown stations and the equipment used to maintain safe shutdown should be designed to accommodate a single failure.

By a letter dated April 2, 1984, the applicant stated that the design bases for the remote shutdown station are:

(1) Redundant safety-grade remote shutdown capability is provided.

(2) Two transfer switch panels (TSPs) and one auxiliary shutdown panel (ASP) are located in three separated fire zones.

(3) A communication network is provided from the ASP to important plant locations where the safe shutdown equipment is located.

(4) No jumper is required to transfer control from the main control room to the auxiliary shutdown panel.

(5) The design is such that transfer of equipment control from the main control room to the auxiliary shutdown panel will not change the status of the equipment.

(6) Loss of offsite power will not negate shutdown capability from the remote shutdown area.

(7) Access to remote shutdown areas is under administrative control. Whenever the ASP or TSP cabinet door opens, an annunciator alerts the operator in the main control room. Also, each transfer switch mounted on the TSP is annunciated in the main control room whenever a control is transferred.

(8) The following design criteria are applicable to the instrumentation and control devices located on the ASP:

(a) ANSI C37.90, 1978
(b) IEEE Std. 279, 1971
(c) IEEE Std. 308, 1974
(d) IEEE Std. 323, 1974
(e) IEEE Std. 344, 1975
(f) IEEE Std. 338, 1971
(g) IEEE Std. 379, 1972
(h) IEEE Std. 384, 1974
(i) IEEE Std. 420, 1974
(j) NUREG-0588, 1979
(k) RG 1.75, 1974
(l) 10 CFR 50, Appendix R

The staff has reviewed the control schematic and the panel layout drawings of the ASP and TSP, and finds that the remote shutdown system design is acceptable.

7.4.2.4 Testing for Remote Shutdown Operation

During the review process, a concern was raised by the staff regarding the remote shutdown capability and the need for a test to verify design adequacy. The applicant stated that emergency procedures will be prepared to include remote shutdown and a test will be conducted during startup testing to confirm the capability for remote shutdown. The test description is outlined in FSAR Table 14.2-2, Item 25. The staff finds the applicant's commitment for remote shutdown operation testing acceptable.

7.4.3 Evaluation Conclusion

The review of systems required for safe shutdown included the sensors, circuitry, redundancy features, and actuated devices that prevent the reactor from returning to criticality and provide means for adequate residual heat removal. The review included the FSAR descriptive information, logic diagrams, single-line diagrams, schematic diagrams, and piping and instrumentation diagrams.

On the basis of its audit review of the system designs for conformance to the guidelines, the staff finds that there is reasonable assurance that the systems conform to the applicable guidelines.

The staff's review has included the identification of those systems and components required for safe shutdown that are designed to survive the effects of earthquakes, other natural phenomena, abnormal environments, and missiles.

On the basis of its review, the staff concludes that the applicant has identified those systems and components consistent with the design bases for the systems. Sections 3.10 and 3.11 of this SER address the qualification programs to demonstrate the capability of these systems and components to survive applicable events. Therefore, the staff finds that the identification of these systems and components satisfies this aspect of GDC 2 and 4.

The staff concludes that instrumentation and controls have been provided to maintain variables and systems that can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems within prescribed operating ranges during plant shutdown. Therefore, the staff finds that the systems required for safe shutdown satisfy the requirements of GDC 13.

Instrumentation and controls have been provided within the control room to allow actions to be taken to maintain the nuclear power unit in a safe condition during shutdown including a shutdown following an accident. Equipment at appropriate locations outside the control room has been provided with (1) a design capability for prompt hot shutdown of the reactor, including instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

The staff in its review of the instrumentation and controls required for safe shutdown has examined the dependence of these systems on the availability of EAS systems. On the basis of its review, the staff concludes that the designs of EAS systems are compatible with the functional performance requirements of the systems reviewed in this section. Therefore, it finds the interfaces between the designs of safe shutdown systems and the design of EAS systems acceptable.

The staff review of the instrumentation and control systems required for safe shutdown included a review of the conformance to the requirements for testability, operability with onsite and offsite electrical power, and single failures consistent with the GDC applicable to safe shutdown systems. The staff concludes that, in general, these systems are testable and are operable on either onsite or offsite electrical power and that the controls associated with redundant safe shutdown systems are independent and satisfy the requirements of the single-failure criterion.

In summary, the staff concludes that the systems required for safe shutdown meet GDC 2, 4, 13, and 19 and RGs 1.47, 1.53, and 1.62, and therefore are acceptable.

7.5 Information Systems Important to Safety

7.5.1 Description

The applicant has conducted an analysis to identify the appropriate variables for the operator to monitor conditions in the reactor coolant system, the secondary heat removal system, the containment system, the engineered safety features systems, and the safe shutdown systems. The safety-related display instrumentation system provides the information necessary for the operator to perform the required manual safety functions following a reactor trip. It provides information for all operating conditions, including anticipated operational occurrences and accidents and postaccident conditions.

The instrumentation identified in FSAR Table 7.5-1 includes the following information for each variable identified:

(1) instrument range
(2) environmental qualification
(3) seismic qualification
(4) display methodology
(5) type and category (according to the definition in RG 1.97, Rev. 2)
(6) schedule for implementation

The qualification for these instruments is discussed in Sections 3.10 and 3.11 of this report.

7.5.2 Specific Findings

7.5.2.1 Loss of Non-Class 1E Instrumentation and Control Power System Bus During Operation (IE Bulletin 79-27)

The staff requested that the applicant review the adequacy of emergency operating procedures to be used by control room operators to attain safe shutdown on loss of any Class 1E or non-Class 1E bus supplying power to safety- or nonsafety-related instrument and control systems. This issue was addressed for operating reactors through IE Bulletin 79-27. In FSAR Amendment 5, the applicant responded that Millstone Unit 3 can achieve a cold shutdown condition without the use of any non-Class 1E power. All the equipment required to achieve a cold shutdown is redundant and is powered from redundant Class 1E buses, which satisfies the single-failure criterion. However, the staff pointed out that loss of a single instrument bus could affect the interlock circuits to isolate both trains of the residual heat removal (RHR) system; therefore, the applicant's response did not adequately address the concerns identified in IE Bulletin 79-27. By a letter dated April 2, 1984, the applicant stated that a single failure could affect the RHR system operation. However, plant procedures will be developed to allow for the manual opening of the isolation valve outside the containment. The staff's evaluation of the RHR system isolation valve interlocks is given in Section 7.6.2.1 of this report. The staff finds that the applicant's response to the IE Bulletin 79-27 concern is acceptable.

7.5.2.2 Bypass and Inoperable Status Panel

A system level bypass and inoperable indicator is provided for each protection system. There is a separate indicator for each train. The indicator is operated automatically when

(1) The action is deliberate.
(2) The action is expected to occur more often than once a year.
(3) The action is not a designed operational bypass.
(4) The action renders the system inoperable, not merely potentially inoperable.

Each bypass indicator can be manually operated for an event that renders a safety system inoperable but does not automatically operate the system bypass indicator. The bypass indicators are accompanied by an audible alarm. The indication system is isolated from the safety system. No fault in the indication system can impair the safety system's performance of the protective function. The bypass indication and annunciation can be tested during normal plant operation.

The staff has reviewed the design drawings and finds that the bypass and inoperable status indication system is in conformance with RG 1.47 and BTP ICSB-21 and is, therefore, acceptable.

7.5.2.3 NUREG-0737, Item II.D.3, Direct Indication of Relief and Safety Valve Positions

The two pressurizer power-operated relief valves (PORVs) are operated automatically or by remote manual control. Each valve is provided with positive open/closed indication lights in the control room. The three safety valves are also provided with positive open/closed indication lights. The temperature in each of the safety valve and PORV discharge lines is measured and indicated in the control room. An increase in a discharge line temperature is an indication of leakage or relief through the associated valve. High temperature will be alarmed in the control room. The valves' position-indicating limit switches are seismically and environmentally qualified. The staff finds that the design is in conformance with the Action Plan guidelines and is, therefore, acceptable.

7.5.2.4 NUREG-0737, Item II.F.1, Accident Monitoring Instrumentation, Positions (4), (5), and (6)

Positions (4), (5), and (6) of this Action Plan item require installation of the extended range containment pressure monitors, containment water level monitors, and containment hydrogen concentration monitors. Table 7.5-1 of the FSAR indicated that the information on these parameters is as follows:

(1) containment pressure (extended range)

- (a) The instruments are environmentally and seismically qualified.
- (b) The instrument range extended from 0 to 200 psia.
- (c) Two channels are provided.
- (d) Two indicators and one dual recorder are provided in the control room.

(2) containment water level (wide range)

- (a) The instruments are environmentally and seismically qualified.
- (b) The instrument range extended from 0 to 1,500,000 gal.
- (c) Two indicators are provided.
- (d) Two indicators and one recorder are provided in the control room.

(3) containment hydrogen monitor

- (a) The instruments are environmentally and seismically qualified.
- (b) The instrument range extended from 0% to 10%.
- (c) Two channels are provided.
- (d) Two indicators and one recorder are provided.

The information listed above satisfies the requirements of NUREG-0737, Item II.F.1, Positions (4), (5), and (6), except for the instrument accuracy requirement. This information should be provided and justified to be adequate for the intended function. This is a confirmatory item.

7.5.2.5 NUREG-0737, Item II.F.2, Instrumentation for Detection of Inadequate Core Cooling

The applicant has not described his design for this item. The staff's evaluation is given in Section 4.4 of this report.

7.5.2.6 Instrumentation for Monitoring Postaccident Conditions - RG 1.97, Revision 2, Requirements

Generic Letter 82-33 included additional clarification regarding RG 1.97, Revision 2, requirements for emergency response capability. On October 3, 1983, the staff requested specific information on conformance with RG 1.97, Revision 2 (Q420.6). By letters dated December 16, 1983, and January 13, 1984, the applicant provided the responses to Question Q420.6. Deviations from the guidance in RG 1.97 were identified. Until the staff completes its review of Millstone Unit 3 design to determine compliance with recommendations in RG 1.97, Revision 2, recommendations, a license condition will be imposed requiring the satisfactory resolution of all the review findings.

7.5.3 Evaluation Conclusion

The information systems important to safety provide the operator with information on the status of the plant to allow manual safety actions to be performed when necessary. The scope of review included tables of system variables and component status to be indicated, function control diagrams, electrical and physical layout drawings, and descriptive information. The review included the applicable acceptance criteria and guidelines and design bases, including those for indication of bypasses or inoperable safety-related systems. The review also included the applicable acceptance criteria and guidelines and design bases, including those for indication of bypassed or inoperable safety-related systems. In addition, the review included the applicant's analyses of how the design of information systems conforms to the SRP. The staff concludes that the information systems important to safety are acceptable and meet GDC 2, 4, 13, and 19.

On the basis of its audit review of the system design for conformance to the guidelines, the staff finds that there is reasonable assurance that these systems conform to the guidelines applicable to these systems.

The staff review included the identification of those systems and components for the information systems that are designed to survive the effects of earthquakes, other natural phenomena, abnormal environments, and missiles. On the basis of its review, the staff concludes that the applicant has identified those systems and components consistent with the design basis for those systems. Sections 3.10 and 3.11 of this SER address the qualification programs to demonstrate the capability of these systems and components to survive these events. Therefore, the staff finds that the identification of these systems and components satisfies this aspect of GDC 2 and 4.

The staff concludes that the information systems important to safety, including the accident monitoring instrumentation, are consistent with the plant safety analysis and show substantial compliance with RG 1.97, Revision 2. Therefore, the staff finds that the information systems satisfy the requirements of GDC 13 for monitoring variables and systems over their anticipated ranges for normal operation, anticipated operational occurrences, and accident conditions. Further, the staff finds that conformance to GDC 13 and the applicable guidelines satisfies the requirements of GDC 19 with respect to information systems provided in the control room from which actions can be taken to operate the unit safely under normal conditions and to maintain it in a safe condition under accident conditions.

7.6 Interlock Systems Important to Safety

7.6.1 Description

This section addresses the safety-related interlocks that

(1) prevent the overpressurization of low-pressure systems

(2) prevent the overpressurization of the primary coolant system during low-temperature operation

(3) ensure the availability of emergency core cooling system (ECCS) accumulators

(4) prevent an accidental startup of an isolated reactor coolant loop

The objective of the review was to confirm that design considerations such as redundancy, independence, single failures, qualification, bypasses, status indication, and testing are consistent with the design bases of these safety-related systems.

7.6.2 Specific Findings

7.6.2.1 Residual Heat Removal System Isolation Valves Interlock

The residual heat removal (RHR) system isolation valve interlocks are provided to prevent overpressurization of the RHR system. There are three motor-operated valves in series in each of the two RHR pump suction lines from the reactor coolant system (RCS) hot legs. The two valves located close to the containment walls, one outside and one inside the containment, are provided with interlocks. The third valve inside the containment is not interlocked and is operated by a keylock control switch, which is under administrative control.

Two pressure transmitters powered from separate safety power trains are used for the isolation valve interlocks. Each valve is interlocked to prevent it from opening if RCS pressure is greater than 425 psig and to automatically close it if RCS pressure exceeds 700 psig. Valve position indication is provided in the control room and at the auxiliary shutdown panel for each valve.

The redundant valve interlock design includes independence, separation, and diversity. The staff finds that the design satisfies BTP ICSB-3, "Isolation of Low Pressure Systems From the High Pressure Reactor Coolant System." The plant procedures provide the capability to manually open the isolation valve outside the containment. The staff finds that the design of the RHR system isolation valves is acceptable.

7.6.2.2 Isolation of Low-Pressure Systems From the High-Pressure Reactor Coolant System

GDC 15 requires that the reactor coolant system (RCS) and the associated auxiliary, control, and protection system shall be designed with sufficient margin to ensure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences. The staff requested that the applicant identify all points of interface between the RCS and the systems whose design pressure is less than that of the RCS and discuss, for each interface, the degree of conformance with the requirements of BTP ICSB-3 and how the associated interlock circuits conform to the requirements of IEEE Std. 279.

By a letter dated April 2, 1984, the applicant identified the following interfaces of the low-pressure systems with the reactor coolant system:

(1) residual heat removal system suction lines
(2) reactor coolant system letdown line
(3) excess letdown lines
(4) sample system connections
(5) charging line connection
(6) ECCS discharge line connections

The applicant described the isolation provisions between the RCS and each low-pressure system. The staff finds that the design satisfies BTP ICSB-3 and is, therefore, acceptable.

7.6.2.3 Reactor Coolant System Overpressure Protection During Low-Temperature Operation

The pressurizer power-operated relief valves (PORVs) are used to provide overpressure protection of the RCS during low-temperature operation. The PORVs are automatically opened when RCS pressure exceeds a programmed setpoint based on RCS temperature. During normal operation this system is manually blocked to preclude single failure resulting in inadvertent operation of a PORV. The wide-range RCS temperature measurements are used to provide the programmed overpressure setpoint. One train uses an auctioneered lowest hot-leg-temperature signal, and the other uses an auctioneered lowest cold-leg-temperature signal. During a plant shutdown a low RCS temperature alarm alerts the operator to arm the system for low-temperature operation of the RCS. When the system is armed, an alarm will occur if the block valve upstream of the PORV is not fully open. Also an alarm is provided to alert the operator when RCS pressure approaches the programmed setpoint for PORV operation. The staff reviewed the electrical schematics for pressurizer PORV control and the block valve control for all modes of operation. The staff finds the design acceptable.

7.6.2.4 Accumulator Isolation Valve Interlock

A motor-operated isolation valve is provided at each accumulator outlet. These valves are normally open during plant operation and closed during plant shutdown. To prevent an inadvertent closing or opening of these valves, power is locked out from the valve motor and its control circuit. Administrative control is required to ensure that power is restored to the valve control circuit during plant shutdown and startup. These valves are interlocked so that they

(1) open automatically on receipt of a safety injection signal

(2) open automatically whenever the RCS pressure is above the safety injection unblock (P-11) setpoint

(3) cannot be closed as long as the safety injection signal is present

Administrative controls require the performance of a periodic check valve leakage test. The interlock will ensure that the safety function is maintained during testing.

The accumulator motor-operated valves are provided with indicating lights located at the control switches on the main control board and auxiliary shutdown panel. These lights are actuated by the valve motor operator limit switches. Another set of indicating lights is provided at the safeguard status panel. The status panel lights are actuated by a steam mounted valve position limit switch, which is independent from the motor operator limit switches. The power source for indicating lights on the control panel is from a 120-Vac Class 1E instrument power bus that is independent from the valve motor control power. The power source for the status panel is from a separate power supply. Therefore, the power lockout will not affect either set of the indication lights. An alarm will sound when a limit switch senses that the valve is not fully open. The staff finds that the design satisfies the BTP ICSB-4 and is, therefore, acceptable.

7.6.3 Evaluation Conclusion

The staff concludes that the design of the interlock systems important to safety is acceptable and meets the relevant requirements of GDC 2 and 4.

The review of the interlock systems important to safety included the interlocks to prevent overpressurization of low-pressure systems when they are connected to the primary coolant system. The staff position with regard to this interlock system is BTP ICSB-3, "Isolation of Low Pressure Systems From the High Pressure Reactor Coolant System." On the basis of its review, the staff concludes that the design of this system satisfies the staff's guidelines.

The staff review included the interlock for the ECCS accumulator isolation valve. The staff's position with regard to this interlock system is BTP ICSB-4, "Requirements of Motor Operated Valves in the ECCS Accumulatory Lines." On the basis of its review, the staff concludes that these interlocks satisfy the staff's guidelines. On the basis of its review of the interlock systems important to safety, the staff concludes that the systems' design bases are consistent with the plant safety analysis and their importance to safety. Further, the staff concludes that the aspects of the design of those systems with respect to single failures, redundancy, independence, qualification, and testability are adequate to ensure that the functional performance requirements of these systems will be met and that they meet the applicable requirements of GDC 2 and 4.

7.7 Control Systems

7.7.1 Description

The plant control systems that are not relied on to perform safety functions but control plant processes that have an impact on plant safety are described in this section and include the following:

(1) reactor control system
(2) rod control system
(3) monitoring and indicating systems
(4) plant control interlocks
(5) pressurizer pressure control
(6) pressurizer level control
(7) steam generator water level control
(8) steam dump control
(9) incore instrumentation system
(10) boron concentration measurement system

The reactor control system enables the nuclear plant to follow load changes automatically, including a 10% step change in load or a 5% per minute rate of load change. The system maintains coolant average temperature following a load change with acceptable limits. The reactor control system controls the reactor coolant average temperature by regulation of control rod bank position. The core axial power distribution is controlled manually during load following maneuvers by changing the boron concentration in the reactor coolant system.

The rod control system provides for reactor power modulation by manual or automatic control of control rod banks in a preselected sequence. It displays control rod positions, alerts the operator in the event of control rod deviation exceeding a preset limit, and alerts the operator to inadequate shutdown margins resulting from excessive control rod insertion. The automatic rod control system is designed to maintain a programmed average temperature in the reactor coolant by regulating the reactivity within the core. The automatic rod control is performed between 15% and 100% of rated power. Power is supplied to rod drive mechanisms by two motor generator sets operating from two separate 480-V three-phase buses. Each generator is the synchronous type and is driven by a 200-hp induction motor. The ac power is distributed to the rod control power cabinets through the two series-connected reactor trip breakers. The reactor trip breakers are part of the safety system as described in Section 7.2 of this report.

The monitoring and indicating systems include:

(1) nuclear instrumentation monitoring

   (a) nuclear power level
   (b) axial flux imbalance
   (c) upper radial tilt
   (d) lower radial tilt

(2) rod position monitoring

(a) digital rod position indication
(b) demand position
(c) rod insertion limits
(d) rod deviation alarm and rod bottom alarm

The plant control interlocks prevent further withdrawal of the control rod banks either by a control system malfunction or an operator error. The interlocks are derived from nuclear instrument channels or reactor coolant overtemperature-overpower channels. The interlocks also limit automatic turbine load increases during a rapid-return-to-power transient (through the negative moderator coefficient). The interlock can be cleared by an increase in coolant temperature, which is accomplished by reducing the boron concentration in the coolant.

The reactor coolant pressure is controlled by using either the pressurizer heaters or spray plus PORV steam relief for large transients. The water inventory in the RCS is maintained by the chemical and volume control system. During normal plant operation, the charging flow varies to match the flow demanded of the pressurizer water level controller. The pressurizer water level is programmed as a function of coolant average temperature. During startup and shutdown operations, the charging flow is manually regulated to maintain pressurizer water level.

The steam generator level is programmed by a three-element feedwater controller, which regulates the feedwater valves by continuously comparing the feedwater flow signal, the water level signal, the programmed level setpoint, and the steam flow signal. During startup or low-power operation, a feed-forward control scheme uses steam generator level and nuclear power signals to position a bypass control valve, which is parallel with the main feedwater regulating valve.

The steam dump system is designed to accept a 50% load rejection without tripping the reactor. The system functions automatically by bypassing steam directly to the condenser and/or atmosphere to maintain the load on the primary system. The rod control system can then reduce the reactor coolant temperature to a new equilibrium value without causing overtemperature and/or overpressure conditions.

A demand signal for the load-rejection steam dump controller is generated if the difference between the reference reactor coolant average temperature (based on turbine impulse chamber pressure) and the measured reactor coolant average temperature exceeds a preset value.

The incore instrumentation system consists of chromelalumel thermocouples at fixed core outlet positions and movable miniature neutron detectors at selected fuel assemblies. The thermocouple readings are monitored by the plant computer. The movable detectors can perform flux mapping at various core quadrant locations to obtain a flux map for any region of the core. The data collection, calculation, and recording are performed by the plant computer.

The boron meter determines the relative concentration of boron in the sample fluid. The boron concentration measurement system is designed for use as an operating aid.

7.7.2 Specific Findings

7.7.2.1 Control System Failures Caused by Malfunctions of Common Power Source or Instrument Line

To provide assurance that the FSAR Chapter 15 analyses adequately bound events initiated by a single credible failure or malfunction, the staff asked the applicant to identify any power source or sensors that provide power or signals to two or more control functions and demonstrate that failures or malfunctions of these power sources or sensors will not result in consequences more severe than those of the Chapter 15 analyses or beyond the capability of the operator or the safety systems. By a letter dated April 11, 1984, the applicant provided a response to this concern. A detailed analysis of the effects of power source, sensor, and impulse line failure was performed for each of the following control systems:

(1) reactor control
(2) steam dump
(3) pressurizer pressure control
(4) pressurizer level control
(5) feedwater control

The applicant has provided a summary of the events resulting from each postulated failure and identified the specific Chapter 15 analysis that delineates the bounding consequences of the failure. The staff has reviewed the bases for the applicant's study and concludes that there is reasonable assurance that the consequences of single failures within the control systems are bounded by analyses in FSAR Chapter 15 and, therefore, are acceptable.

Unresolved Safety Issue A-47, "Safety Implications of Control Systems," will address control system design and the need for any control system design modifications. The applicant will be required to address any new guidance that may result from the resolution of the unresolved safety issue.

7.7.2.2 Control System Failure Caused by High-Energy-Line Breaks

Operating reactor licensees were informed by IE Information Notice 79-22 that if certain nonsafety-grade control equipment were subjected to the adverse environment of a high-energy-line break, this may impact the safety analyses and the adequacy of the protection functions performed by the safety-grade equipment. The staff has requested a review to determine whether the harsh environment associated with high-energy-line breaks might cause control system malfunction and result in a consequence more severe than those of the FSAR Chapter 15 analyses or beyond the capability of operators or safety systems.

By a letter dated May 4, 1984, the applicant provided a response to this concern. The applicant performed an analysis on four control systems that could potentially malfunction as a result of a high-energy-line break inside or outside containment. These control systems include

(1) steam generator power-operated relief valve (PORV) control
(2) pressurizer power-operated relief valve control

(3) main feedwater control
(4) automatic rod control

A review was made of the above four control systems for the environmental qualification of equipment. In the event of a high-energy-line break in the main steam valve building, the main steamline PORV could fail in the open or closed position as a result of failure of a nonsafety-related I/P converter, which modulates the PORV. If the PORV fails open, the safety-related motor-operated isolation valve, which is in series with the PORV, can be modulated from the main control board or from the auxiliary shutdown panel to control steam generator pressure. If the PORV fails closed, the safety-related motor-operated bypass valve, which is in parallel with the PORV, can be modulated from the main control board or from the auxiliary shutdown panel to control steam generator pressure. The staff finds that the consequence of main steamline PORV failure is acceptable.

Each feedwater control valve or bypass valve has a flow controller with a nonsafety-related I/P converter that modulates the associated valve. The I/P converters could fail as a result of a high-energy-line break outside the containment. The feedwater control and bypass valves could fail open or closed. However, the protection system would initiate a feedwater isolation signal to override the control signal and start the auxiliary feedwater system. Therefore, the feedwater control system failure is bounded by safety analysis.

The instrumentation for the pressurizer PORV control system and the automatic rod control system is fully qualified for an adverse environment. A steamline break inside or outside containment will not cause a malfunction within these systems.

The staff has reviewed the basis for the applicant's analysis and concludes that there is reasonable assurance that the consequences of a control system malfunction as a result of a steamline break inside or outside containment are bounded by analyses in FSAR Chapter 15 and, therefore, are acceptable.

7.7.2.3 Freeze Protection System

Safety-related systems requiring heat tracing are heated by circuits powered from two independent control panels (one primary, one backup). The power to the two control panels is from two separate safety train power sources. The control panels are not safety grade. The safety train power source is protected from this nonsafety service by an isolation transformer. The primary heat tracing is energized upon a low ambient temperature signal. As the ambient temperature continues to decline, the backup heat tracing will be automatically energized. A temperature sensor on the piping will alarm at the primary panel when it senses a temperature below the setpoint of the backup heat tracing. Should the temperature of the piping continue to drop, a second temperature sensor on the piping will alarm at the backup panel. Both alarms will cause an alarm to sound in the control room identifying trouble at the control panels.

All safety-related instrument-sensing lines with freeze protection are temperature monitored and alarmed. Because there are two separate heat tracing and monitoring systems, each system has an independent power source; a single failure in either of the two systems will not affect the capability of the other system. The staff finds that the design satisfies RG 1.151 and is, therefore, acceptable.

7.7.2.4 NUREG-0737, Item II.K.3.9, Proportional Integral Derivative (PID) Controller Modification

Westinghouse recommended that the derivative time constant in the pressurizer PORV PID controller be set to "off" to address this Action Plan item. This action removes the derivative action from the controller so that the actuation signal to this valve is no longer sensitive to the rate of change of pressurizer pressure. The applicant has implemented this recommendation. The staff finds that the applicant is in compliance with the Action Plan guidelines for this item.

7.7.3 Evaluation Conclusion

The control systems used for normal operation, which are not relied on to perform safety functions but which control plant processes having a significant impact on plant safety, have been reviewed. These control systems include the reactivity control systems and the control systems for the primary and secondary coolant systems.

The staff concludes that the control systems are acceptable and meet the relevant requirements of GDC 13 and 19.

On the basis of its review of the plant transient response to normal load changes and anticipated operational occurrences such as reactor trip, turbine trip, and upsets in the feedwater and steam bypass systems, the staff concludes that the control systems are capable of maintaining system variables within prescribed operating limits. Therefore, it finds that the control systems satisfy this aspect of GDC 13.

The staff review of control systems included features of these systems for both manual and automatic control of the process systems.

The staff concludes that the features for manual and automatic control facilitate the capability to maintain plant variables within prescribed operating limits. It finds that the control systems permit actions that can be taken to operate the plant safely during normal operation, including anticipated operational occurrences, and, therefore, the control systems satisfy GDC 19 with regard to normal plant operations.

The conclusions of the analyses of anticipated operational occurrences and accidents presented in Chapter 15 of the FSAR have been used to confirm that plant safety is not dependent on the response of the control systems. The staff concludes that failure of the systems themselves or as a consequence of the failure of a supporting system - such as power source - does not result in plant conditions more severe than those bounded by the analyses of anticipated operational occurrences.

8 ELECTRIC POWER SYSTEMS

8.1 General

The staff has reviewed the applicant's designs, design criteria, and design bases for the Millstone electric power systems in accordance with SRP Section 8.1, Table 8-1, "Acceptance Criteria and Guidelines for Electric Power Systems" (NUREG-0800). These acceptance criteria and guidelines include the applicable GDC and guidelines of branch technical positions, RGs, and NUREGs.

The following sections provide the staff's evaluation of the offsite and onsite electric power system design and how it meets the requirements of the above-cited acceptance criteria. The staff will also visit the site to view the installation and arrangements of electrical equipment and cables, to review confirmatory electric drawings, and to verify test results for the purpose of verifying the adequacy of the design and proper implementation of the design criteria. The confirmatory site visit will be completed before the license is issued, and if any problems are found, they will be addressed in a supplement to this SER.

The conclusions in the following sections are subject to acceptable implementation of design changes if any, that may be required as a result of the staff's site visit.

8.2 Offsite Electric Power System

The safety function of the offsite power system (assuming the onsite power system is not functioning) is to provide sufficient capacity and capability to ensure that the structures, systems, and components important to safety perform as intended. The objective of the staff review is to determine that the offsite power system satisfies the requirements of GDC 5, 17, and 18 and will perform its design function during all plant operating and accident conditions.

8.2.1 Compliance With GDC 5

The applicant has met the requirements of GDC 5 with respect to sharing of circuits of the preferred power system.

The following item addresses the problem area revealed during the staff review and its resolution.

8.2.1.1 Description and Analysis Demonstrating Compliance With GDC 5

An analysis with a description of design provisions demonstrating that the offsite power system meets the requirements of GDC 5 had not been presented in Section 8.2 of the FSAR in accordance with the guidelines of RG 1.70. By letter dated June 12, 1984, the applicant indicated that only the offsite power system switchyards are shared between Millstone Units 1, 2, and 3. This sharing is permitted by GDC 17; thus, the design meets the requirements of GDC 5 and 17 and is acceptable.

8.2.2 Compliance With GDC 17

The applicant has met (except as noted) the requirements of GDC 17 with respect to the offsite power system's (1) capacity and capability to permit functioning of structures, systems, and components important to safety; (2) provisions to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit or loss of power from the onsite electric power supplies; (3) physical independence of circuits; and (4) availability of circuits.

The following items address the problem areas revealed during the staff review and their resolution or status.

8.2.2.1 Physical Separation of Offsite Circuits Within a Common Right-of-Way

As described in Section 8.2.1 of the FSAR, there are four transmission lines between Millstone and Hunts Brook Junction that follow a common right-of-way. It is the staff position that no other transmission lines cross over these four lines and that the four lines be physically separate and independent so that no single event such as a tower falling or line breaking will be able to simultaneously affect all circuits in such a way that none of the four circuits can be returned to service in time to prevent fuel design limits or design conditions of the reactor coolant pressure boundary from being exceeded. Line crossovers and physical separation of these four transmission lines were not described in the FSAR.

The applicant, by letter dated August 29, 1983, documented that the design is in compliance with the above staff position and provided a description of the design. On the basis of this description, the staff concludes that the transmission lines are physically separate and independent in accordance with the above-stated staff position, meet the requirements of GDC 17, and are, therefore, acceptable pending incorporation of the description of the design in Section 8.2 of the FSAR.

8.2.2.2 Physical Separation of Offsite Circuits Between Switchyard and Class 1E System

As implied in Section 8.1.2 of the FSAR, the Millstone design provides two immediate access offsite circuits between the switchyard and the 4.16-kV Class 1E buses. It is the staff position that these two circuits be physically separate and independent so that no single event can simultaneously affect both circuits in such a way that neither can be returned to service in time to prevent fuel design limits or design conditions of the reactor coolant pressure boundary from being exceeded. The physical separation and independence of these two circuits had not been described or analyzed in the FSAR.

The applicant, by letter dated June 12, 1984, presented additional information in regard to these circuits. On the basis of the additional information, the staff concludes that the design meets the above-stated staff position and meets GDC 17, and is acceptable.

8.2.2.3 Verification Testing for Generator Circuit Breakers

As described in Section 8.3.1.1.1 of the FSAR, the Millstone design arrangement provides two immediate access offsite circuits. One of these circuits uses a generator circuit breaker to isolate the turbine generator from the main and normal station service transformers. Other facilities that use generator circuit breakers have been required to perform verification testing. The applicant, by letter dated August 29, 1983, provided the verification test program and results. On the basis of these test results, it appears that the capability of the generator breakers has been adequately demonstrated and is acceptable. However, subsequent to the staff's request for information, a revision (NUREG-0800) to the SRP (NUREG-75/087) was issued that provided more specific guidelines with respect to generator circuit breakers. The applicant was, therefore, further requested to review these specific guidelines with respect to the test results and provide a positive statement of compliance or justification for any deviations. The specific guidelines are given in Appendix A to SRP Section 8.2 and are dated July 1983. By letter dated June 12, 1984, the applicant stated that the design meets the requirements of Appendix A of SRP Section 8.2. On the basis of this statement of compliance, the staff concludes that the design meets the more specific guidelines and is, therefore, acceptable.

8.2.2.4 Grid Stability

It is the staff position that the Millstone grid stability analysis must show that loss of the largest single supply to the grid does not result in the complete loss of preferred power. The analysis should consider the loss, through a single event, of the largest capacity being supplied to the grid, removal of the largest load from the grid, or loss of the most critical transmission line. The combined capacity of Millstone Units 1, 2, and 3 is to be supplied to the grid through the common Millstone switchyard. The combined capacity of the three units appears to be the largest capacity being supplied to the grid and should be considered in the Millstone grid stability analysis.

The applicant, by letter dated August 29, 1983, indicated that the simultaneous loss of the entire output of the Millstone station (Units 1, 2 and 3) was modeled or analyzed for one set of operating conditions with the results indicating grid stability. However, the applicant further indicated that it is possible that a similar test under a more severe set of operating conditions may result in grid instability. In justification the applicant documented that in his engineering judgment the probability of losing all three units simultaneously is extremely small because of preventive measures included in the design of the offsite power system.

These preventive measures include an automatic generation rejection scheme and independent pole tripping with independent relays and trip coils for switchyard breakers. Thus, the applicant determined that it is reasonable to count on onsite power sources to supply the necessary station service power requirements in the unlikely event that all three Millstone units should be lost at once accompanied by total loss of offsite power.

The staff agrees with the applicant and concludes that the preventive measures included in the design of the offsite power system minimize the probability of losing electric power from the transmission network as a result of, or coincident with, loss of power generated by the nuclear power unit, meet the requirements of GDC 17, and are, therefore, acceptable.

8.2.2.5 Generation Rejection Scheme

There are four transmission circuits that connect the Millstone switchyard to the grid system. The four circuits are routed on two tower lines - two circuits per tower line. Section 8.1.3 of the FSAR indicates that a simultaneous failure of either of the two tower lines with only one circuit in service on the other tower line may result in instability of Millstone generation. The applicant, to prevent instability, has installed a protection scheme to automatically reduce generator output at Millstone Unit 3.

The applicant, by letter dated August 29, 1983, provided a description of the protection scheme. However, to conclude that the design meets GDC 17 and 18 for the proposed mode of operation (one of four offsite transmission lines out of service), the staff requires that additional description of surveillance, operability requirements, and analysis demonstrating compliance with the requirements of GDC 17 and 18 be documented in the FSAR. By letter dated June 12, 1984, the applicant provided the additional description of surveillance, operability requirements, and analysis demonstrating compliance with GDC 17 and 18. On the basis of this analysis, the staff concludes that the design meets GDC 17 and 18 and is acceptable pending incorporation of the description of surveillance, operability requirements, and analysis in Section 8.2 of the FSAR. Surveillance and operability requirements for the protection scheme will be included in the Technical Specifications.

8.2.2.6 Description and Analysis Demonstrating Compliance With GDC 17

A system description and analysis sufficient to demonstrate compliance with GDC 17 had not been presented in Section 8.2 of the FSAR in accordance with the guidelines of RG 1.70. By letter dated June 12, 1984, the applicant presented the required description and analysis. This item is, therefore, resolved.

8.2.3 Compliance With GDC 18

The applicant has met the requirements of GDC 18 with respect to the capability to test systems and associated components during normal plant operation and the capability to test the transfer of power from the nuclear power unit, the offsite preferred power system, and the onsite power system. The following item addresses the problem area revealed during the staff review and its resolution.

8.2.3.1 Description and Analysis Demonstrating Compliance With GDC 18

A system description and analysis sufficient to demonstrate compliance with GDC 18 had not been presented in Section 8.2 of the FSAR in accordance with the guidelines of RG 1.70. By letter dated June 12, 1984, the applicant presented the required description and analysis. This item is, therefore, resolved.

8.2.4 Evaluation Findings

The review of the offsite power system for the Millstone plant covered single-line diagrams, station layout drawings, schematic diagrams, and descriptive information. The basis for acceptance of the offsite power system in the staff review was conformance of the design criteria and bases to the Commission's regulations as set forth in the GDC of Appendix A to 10 CFR 50. The staff concludes that the plant design meets the requirements of GDC 5, 17, and 18 and conforms to the applicable guidelines of RGs and branch technical positions and is acceptable, except as noted in the preceding sections.

8.3 Onsite Power Systems

The safety function of the onsite power system (assuming the offsite power system is not functioning) is to provide sufficient capacity and capability to ensure that the structures, systems, and components important to safety perform as intended. The objective of the review is to determine that the onsite power system satisfies the requirements of GDC 2, 4, 5, 17, 18, and 50 and will perform its intended function during all plant operating and accident conditions.

The onsite power system consists of an ac power system and a dc power system. Compliance with GDC 2, 4, 5, 18, and 50 as they relate to both ac and dc systems is evaluated in Section 8.3.3 of this SER. Compliance with GDC 17 as it relates to ac systems is evaluated in Section 8.3.1 and as it relates to dc systems in Section 8.3.2 of this SER.

8.3.1 Onsite AC Power System's Compliance With GDC 17

The applicant has met (except as noted) the requirements of GDC 17 with respect to the onsite ac system's (1) capacity and capability to permit functioning of structures, systems, and components important to safety, (2) independence, redundancy, and testability to perform the safety function assuming a single failure, and (3) provisions to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit or the loss of power from the transmission network.

The following items address the problem areas revealed during the staff review and their resolution or status.

8.3.1.1 Separation Between Onsite and Offsite Circuits

Each of the 4.16-kV Class 1E buses at Millstone is supplied power from preferred offsite and standby onsite circuits. It is the staff position that these circuits should not have common failure modes. Physical separation and independence of these circuits was not described or analyzed in the FSAR.

The applicant, by Figure 8.3-7 of Amendment 3 to the FSAR, provided the required description of the physical separation and independence between onsite and offsite circuits. On the basis of this description, the staff concludes that onsite and offsite circuits are independent, meet GDC 17, and are acceptable.

8.3.1.2 Positive Statement of Compliance With Branch Technical Position PSB-1

Branch Technical Position (BTP) PSB-1 was not identified in Table 8.1-2 of the FSAR; thus, a positive statement as to compliance with staff guidelines had not been provided.

The applicant, by letter dated August 29, 1983, stated that BTP PSB-1 is currently under review and would be addressed in a future amendment to the FSAR. By letter dated June 12, 1984, the applicant revised Table 8.1-2 of the FSAR to indicate compliance with BTP PSB-1. This item is, therefore, resolved.

8.3.1.3 Description of Compliance With Position 1 of BTP PSB-1

Section 8.3.1.1.4 of the FSAR indicates that a degraded voltage scheme with 2-out-of-4 logic is provided on each of the 4.16-kV Class 1E buses. The applicant was requested to provide references to electric schematic drawings that describe the degraded voltage scheme and provide a description, with voltage and time setpoints, to indicate how the Millstone design complies with the guidelines of Position 1 of BTP PSB-1 (NUREG-0800, Appendix 8A).

The applicant, by letter dated August 29, 1983, provided the requested references to electric schematic drawings and indicated that compliance of the design with Position 1 of the BTP was currently under review and would be addressed in a future amendment. By letter dated June 12, 1984, the applicant provided the requested description. On the basis of this description, the staff concludes that the design meets the guidelines of Position 1 of BTP PSB-1 and is acceptable pending incorporation of the description in Section 8.3 of the FSAR. The referenced electrical schematic drawings will be reviewed as part of the staff's confirmatory site visit. If any problem areas are identified, they will be reported in a supplement to this SER.

8.3.1.4 Automatic Reset of the Load Sequencer on Low Voltage

As stated in Section 8.3.1.1.3 of the FSAR, the emergency generator load sequencer (EGLS) has the capability to automatically reset during a sustained low voltage condition on the essential bus. It is the staff concern that this capability may unnecessarily delay the connection of the required mitigating loads within the times allowed by the accident analysis.

The applicant by Amendment 3 revised the FSAR to indicate that automatic reset occurs only when there is a loss of offsite power subsequent to an accident signal. On the basis of this revision to the FSAR, the staff concludes that the load sequencer is used to (1) sequence loads on onsite power sources when there is an accident signal with concurrent or delayed loss of offsite power and (2) block load-required safety loads on offsite power sources when there is an accident signal; thus, the load sequencer is used to connect loads on both onsite and offsite power sources. The evaluation and acceptability of this item is included in Section 8.3.1.6 of this SER.

8.3.1.5 Adequacy of Station Electric Distribution System Voltage

It is the staff position that the voltage levels at the safety-related loads should be optimized for the maximum and minimum load conditions that are expected throughout the anticipated range of voltage variations of the offsite power sources. The applicant was requested to (1) perform a voltage analysis and verification by actual measurement in accordance with the guidelines of Positions 3 and 4 of BTP PSB-1 (NUREG-0800, Appendix 8A) and (2) provide the voltage at the terminals of each Class 1E load as determined by analysis for all modes of plant operation.

By letter dated June 12, 1984, the applicant provided additional clarification. With respect to Position 3, the applicant indicated that a voltage analysis is being performed to ensure that voltages for all modes of plant operation and accident conditions will remain above the steady-state and transient start design voltage rating of all Class 1E loads. The staff concludes that this analysis meets the guidelines of Position 3 and is acceptable pending verification of the analysis results.

With respect to Position 4, the applicant indicated that verification testing of the above analysis would be performed on one offsite power circuit and would not be performed on 120/208-V buses that are fed from regulated power supplies. On the basis of discussions with the applicant and the clarification provided, the staff concludes that the extent of testing meets the intent of Position 4, and that the analysis will be verified by test in accordance with the guidelines of Position 4 and is acceptable. The test results will be verified as part of the staff's normal site visit review program. If any problems are identified they will be reported in a supplement to this SER.

8.3.1.6 Reliability of the Load Sequencer

With respect to the use of a common load sequencer between onsite and offsite power sources at Millstone Unit 3, the applicant was requested to provide the results of a reliability analysis that demonstrates that overall reliability, availability, or capability of onsite and offsite power sources to supply power to safety loads on demand has not been significantly reduced by the use of a common load sequencers.

By letter dated June 12, 1984, the applicant provided additional information and the results of analysis concerning the Millstone Unit 3 load sequencer. On the basis of the additional information and the automatic and manual test capability, the staff considers this item resolved.

Surveillance requirements for the operability of the load sequencer logic will be included in the Technical Specifications.

8.3.1.7 Diesel Generator Protective Relaying

Section 8.3.1.1.3 of the FSAR indicates that diesel generator protective relaying is bypassed under accident conditions in accordance with BTP ICSB-17. The applicant by letter dated August 29, 1983, provided drawing reference numbers that describe the design of the bypass circuitry, the 2-out-of-3 logic circuitry, and the relaying that is not bypassed under accident conditions. These drawings will be reviewed with the applicant as part of the staff's confirmatory site visit. If any problem areas are identified they will be reported in a supplement to this SER.

8.3.1.8 Compliance With Position 4 of RG 1.9

Section 8.1.7 of the FSAR indicates that the diesel generator voltage (before connection of the first load block) may drop below the 75% minimum level permitted by Position 4 of RG 1.9 (Rev. 2).

The applicant, by letter dated August 29, 1983, indicated that the voltage dip to levels below 75% is considered inconsequential to the successful loading of the standby generator unit. Subsequently, by letter dated June 12, 1984, the applicant stated that the voltage dip is of little consequence because of its transient nature. It will not be of sufficient duration to cause the pickup of instantaneous overcurrent relays, and at the 480-V motor control distribution system level, it will not last through the motor controller's contactor pickup time. The staff agrees with the applicant's assessment that the voltage dip is of little consequence and, therefore, concludes that the design meets GDC 17 and is acceptable.

8.3.1.9 Diesel Generator Load Shedding Test

Section 1.8 of the FSAR indicated that the Millstone design did not comply with Position C.2(a)4 of RG 1.108. The applicant implied that the diesel generator load shedding test will be conducted using the 2,000-hour rating for rejection of the single largest load and the continuous rating for complete loss of load.

By letter dated August 29, 1983, the applicant indicated that loss of the largest load and the complete loss of load tests would be conducted using the 2,000-hour rating of the diesel generator. This testing meets the guidelines of RG 1.108 and Section 6 of IEEE Std. 387-1977 and is, therefore, acceptable.

However, by letter dated June 12, 1984, the applicant indicated that loss of the largest load and the complete loss of load tests would be conducted using the continuous rating, versus the 2,000-hour rating. The staff concludes that testing at the continuous rating of the diesel generator also meets the guidelines of RG 1.108 and Section 6 of IEEE Std. 387-1977 and is, therefore, acceptable.

8.3.1.10 Diesel Generator Testing at 2,000-Hour Rating

Section 1.8 of the FSAR indicated that the Millstone design did not comply with Position C.2(a)3 of RG 1.108. The applicant, by Amendment 3, revised the FSAR to state that the diesel generator will be tested at the 2,000-hour rating for 22 hours. The applicant also defined the 2,000-hour rating to be 5,335 kW and the maximum rating at which the diesel generator can be operated. On the basis of the applicant's response, the staff concluded that the 2,000-hour rating is being used as the continuous rating of the diesel generator and that the diesel generator is being tested accordingly. Thus, the 2-hour overload test also required by Position C.2(a)3 of RG 1.108 should be greater than the 2,000-hour rating of the diesel generator. Generally, the 2-hour rating is 10% greater than the continuous rating. The applicant, by letter dated June 12, 1984, indicated that the diesel generator would be tested at its continuous rating of 4,986 kW for 22 hours and at 10% above the continuous rating for 2 hours. The staff concludes that this testing meets the guidelines of RG 1.108 and is acceptable.

8.3.1.11 Diesel Generator Load Acceptance Test After Operation at No Load

Section 6.4.2 of IEEE Std. 387-1977 requires, in part, that the load acceptance test consider the potential effects on load acceptance after prolonged no-load or light-load operation of the diesel generator. The applicant was requested to provide the results of the load acceptance test or analysis that demonstrates the capability of the diesel generator to accept the design-accident load sequence after prolonged no-load operation over the full range of ambient air temperatures that may exist at the diesel engine air intake.

In response, the applicant, by letter dated August 29, 1983, provided the results of a manufacturer's analysis. The analysis indicated that the only limitation to prolonged operation more than 24 hours at no load or light load is the accumulation of combustion and lubrication products in the exhaust system. The results of this analysis do not state that the diesel generator will maintain its capability to meet load acceptance test requirements after 24 hours of operation at no load or light load. How this no-load capability is considered in the load acceptance tests will be pursued with the applicant, and the results of the staff review will be reported in a supplement to this SER.

8.3.1.12 Capability of the Diesel Generator To Respond to an Accident Signal When in the Test Mode

In accordance with Section 5.6.2.2(1) of IEEE Std. 387-1977, Section 5 of IEEE Std. 338-1977, and Position C.2a(8) of RG 1.108, it is the staff position that the diesel generator, when in the test mode and parallel with the offsite power system, be capable of responding to an accident signal.

The applicant, by letter dated August 29, 1983, described how the Millstone design meets the staff position. On the basis of the description, the staff concludes that the design is capable of responding to an accident signal, meets the position, and is acceptable.

8.3.1.13 Diesel Generator Bypass and Inoperable Status Indication

By Amendment 3 to the FSAR, the applicant expanded Section 8.3.1.1.3 of the FSAR to describe control room status indicators for equipment that, when made inoperable, can render the diesel generator incapable of responding to an automatic start signal. It is the staff position, in accordance with Position 2.2 of BTP PSB-2, that all status indicators that indicate that the diesel generator is incapable of responding to an automatic start signal be alarmed in the control room and at the diesel generator unit with wording that indicates that the diesel generator unit is incapable of responding to an automatic start signal. On the basis of discussions with the applicant at a meeting on May 14, 1984, the staff concludes that the subject alarm is provided and that the design meets the guidelines of the staff position and is acceptable.

8.3.1.14 Testing To Demonstrate Capability of the Diesel Generator To Accept the Design-Accident Load

Section 1.8 of the FSAR indicates that the Millstone design does not comply with Position C.2(a)2 of RG 1.108. The capability of the diesel generator to accept the design-accident load sequence is to be demonstrated under conditions as close to design load as possible. By letter dated August 29, 1983, the applicant, in clarification of this exception, indicated that several pumps will not be delivering their design flow because they will be on recirculation.

The staff concludes that the proposed testing meets the intent of Position C.2(a)2 of RG 1.108 and is, therefore, acceptable.

8.3.1.15 Physical Independence

Physical independence criteria for the redundant onsite ac power system are the same as those for the onsite dc system and are, thus, addressed in Section 8.3.3 of this report.

8.3.2 Onsite DC System's Compliance With GDC 17

The applicant has met (except as noted) the requirements of GDC 17 with respect to the onsite dc system's (1) capacity and capability to permit functioning of structures, systems, and components important to safety; (2) independence, redundancy, and testability to perform their safety function assuming a single failure; and (3) provisions to minimize the probability of losing electric power from any of the remaining supplies as a result of, or coincident with, the loss of power generated by the nuclear power unit or the loss of power from the transmission network.

The following items address the problem areas revealed during the staff review and their resolution or status.

8.3.2.1 Design and Qualification of DC System Loads for Voltage Variations

Loads connected to the dc bus may be subject to voltage variations from 90 to 143 V as a result of battery discharge and equalizing charge. It is the staff position that dc loads be designed and qualified to operate when subject to these voltage variations.

In response to this position, the applicant documented, by letter dated June 12, 1984, that dc components are specified to operate between 90 and 140 V. On the basis of discussions with the applicant at a meeting on May 14, 1984, the staff concludes that the term "specified" means that Class 1E dc loads are designed and qualified to operate between 90 and 140 V. The design, therefore, meets the staff position and is acceptable.

8.3.2.2 DC System Monitoring and Annunciation

The specific requirements for dc power system monitoring derive from the generic requirements in Sections 5.3.2(4), 5.3.3(5), and 5.3.4(5) of IEEE Std. 308-1974 and in RG 1.47, "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems." In summary, these generic requirements state that the dc system (batteries, distribution systems, and chargers) shall be monitored to the extent that it is shown to be ready to perform its intended function.

It is the staff position that the following indications and alarms of the status of the Class 1E dc power system shall be provided in the control room:

(1) battery float charge (ammeter)
(2) battery circuit output current (ammeter)
(3) battery charger output current (ammeter)

(4) dc bus voltage (voltmeter)
(5) battery discharge alarm
(6) dc bus overvoltage alarm
(7) dc system ground alarm
(8) battery disconnect open alarm
(9) battery charger disconnect open alarm
(10) battery charger failure alarm (one alarm for a number of abnormal conditions that are usually indicated locally)

The staff has concluded that the above-cited monitoring, augmented by the periodic test and surveillance requirements that are included in the Technical Specifications, provides reasonable assurance that the Class 1E dc power system is ready to perform its intended safety function.

By letter dated August 29, 1983, the applicant indicated that the battery float charge (ammeter), battery charger output current (ammeter), battery discharge alarm, and battery disconnect open alarm have not been provided in the control room.

In regard to the battery float charge, the applicant has indicated that battery float charge indication can be obtained locally by use of a normally disconnected ammeter. This meets the intent of the staff position and is acceptable. Periodic surveillance of the battery float charge will be included in the plant Technical Specifications.

In regard to the battery charger output current, the applicant, by letter dated June 12, 1984, indicated that battery charger output current is indicated locally at the battery charger. This meets the intent of the staff position and is acceptable.

In regard to the battery discharge alarm, the applicant indicated that the ammeter and undervoltage alarm is provided in the control room. On the basis of discussions with the applicant at a meeting on May 14, 1984, the staff concludes that the undervoltage trip setpoint will be set so that an immediate alarm will be sounded in the control room when the battery versus the battery charger is supplying the load or the battery is being discharged. This design meets the objective of the staff position and is acceptable.

In regard to the battery disconnect open alarm, the applicant revised his response to indicate that a battery breaker position annunciation alarm has been provided. This alarm meets the staff position and is acceptable.

8.3.2.3 Restoration of AC Power Within 2 Hours

Section 8.3.2.1 of the FSAR indicates that power will be available to dc system loads for at least 2 hours in the event of loss of all ac power. After 2 hours it has been assumed that ac power is either restored or that the emergency generators are available to energize the battery chargers. On the basis of the staff's review of recent applications, this period for restoration of ac power appears to be too short. The applicant was requested to provide the basis and operational experience data for the assumption that ac power can be restored within 2 hours.

By letters dated August 29, 1983, and June 12, 1984, the applicant (1) provided the results of an analysis that indicated that the frequency of station blackout from which offsite power is not restored within 2 hours is 6.6 x 10-6 per year and (2) indicated that the procedures and training program in accordance with Generic Letter 81-04 will be completed before the fuel loading date for Millstone Unit 3. The applicant's commitment to Generic Letter 81-04 meets the review guidelines for this item. This item is, therefore, considered complete.

8.3.2.4 Physical Independence

Physical independence criteria for the redundant onsite dc power system are the same as those for the onsite ac system and are, thus, addressed in Section 8.3.3.3 of this report.

8.3.3 Common Electrical Features and Requirements

This section presents common electrical features and requirements of the onsite ac and dc power system that deal with distinct aspects of the onsite ac and dc power systems. The common electrical features and requirements addressed in this section are discussed in the following sections.

8.3.3.1 Compliance With GDC 2 and 4

The applicant has met (except as noted) the requirements of GDC 2 and 4 with respect to structures, system, and components of the onsite ac and dc power system being capable of withstanding the effects of natural phenomena such as earthquakes, tornados, hurricanes, floods, missiles, and environmental conditions associated with normal operation and postulated accidents. The onsite power system and components (1) are located in seismic Category I structures, which provide protection from the effects of tornados, tornado missiles, turbine missiles, and external floods; (2) have been given a quality assurance designation of Class 1E; (3) have been designated to be seismically and environmentally qualified; and (4) are to be designed to accommodate, or are to be protected from, the effects of missiles and environmental conditions associated with normal operation and postulated accidents.

The following items address the problem areas revealed during the staff review and their resolution or status.

8.3.3.1.1 Submerged Electrical Equipment as a Result of a Loss-of-Coolant Accident

It is the staff's concern that following a loss-of-coolant accident (LOCA), fluid (from the reactor coolant system and from operation of the emergency core cooling systems) may collect in the primary containment and reach a level that may cause certain electrical equipment located inside the containment to become submerged and thereby rendered inoperable. Both safety- and nonsafety-related electrical equipment is of concern because their failure may cause electrical faults that could compromise the operability of redundant emergency power sources or the integrity of containment electrical penetrations. In addition, the safety-related electrical equipment that may be submerged is also of concern if this equipment is required to mitigate the consequences of the accident for both the short-term and long-term emergency core cooling system functions and for containment isolation.

The staff's position, in regard to submerged equipment, is that all electrical equipment must be located above the maximum possible flood level or be qualified for submerged operation, or the lack of qualification must be justified.

The applicant was requested to identify all electrical equipment, both safety and nonsafety, that may become submerged as a result of a LOCA. For all such equipment that is not designed and qualified for service in such an environment, the applicant was requested to provide analyses to determine the following:

(1) the safety significance of the failure of this electrical equipment (e.g., spurious actuation or loss of actuation function) as a result of flooding

(2) the effects on Class 1E power sources serving this equipment as a result of such submergence

(3) any proposed design changes resulting from this analysis

The applicant, by letter dated June 12, 1984, indicated that there is both safety- and nonsafety-related equipment located inside containment that may become submerged as a result of a LOCA that is not designed or qualified for submergence.

In regard to the safety significance of the failure of this equipment (e.g., spurious actuation or loss of actuation function) as a result of its submergence, the applicant provided the results of an analysis that indicate that this equipment performs no post-LOCA safety function and that its failure position will not affect station shutdown capability. On the basis of these results, the staff concludes that the loss of actuation function and spurious actuation resulting from a LOCA-induced submergence are not concerns and are, therefore, acceptable.

In regard to the effects on Class 1E power sources serving this equipment, the applicant has separated the equipment into three groups and addressed the effects each group has on their power supplies as follows.

In regard to the equipment in group one, the applicant identified 12 safety-related motor-operated valves that will be deenergized during normal plant operation. As part of the Millstone Unit 3 Technical Specifications, the staff will require periodic verification that these valves are in fact deenergized during normal plant operation. With the imposition of this requirement, the staff concludes that there is reasonable assurance that these valves will be deenergized given a LOCA and will not adversely affect Class 1E power supplies. The design is, therefore, acceptable.

In regard to the equipment in groups two and three, the applicant has identified two safety-related valves and an unspecified number of nonsafety-related electrical equipment. The applicant has stated that each of the circuits associated with this equipment is provided with two series connected interrupting devices that meet the guidelines of RG 1.75 for an isolation device.

The staff does not agree that two series connected interrupting devices meet RG 1.75 guidelines for an isolation device; however, with respect to the degree of protection that these two series connected Class 1E breakers provide their associated power supplies, the staff concludes that there is reasonable assurance that the power supplies will not fail as a result of submergence of equipment. The design is, therefore, acceptable.

8.3.3.1.2 Missile Protection for Cables

Section 8.3.1.4.2 of the FSAR stated, in part, that Class 1E cables of only one train will be installed in potential missile-producing areas or adequate missile protection will be provided when Class 1E cables of redundant trains are installed in missile-producing areas. On the basis of this statement, it appeared that Class 1E equipment and cables of each redundant division were not to be protected from the effects of accident-generated missiles.

By Amendment 3, the applicant revised the FSAR to state that, in general, Class 1E equipment is not installed in potential missile-producing areas. Where this is not practical, suitable missile protection is provided. On the basis of this statement, the staff concludes that adequate protection will be provided to Class 1E equipment of both redundant divisions. This item is, therefore, considered resolved.

8.3.3.1.3 Design Criteria for Independence of Redundant Systems

The applicant has defined design criteria for the independence and availability of Class 1E systems in Section 8.3.1.4.1 of the FSAR. The definition includes the statement that "separation of equipment is maintained to prevent loss of redundant features for single events and accidents." Similarly, Section 8.3.1.1.2 of the FSAR states that redundant Class 1E buses are physically and electrically separated so that any credible event that might affect one bus will not jeopardize proper operation of the other bus.

The above statements imply that, with sufficient separation, only one of the redundant Class 1E divisions need be protected from the effects of any design-basis event or accident. Such a design does not meet the protection requirements of GDC 2 and 4, the single-failure requirement of GDC 17, or the guidelines of IEEE Std. 308-1974.

By Amendment 3 to the FSAR and by letter dated June 12, 1984, the applicant stated that each redundant safety-related system is protected. On the basis of this statement, the staff concludes that Class 1E equipment will meet the protection requirement of GDC 2 and 4 and the single-failure requirement of GDC 17 and is, therefore, acceptable.

8.3.3.2 Compliance With GDC 5

The applicant has met the requirements of GDC 5, "Sharing of Structures, Systems, and Components," with respect to the structures, systems, and components of the ac and dc onsite power systems.

8.3.3.3 Physical Independence - Compliance With General Design GDC 17

8.3.3.3.1 Compliance of Associated Circuits With Requirements of Class 1E Circuits

In Section 1.8 of the FSAR, the following clarification to Position C.4 of RG 1.75 was identified: "Associated circuits are identified by the same color code as the Class 1E circuits with which they are associated. This color code exists up to and including an isolation device."

Position C.4 of RG 1.75 requires that associated circuits (up to and including an isolation device) be subjected to all the requirements placed on Class 1E circuits unless it can be demonstrated that the absence of such requirements cannot significantly reduce the availability of Class 1E circuits. The applicant's clarification implied that associated circuits meet only the color code requirement rather all the requirements placed on Class 1E circuits.

The applicant, by Amendment 3 to the FSAR, stated that associated circuits meet all the requirements placed on Class 1E circuits up to and including an isolation device. On the basis of this statement, the staff concludes that the design for associated circuits is in accordance with the guidelines of IEEE Std. 384 as augmented by RG 1.75 and is, therefore, acceptable.

8.3.3.3.2 Frequency of Cable Identification Markings

In Section 1.8 of the FSAR, the applicant identified an exception to Position C.10 of RG 1.75. Class 1E cables are to be marked at intervals not exceeding 15 ft rather than 5 ft as required by the RG. In justification of the exception, the applicant documented that the 5-ft requirement is a typographical error in RG 1.75. The staff disagrees. The 5-ft requirement is not considered a typographical error. A 5-ft maximum marking distance is considered necessary to facilitate easy visual verification that the cable installation is in conformance with separation criteria.

The applicant, by letter dated August 29, 1983, restated that the 5-ft marking requirement is a typographical error and added the statement that a 15-ft maximum marking distance is sufficient to facilitate easy visual verification that the cable installation is in conformance with separation criteria. The staff will confirm this aspect of the design as part of its site visit. This item is, therefore, considered complete. If problem areas are identified during the site visit, they will be reported in a supplement to this report.

8.3.3.3.3 Routing of Power Circuits in the Cable Spreading Area

In FSAR Section 1.8, the applicant identified an exception to Position C.12 of RG 1.75. Position C.12 indicates that (1) power supply feeders to the instrument and control room distribution panel installed in enclosed raceways should not be considered acceptable, (2) traversing power circuits separated from other circuits in the cable spreading area by a minimum distance of 3 ft and barriers should not be considered acceptable, and (3) traversing power circuits routed in imbedded conduit, which in effect removes them from the cable spreading area, should be considered acceptable.

Power circuits that traverse the cable spreading area at Millstone are installed in enclosed raceways (rigid steel conduit). In accordance with Position C.12 of RG 1.75, this routing should not be considered acceptable.

The applicant, by letter dated August 29, 1984, documented that potential electrical fires caused by fault current in the power cables are not considered a hazard. Fires resulting from fault current, if possible, would be contained in the rigid steel conduit. The staff agrees with the applicant and concludes that rigid steel conduit provides an acceptable level of assurance that other circuits located in the cable spreading area will not be affected by failure of the traversing power circuits.

In regard to failure of these traversing power circuits along with other circuits located in the cable spreading room, control room, or instrument rack room, as a result of the design-basis-event fire, the applicant, by letter dated June 12, 1984, indicated that total loss of these circuits will not compromise the capability to achieve cold shutdown. The effect that electrical failure of these traversing circuits may have on power supplies needed to achieve safe shutdown was discussed with the applicant at a meeting on May 14, 1984. On the basis of this discussion, the staff concludes that loss of power supplies resulting from failure of circuits has been considered in the procedures used for alternate shutdown. The staff's review of these procedures and the alternate shutdown capability is reported in Section 9.5.1 of this SER.

The existence of traversing power circuits in the control room and instrument rack room had not been addressed. The applicant, by letter dated June 12, 1984, stated that power circuits do not traverse the control or instrument rack rooms. On the basis of this statement, the staff considers this item resolved.

In regard to external energetic events, the applicant, by letter dated June 12, 1984, indicated that the control room and instrument rack room as well as the cable spreading area are protected areas and are not subject to external energetic events such as floods, high-energy-pipe breaks, and missiles. Pending documentation of these design criteria in the FSAR, the staff concludes that the control and instrument rack rooms do not contain high-energy equipment such as switchgear, transformers, rotating equipment, or potential sources of missiles or pipe whip; meet the review guidelines of Section 5.1.3 of IEEE Std. 384-1974 for the cable spreading area, and are acceptable.

In regard to the routing of power circuits to distribution panels located in the control room and instrument rack room, the applicant, by letter dated June 12, 1984, indicated that these circuits are routed in rigid conduit in the control and instrument rack rooms with flexible conduit used at the entrances to panels. This design meets the intent of Position C.12 of RG 1.75 and is acceptable.

8.3.3.3.4 Adequacy of Cable Separation Inside Balance-of-Plant Cabinets

In Section 1.8 of the FSAR, the applicant has taken exception to Position C.16 of RG 1.75 and Section 5.6.2 of IEEE Std. 384-1974. Minimum separation between redundant Class 1E wire or between Class 1E and non-Class 1E wire bundles is identified to be 1 in. rather than 6 in. inside control switchboards and instrument cabinets.

The applicant, by letter dated August 29, 1983, restated his justification for 1-in. minimum separation. The subject control switchboards and instrument cabinets are located in a protected area and are not subject to external energetic events such as flood, high-energy-pipe rupture, and missiles. Electrically generated fires caused by fault current are not considered to be a hazard because of the use of fire-retardant material and low-energy cables. The 1-in. separation is justified because it will prevent interaction between wire bundles resulting from electrical potentials or heated wire caused by electrical faults. On the basis of the above justification, the staff concludes that the subject 1-in. separation provides sufficient independence between redundant circuits and an acceptable level of protection to Class 1E circuits in accordance with the independence and single-failure requirements of GDC 17 and is, therefore, acceptable.

8.3.3.3.5 Interconnection Between Redundant Class 1E Divisions

A third or spare charging pump may be connected to either Class 1E bus 34C or 34D. By Amendment 3 to the FSAR, the applicant described the interlocks that preclude two charging pumps from being powered from the same Class 1E bus and redundant buses from being tied together.

On the basis of the description, the staff concludes that the electrical and key interlock design will preclude interconnection of redundant systems, meets the independence requirements of GDC 17, and is acceptable.

8.3.3.3.6 Transfer of Loads Between Redundant Divisions

Section 9.5.4.3 of the FSAR states, in part, that one fuel oil transfer pump on each fuel oil storage tank is arranged to allow transfer from the A electrical bus to the B electrical bus, or vice versa, by means of a 480-V, seismically qualified Class 1E transfer switch manually operated under administrative control.

It is the staff's position that the design of each interconnection should prevent a single failure or inadvertent closure of one interconnecting device from compromising division independence. An acceptable design includes a minimum of two series-connected disconnect devices that are physically separated, interlocked, administratively kept normally open, and annunciated in the control room upon closure.

The applicant, by letter dated August 29, 1983, identified all interconnections and described how each met the above staff position. On the basis of these descriptions, the staff concludes that the design for each interconnection meets the above-stated position and is acceptable with the following exception. Interconnection of the second fuel oil transfer pump is not annunciated in the control room upon closure. On the basis of a discussion with the applicant at a meeting on May 14, 1983, the staff concludes that the key interlock design ensures that two breakers between redundant buses will always be open; thus, annunciation for inadvertent closure of one interconnecting device is not required. The design meets the intent of the staff position and is acceptable.

8.3.3.3.7 Physical and Electrical Separation of Heat Tracing Circuits

Section 8.3.1.1.4(8) of the FSAR indicates that all safety-related pipe lines or valves subject to freezing or boron precipitation are electrically heat traced and insulated. Two heat tracing circuits are provided for each line subject to freezing. One circuit is connected to safety division A; the other circuit is connected to division B. Because heat tracing circuits are routed in close proximity on the same pipe line, it is the staff's concern that failure of one circuit may cause failure of both circuits and that these failures may reflect back to cause failure of both their respective power supplies.

The applicant, by letter dated June 12, 1984, indicated that these circuits are isolated from their Class 1E power supplies by two series-connected Class 1E circuits breakers, and one current-limiting transformer and are routed separately in dedicated conduits to their heat trace loads and only one of the two circuits is normally energized. The staff concludes that this design meets the guidelines of IEEE Std. 384-1984 as augmented by RG 1.75, meets the protection and independence requirements of GDC 17 and is, therefore, acceptable.

8.3.3.3.8 Adequacy of Protection Provided Class 1E Circuits From the Effects of Non-Class 1E Circuits

In Section 1.8 of the FSAR, the applicant with respect to compliance with Position C.7 of RG 1.75 has identified the following exceptions to separation between Class 1E and Non-Class 1E circuits located in the cable spreading, control, or instrument rack rooms.

(1) Where plant arrangement precludes the minimum vertical separation of 3 ft, the following provides the necessary separation: 10-in. minimum separation (measured from the bottom of the top tray to the top of the bottom tray) with a solid tray bottom on the upper tray, a solid tray cover on the bottom tray, or a barrier (meeting the guidelines of RG 1.75) interposed between trays.

(2) Where plant arrangement precludes 10 in. of vertical separation or 10 in. of horizontal separation, the following provides the necessary separation: 1-in. minimum separation, with the Class 1E circuits installed in an enclosed raceway, the non-Class 1E circuits installed in an enclosed raceway, or a barrier interposed between raceways.

Given the low-energy nature of the cables, it is the staff's judgment that the above separation provides adequate protection of Class 1E circuits from non-Class 1E circuits, meets the objectives of RG 1.75 and the requirements of GDC 2, 4, and 17, and is, therefore, acceptable.

8.3.3.3.9 Use of a Battery Charger as an Isolation Device

Section 8.3.2.1.1 of the FSAR states that battery charger 5 is powered from a Class 1E emergency bus, furnishes dc power to nonsafety loads, and meets all the requirements of an isolation device. The staff does not agree that the charger meets all requirements of an isolation device; therefore, the applicant was requested to provide test results and/or analysis that demonstrates that any failure or combination of failure or malfunction in the nonsafety circuits including hot short will not cause unacceptable influence on Class 1E circuits.

In response, the applicant, by letter dated August 29, 1983, indicated that (1) the output cables from the charger to the distribution switchboard are run in a dedicated conduit to preclude hot short from an external voltage source and (2) short-circuit tests will be conducted.

Subsequently, by letter dated June 12, 1984, the applicant indicated that unacceptable influence on the Class 1E system is precluded by the following design features:

(1) The battery charger is Class 1E and is protected by two physically separated Class 1E input circuit breakers.

(2) The dc output circuit to the distribution switchboard is routed in dedicated conduit.

(3) The circuits from the charger to the loads are protected by a dc charger output breaker and main circuit and feeder circuit breakers located in the distribution switchboard.

(4) The circuits from the switchboard to the loads are physically and electrically separated from Class 1E or associated circuits of the other safety division.

(5) Testing of the charger has demonstrated its current-limiting capability.

On the basis of the above design features, the staff concludes that the Class 1E system is adequately protected from failure in the nonsafety system and that independence between redundant safety systems cannot be compromised through the nonsafety system. The design, therefore, meets the guidelines of Section 4.9 of IEEE Std. 308-1974 and Section 4.5(3) of IEEE Std. 384-1974, the independence and single-failure requirement of GDC 17, and is acceptable with one exception.

The applicant's response to this item, provided by letter dated June 12, 1984, was contradicted by the applicant's response to the item presented in Section 8.3.3.3.10 of this report. The applicant stated that all nonsafety circuits beyond an isolation device (except the nonsafety circuits connected to the Class 1E division through the battery charger) are either routed in rigid conduit or maintain the same color as the Class 1E division to which they are connected.

On the basis of discussions with the applicant, it appears that non-Class 1E circuits connected to the Class 1E system through isolation transformers shown on Figure 8.3-3 of the FSAR are also not routed in either dedicated conduit or maintained with the same color as the Class 1E division to which they are connected. This contradiction has been discussed with the applicant and the resolution will be reported in Section 8.3.3.3.10 of this report.

8.3.3.3.10 Transformer Used as an Isolation Device

By Section 8.3.1.1.2 (Item 3) and Figure 8.3-3 of the FSAR, the applicant indicated that non-Class 1E nuclear steam supply system loads are connected to the Class 1E 120-V vital ac buses through transformers that are qualified as isolation devices. The staff disagrees that the transformers are qualified isolation devices.

By letters dated August 29, 1983, and June 12, 1984, the applicant provided results of tests and design provisions to ensure that non-Class 1E circuits are sufficiently isolated and will not cause unacceptable influence on any Class 1E circuit.

The applicant will have to provide clarification of these design provisions, and the results of the staff review will be reported in a supplement to this SER.

8.3.3.3.11 Interrupting Device Actuated Only by Fault Current Used as an Isolation Device

Section 8.3.2.1.1 of the FSAR stated that nonsafety 480-V stub bus 32-3T (which supplies power to a number of nonsafety dc loads located in a nonseismic building) is powered from a Class 1E bus and is automatically shed on loss of offsite power. It is the staff's position that this stub bus should also be automatically shed on an accident signal.

By Amendment 3 to the FSAR, the applicant indicated that the subject stub bus is designed to be automatically disconnected on accident signal as well as loss of power signal. This design meets the above staff position and is, therefore, acceptable.

8.3.3.3.12 Use of Interrupting Devices Actuated by Fault Current as an Isolation Device

Section 1.8 of the FSAR indicates an exception to Position C.1 of RG 1.75 by stating that interrupting devices actuated by fault current are isolation devices when justified by test or analysis. By letter dated June 12, 1984, the applicant indicated that the non-Class 1E circuits isolated by a single breaker are located only in NSSS-supplied circuits. Test and analysis to justify isolation of these circuits is provided in WCAP-8892. The staff evaluation of this item is reported in Section 7.2.2.1 of this report.

8.3.3.3.13 Use of Ventilated Tray Covers

In Section 1.8 of the FSAR, with respect to RG 1.75, the applicant indicated that ventilated tray covers are considered equivalent to solid tray covers. The staff agrees. Either ventilated or solid nonventilated covers meet the intent of RG 1.75 and are acceptable.

8.3.3.3.14 Separation of Cables at Entry, Exit, and Crossing of Raceways

In Section 1.8 of Amendment 3 of the FSAR, the applicant indicated, with respect to general clarification of the guidelines of RG 1.75, that separation at cable entry/exit from cable trays is equivalent to perpendicular cable tray crossing.

By letter dated June 12, 1984, the applicant provided drawings to further clarify separation at entry/exit from cable trays. The drawings depicted a minimum cable separation of 1 in. between enclosed raceways of one division and cables that enter/exit a raceway of a different division.

At a May 14, 1984 meeting, the applicant indicated that this 1-in separation was justified based on separation allowed by IEEE Std. 384-1974. Figure 5 of Section 5.1 of IEEE Std. 384-1974 shows an example of an acceptable arrangement for redundant cable tray crossings where acceptable vertical separation distance cannot be maintained. The applicant interpreted this figure to show an open bottom or ladder-type tray being located 1 in. above a ladder-type cable tray with a solid cover running perpendicular. The basis presented by the applicant for this interpretation was Figure 2 of IEEE Std. 384-1974. Figure 2 clearly states that the trays are solid bottom; thus, trays shown in Figure 5 (without the clearly stated designation that the tray is solid bottom) are open bottom or ladder-type trays. The staff disagreed with this interpretation. Section 5.1 of IEEE Std. 384-1974 clearly states that Figures 2, 3, 4, and 5 illustrate examples of acceptable arrangements of enclosed raceways.

It is, thus, the staff position (in accordance with the guidelines of IEEE Std. 384-1974 as augmented by RG 1.75) that two enclosed raceways separated by a minimum of 1 in. provides an acceptable arrangement for physical separation of redundant raceways where minimum separation distance between nonenclosed or ladder-type trays cannot be maintained.

Subsequently, by letter dated July 2, 1984, the applicant provided revised drawings (FSAR Figure 8.3-8, Revision 1, June 15, 1984) to show a minimum of 6 in. of separation between a bare Class 1E cable and a non-Class 1E covered cable tray or enclosed raceway located below the Class 1E cable. The staff concludes that this specific configuration provides adequate protection for Class 1E circuits, meets the requirements of GDC 17, and is acceptable.

8.3.3.3.15 Coordination of Breakers

In Section 1.8 of Amendment 3 to the FSAR, the applicant, with respect to compliance with Position C.1 of RG 1.75, indicated that coordination is not required between two series-connected breakers used as isolation devices. The staff agrees that coordination is not required between the two series-connected breakers; however, it is the staff's position that coordination is required between each of the breakers and their main supply breaker.

The applicant, by letter dated June 12, 1984, indicated that this coordination is provided and that the interrupting devices will be tested and calibrated periodically to ensure that proper coordination is maintained. The staff concludes that this design for breaker coordination meets the staff's position for coordination and is acceptable. Periodic testing and calibration will be included in the Millstone Technical Specifications.

8.3.3.3.16 Design Criteria of Associated Circuit From the Isolation Device to Load

In Section 1.8 of Amendment 3 to the FSAR, the applicant, with respect to compliance with Position C.1 of RG 1.75, stated that non-Class 1E equipment connected to Class 1E power supplies is (1) identified with the same color code from the source to the load as the Class 1E power source to which it is connected, (2) connected to the power source through two separate series Class 1E breakers or fuses, and (3) routed in rigid steel conduit except for selected loads.

By Amendment 8 to the FSAR, the applicant provided further clarification. On the basis of this clarification, the staff concludes that (1) non-Class 1E circuits from the Class 1E power supplies to the loads, irrespective of isolation devices, are either routed in rigid conduit or are color coded the same as their associated power supplies and are, thus, routed with Class 1E circuits and (2) the design meets the guidelines of RG 1.75, the requirements of GDC 17, and is, therefore, acceptable.

8.3.3.3.17 Use of Silicon Dioxide as an Enclosure for Cables

The applicant, by letter dated June 12, 1984, indicated that short lengths of cable (less than 10 ft) enclosed in a protective wrap of woven silicon dioxide are considered to be protected to the same degree as the same cable in an enclosed raceway. Equivalency of a silicon dioxide versus metal enclosure will be pursued with the applicant. The results of the staff review will be reported in a supplement to this SER.

On the basis of discussions with the applicant, manufacturer's description and testing, and testing performed by other utilities for this specific application of the wrap, the staff agrees that a wrap of woven silicon dioxide is equivalent to a metal enclosure with respect to protection from electrical failures. The staff, therefore, concludes that the subject wrap is acceptable for short lengths of cable that do not require structural support or restraint to maintain required physical separation from adjacent raceways or similarly wrapped cables.

8.3.3.3.18 Use of Non-Class 1E Metal-Clad Cable

In Section 1.8 of Amendment 8 to the FSAR, the following general clarifications to RG 1.75 were identified.

Metal-clad cable, type MC, used in low-energy, 120-Vac and 125-Vdc nominal circuits and in low-density applications is considered adequately protected. As such, the minimum separation between these cables and other cables, or raceway (where required), is 1 in. These cables are further described below:

(1) Type MC cable is a factory assembly of conductors, each individually insulated and enclosed in a metallic sheath of interlocking tape or a smooth or corrugated tube.

(2) The largest conductor size number is 10 AWG.

(3) There are no more than six conductors.

(4) There are no more than three number 10 AWG conductors, and the remaining conductors are of a smaller size.

(5) Aluminum sheath cable (a type MC cable in which the aluminum is continuously welded) and/or interlocked armor cable may have an overall jacket of neoprene or hypalon.

The staff concludes that the metal-clad cable separated by 1 in. from Class 1E raceways provides adequate protection for Class 1E circuits and is acceptable.

8.3.3.3.19 Use of SO or SJO Cords for Lighting Fixtures

By letter dated June 12, 1984, the applicant identified the following general clarification to RG 1.75

Type SO or SJO cords for lighting drops to fixtures are size 12 AWG or smaller and supply 120-Vac or 125-Vdc low energy in low-density applications. Adequate protection provided by 1 in. or greater distance to Class 1E raceways.

On the basis of the low energy and low-density application for these lighting cords, the staff concludes that 1-in. separation between Class 1E raceways and the lighting cord provides adequate protection to Class 1E circuits and is acceptable.

8.3.3.4 Compliance With the Guidelines of NUREG-0737

Two items related to GDC 17 are identified in NUREG-0737. These items are II.E.3.1, "Emergency Power Supply for Pressurizer Heaters," and II.G.1, "Emergency Power for Pressurizer Equipment." The background, the NUREG position, and clarification of the positions are included in the NUREG report.

The applicant's descriptions of compliance for these items was provided in a letter dated August 29, 1983.

II.G.1 Emergency Power for Pressurizer Equipment

A description of compliance with each of the four clarifications associated with this item was not included in the August 29, 1983, letter. By letter dated June 12, 1984, the applicant provided the description of compliance. On the basis of the description, the staff concludes that the Millstone design meets Item II.G.1 and its clarifications (except Clarification 1) and is acceptable.

Clarification 1 requires that the design retain to the extent practical, the capability to also open the power-operated relief valves (PORVs) and their associated block valves. The applicant has stated that the PORVs and their associated block valves are powered from opposite power trains. Powering the valves from opposite power trains meets the objective of this TMI Action Plan Item but does not meet the recommendations of BTP RSB 5-2 for overpressurization protection while operating at low temperature. To meet the objective of this TMI Action Plan Item (as indicated by Clarification 2), the power for the PORVs and block valves can be supplied from different emergency power sources (e.g., PORV on dc power and the associated block valve on ac power), both power sources emanating from the same division. This single-division orientation of power sources also is in conformance with BTP RSB 5-2 recommendations for reactor overpressurization protection at low temperature. This item will be pursued with the applicant, and the results of the staff review will be reported in a supplement to this SER.

II.E.3.1 Emergency Power Supply for Pressurizer Heaters

A description of compliance with each of the seven clarifications associated with this TMI item was not included in the August 29, 1983, letter. By letter dated June 12, 1984, the applicant provided the description of compliance. On the basis of the description, the staff concludes that the Millstone design meets Item II.E.3.1 and its clarifications (except for Clarification 7) and is acceptable.

Clarification 7 requires that pressurizer heaters be automatically shed from the emergency buses on the occurrence of a safety injection actuation signal. In justification of the design, which does not trip the heaters on safety injection, the applicant stated that the connection of the heaters to the safety-related trains meets RG 1.75. The evaluation and acceptability of this item is included in Sections 8.3.3.3.12 and 8.3.3.3.16 of this SER.

8.3.3.5 Compliance With GDC 18

The applicant has met the requirements of GDC 18, "Inspection and Testing of Electric Power Systems," with respect to the onsite ac and dc power system. The onsite power system is designed to be testable during operation of the nuclear power generating station as well as during those intervals when the station is shut down.

8.3.3.6 Compliance With GDC 50

The applicant has met (except as noted) the requirements of GDC 50, "Containment Design Bases," with respect to electrical penetrations containing circuits of the safety and nonsafety onsite power systems. GDC 50 requires, in part, that the reactor containment structures, including penetrations, be designed so that the containment structure and its internal compartments can accommodate the calculated pressure and temperature conditions resulting from any loss-of-coolant accident without exceeding the design leakage rate and with sufficient margin.

The following items address the problem areas revealed during the staff review and their resolution or status.

8.3.3.6.1 Primary and Backup Fault Protection for Containment Electrical Penetrations

Section 8.3.1.1.4 (Items 2 m and 4) of the FSAR indicates that primary and backup containment electrical penetration protection is provided only where the available fault current exceeds the current-carrying capabilities of penetration conductors for loads connected to safety-related buses that are not qualified to the containment accident environment. This design for containment electrical penetration protection does not meet the guidelines of Position 1 of RG 1.63. Position 1 requires (1) primary and backup protection where maximum available fault current exceeds the current-carrying capability of the penetration versus capability of the conductors and (2) that all conductors that pass through containment electric penetrations must have primary and backup protection as compared with only those that are connected to safety-related buses and loads that are not qualified to the containment accident environment.

In justification of this area of noncompliance with Position 1 of RG 1.63, the applicant by Amendment 3 to the FSAR stated:

> For Class 1E containment circuits which are fully qualified for the containment environment (both accident and normal), the single failure is assumed to be a failure of the circuit to survive the environment for which it is qualified. For this condition, a single protective device properly selected to protect the penetration, fully satisfies the single failure criterion of IEEE [Std.] 279-1971, and the intent of IEEE Std. 317-1976 and Regulatory Guide 1.63, Revision 2.

The staff disagrees. The staff considers the event to be circuit failure and the single failure to be loss of the primary fault current protective device. By letter dated June 12, 1984, the applicant committed to the installation of backup protective devices for all circuits that pass through containment electric penetrations before full-power operation following the first refueling outage. On the basis of this commitment and information presented, the staff concludes that the design after the first refueling outage will meet the guidelines of RG 1.63 and the requirements of GDC 50 and is, therefore, acceptable. With respect to the acceptability of the design during the first fuel cycle, the staff concludes, on the basis of information presented, that all circuits that pass through containment electric penetrations (except for a few normally deenergized motor-operated valve circuits that are qualified to operate in an accident environment) have primary and backup protection, meet the guidelines of Position 1 of RG 1.63, meet the requirements of GDC 50, and are acceptable. In regard to the few Class 1E circuits that do not have the required backup protection, it is the staff's opinion that the simultaneous occurrence of a LOCA, failure of the Class 1E circuit that is qualified to function in the LOCA environment, and failure of the circuit's primary protective device is unlikely during the first fuel cycle. Thus, the staff concludes that there is reasonable assurance that containment electric penetrations will not fail as a result of the event postulated above during the first fuel cycle.

8.3.3.6.2 Compliance of Penetration Protective Devices With Criteria of IEEE Std. 279-1971

In Section 1.8 of the FSAR, the applicant provided clarification as to how the guidelines of RG 1.63 are to be implemented in the Millstone design for protection of containment electrical penetrations. The applicant stated that overcurrent protective devices are not required to comply with the criteria in IEEE Std. 279-1971 (except Section 4.2) and need not be Class 1E or seismically qualified. Position 1 of RG 1.63, on the other hand, states that overcurrent protective devices should conform to the criteria of IEEE Std. 279. The proposed Millstone design did not meet the guidelines of Position 1 of RG 1.63.

In regard to testing, the applicant, by letter dated June 12, 1984, indicated that all penetration protective devices will be subject to periodic calibration and testing. This commitment for testing meets the guidelines of Position 1 of RG 1.63 and is acceptable.

In regard to independence of protective devices, the applicant, by letter dated June 12, 1984, indicated that backup penetration protective devices for non-Class 1E circuits are located in separate MCC-type enclosures from primary protection. This separation meets the guidelines of Position 1 of RG 1.63 and is acceptable for non-Class 1E circuits. In regard to independence of Class 1E penetration protective devices, the applicant indicated that the primary and backup protective devices will not, in all cases, be installed in a separate compartment. In justification, the applicant indicated that all failures within the common compartment are such as to result in the removal of power. A failure of one protective device will not cause the other protective device to fail in such a way that power will not be removed from the containment electric penetration circuit. On the basis of this justification, the staff concludes that this item has been acceptably resolved.

8.3.3.6.3 Application of RG 1.63 Positions to Both Class 1E and Non-Class 1E Circuits

Section 1.8 of the FSAR indicates, with respect to clarification of RG 1.63, that the single-failure provisions shall apply to both Class 1E and non-Class 1E overcurrent protection devices. The staff agrees with this clarification, and it is, therefore, acceptable.

8.3.3.6.4 Tripping Coordination Between Primary and Backup Penetration Protective Devices

In Section 1.8 of the FSAR, the applicant indicated, with respect to clarification of RG 1.63, that an acceptable method of compliance with the single-failure criterion may be the use of redundant or backup interrupting devices and that tripping coordination between primary and backup interrupting devices is not required. The staff agrees with this clarification, and it is, therefore, acceptable.

8.3.3.6.5 Use of Seismically Qualified or Class 1E Penetration Protective Devices

In Section 1.8 of the FSAR, the applicant indicated, with respect to clarification of RG 1.63, that unless required for other considerations, the protection schemes and fault-isolating devices need not be Class 1E or seismically qualified for protection of the penetration. This is in agreement with the staff's interpretation of the guidelines of RG 1.63 and is, therefore, acceptable with the following clarification. Even though circuit protective devices that protect containment electric penetration from failure of Non-Class 1E circuits need not be Class 1E, these devices are considered important to safety and, therefore, must be subject to a quality assurance program commensurate to their importance to safety and be so classified in Section 8.3.2 of the FSAR. Pending confirmation that these protective devices are included in Section 3.2 of the FSAR, this item is considered resolved.

8.3.4 Evaluation Findings

The review of the onsite ac and dc power systems for the Millstone plant covered single-line diagrams, station layout drawings, schematic diagrams, and descriptive information. The basis for acceptance of the onsite power systems in the staff's review was conformance of the design criteria and basis to the Commission's regulations as set forth in the GDC of Appendix A to 10 CFR 50. The staff concludes that the plant design is acceptable; meets the requirements of GDC 2, 4, 5, 17, 18, and 50; and conforms to the applicable guidelines of RGs, BTPs, and NUREG reports. It is acceptable, except as noted in the preceding sections.

9 AUXILIARY SYSTEMS

The staff has reviewed the design of the auxiliary systems necessary for safe reactor operation, shutdown, or fuel storage, or whose failure might affect plant safety, including their safety-related objectives and the manner in which these objectives are achieved. As noted in the following sections, the staff safety evaluation was performed in accordance with the applicable requirements of the SRP.

The auxiliary systems necessary for safe reactor operation or shutdown include the station service water system; reactor auxiliaries cooling water system; ultimate heat sink; the heating, ventilation, and air conditioning systems for the control room and essential safety features areas; essential portions of the main steam and feedwater systems; and the auxiliary feedwater system.

The auxiliary systems necessary to ensure the safety of the fuel storage facility include the new and spent fuel storage system, the spent fuel pool cooling and cleanup system, fuel-handling systems, and the spent fuel pool area ventilation system.

The staff also has reviewed other auxiliary systems to verify that their failure will not prevent safe shutdown of the plant or result in an unacceptable release of radioactivity to the environment. These systems include the demineralized water makeup system; the potable and sanitary water system; the condensate storage system; the turbine plant component cooling water system; the primary grade water storage system; the compressed air system; the heating, ventilation, and air conditioning systems for nonessential portions of the auxiliary building; the waste disposal building; and the turbine building.

9.1 Fuel Storage and Handling

9.1.1 New Fuel Storage

Millstone Unit 3 does not use a separate new fuel storage facility. Instead new fuel assemblies, after receipt and inspection in the new fuel inspection station, will be immediately placed in the spent fuel pool. This eliminates the need for dry storage racks. The staff considers this acceptable. SER Section 9.1.2 contains the staff's review of new and spent fuel storage.

9.1.2 Spent Fuel Storage

The spent fuel storage facility was reviewed in accordance with SRP Sections 9.1.1 and 9.1.2 (NUREG-0800). Conformance with the acceptance criteria, except as noted below, formed the basis for the staff's evaluation of the spent fuel storage facility with respect to the applicable regulations of 10 CFR 50.

The acceptance criteria for the spent fuel storage facility include various portions of the guidelines of American Nuclear Society (ANS) 57.2, "Design

Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations." The guidelines contained in the "Review Procedures" portion of the SRP were used instead of ANS 57.2. Additionally, the acceptance criteria include RG 1.115, "Protection Against Low-Trajectory Turbine Missiles." However, turbine missiles are evaluated separately in Section 3.5.1.3 of this SER.

The fuel-handling building houses the spent fuel storage facility and is designed to seismic Category I criteria as are the storage racks, pool liner, fuel pool gates, canals, and storage pools inside the building. The building is also designed against flooding and tornado missiles, which are discussed in Sections 3.4.1 and 3.5.2 of this SER.

The spent fuel storage facility is not located in the vicinity of any high-energy lines or rotating machinery. Therefore, physical separation is used to protect the spent fuel from internally generated missiles and the effects of pipe breaks, which are discussed in Sections 3.5.1.1 and 3.6.1 of this SER.

Thus, the requirements of GDC 2 and 4 and the guidelines of RGs 1.13 (Position C.1), 1.29 (Positions C.1 and C.2), and 1.117 (Positions C.1 through C.3) are satisfied. For a discussion of compliance with the guidelines of RG 1.115 refer to Section 3.5.1.3 of this SER.

The Millstone Unit 3 spent fuel storage facility is not shared with Millstone Units 1 or 2. Therefore, the requirements of GDC 5 are not applicable.

The new and spent fuel assemblies are stored in racks, which are located under water in the spent fuel pool. There are 756 fuel storage locations in 21 storage racks. For the initial reactor loading, 193 locations will be used to store the new fuel assemblies. For the successive reload batches, 64 locations will be reserved for new fuel assemblies until alternative storage provisions are required.

The spacing and the design of the racks are such that the effective multiplication factor (K_{eff}), for new or spent fuel, will not exceed 0.95 under all conditions, including fuel-handling accidents. The rack arrays have a center-to-center spacing of 10.35 in. Each storage cell incorporates a neutron absorber (boron carbide) encapsulated in stainless steel. The racks are designed to preclude the inadvertent placement of a fuel assembly in other than the prescribed spacing. The racks can withstand the impact of a dropped fuel assembly without unacceptable damage to the fuel and can withstand the maximum uplift forces exerted by the fuel-handling machine.

For a discussion of compliance with the guidelines of RG 1.13 (Positions C.3 and C.4) regarding crane interlocks and controlled ventilation leakage refer to Sections 9.1.4 and 9.4.2 of this SER. Thus, the requirements of GDC 61, as it relates to the facility design for fuel storage, and GDC 62, as it relates to the prevention of criticality by physical systems or configurations, are met.

Control room and local alarms are provided to alert the operator to high and low pool water level and high temperature in the fuel pool. The fuel-handling building also has a radiation monitoring system. These features satisfy the requirements of GDC 63.

On the basis of its review, the staff concludes that the spent fuel storage facility is in conformance with the requirements of GDC 2, 4, 61, 62, and 63 as they relate to protection of spent fuel against natural phenomena, missiles, and environmental effects, radiation protection, prevention of criticality, and monitoring fuel and waste storage, and with the guidelines of RGs 1.13, 1.29, and 1.117 as they relate to the facility's design, seismic classification, and protection against tornado missiles. Therefore, the spent fuel storage facility meets the acceptance criteria of SRP Sections 9.1.1 and 9.1.2.

Nuclear reactor plants include storage facilities for the wet storage of spent fuel assemblies. The safety function of the spent fuel pool and storage racks is to maintain the spent fuel assemblies in a subcritical array during all credible storage conditions. The staff has reviewed the compatibility and chemical stability of the materials (except the fuel assemblies) wetted by the pool water in accordance with SRP Section 9.1.2 and "Review and Acceptance of Spent Fuel Storage and Handling Application," April 1978.

The spent fuel racks will be constructed of type 304 stainless steel, except for the nuclear poison material. The spent fuel pool liner is constructed of stainless steel. Boraflex (Anderson, August 1979) sheets will be used as neutron absorbers in the high density spent fuel storage racks. Boraflex consists of boron carbide powder in a rubberlike silicone polymeric matrix. The spent fuel storage rack configuration is composed of individual storage cells interconnected to form an integral structure. The major components of the assembly are the fuel assembly cells, the Boraflex material, the wrapper, and the upper and lower spacer assemblies.

The upper end of the cell has a funnel-shaped flare for easy insertion of the fuel assembly. The wrapper surrounds the Boraflex material, but is open at the top and bottom to provide for venting of any gases that are generated. The Boraflex sheets sit in a cavity formed by the square inner stainless steel tube and the outer wrapper.

The pool contains oxygen-saturated demineralized water containing boric acid. The water chemistry control of the spent fuel pool has been reviewed elsewhere and found to meet NRC recommendations as discussed in Section 9.1.3 of this report.

The staff's review included FSAR Amendment 7 and letters from the applicant dated July 1 and August 1, 1983, and March 27, 1984.

The pool liner, rack lattice structure, and fuel storage tubes are stainless steel, which is compatible with the storage pool environment. In this environment of oxygen-saturated borated water, the corrosive deterioration of the type 304 stainless steel should not exceed a depth of 6.00×10^{-5} in. in 100 years, which is negligible relative to the initial thickness. Dissimilar metal contact corrosion (galvanic attack) between the stainless steel of the pool liner, rack lattice structure, fuel storage tubes, and the Inconel and Zircaloy in the spent fuel assemblies will not be significant because all of these materials are protected by highly passivating oxide films and are therefore at similar potentials.

Boraflex is composed of nonmetallic materials and therefore will not develop a galvanic potential in contact with the metal components. Boraflex has undergone extensive testing to study the effects of gamma irradiation in various environments and to verify its structural integrity and suitability as a neutron-absorbing material (Anderson, July 1979). The evaluation tests have shown that Boraflex is unaffected by the pool water environment and will not be degraded by corrosion. During tests performed at the University of Michigan, Boraflex was exposed to 1.03×10^{11} rads of gamma radiation with substantial concurrent neutron flux in borated water. These tests indicated that Boraflex maintains its neutron attenuation capabilities after being subjected to an environment of borated water and gamma irradiation. Irradiation will cause some loss of flexibility but will not lead to breakup of the Boraflex material. Long-term borated water soak tests at high temperatures were also conducted (Anderson, August 1978). The tests showed that Boraflex can withstand a borated water immersion of 240°F for 260 days without visible distortion or softening. Boraflex showed no evidence of swelling or loss of ability to maintain a uniform distribution of boron carbide.

The annulus space that contains the Boraflex material is vented to the pool at each storage tube assembly. Venting of the annulus will allow gas generated by the chemical degradation of the silicone polymer binder during heating and irradiation to escape and will prevent bulging or swelling of the inner stainless steel tube.

Tests (Anderson, August 1979) have shown that neither irradiation, environment, nor Boraflex composition has a discernible effect on the neutron transmission of the Boraflex material. The tests also showed that Boraflex does not possess leachable halogens that might be released into the pool environment in the presence of radiation. Similar conclusions were reached regarding the leaching of elemental boron from Boraflex. Boron carbide of the grade normally present in Boraflex will typically contain 0.1 weight percent of soluble boron. The test results have confirmed the encapsulation function of the silicone polymer matrix in preventing the leaching of soluble species from the boron carbide.

To provide added assurance that no unexpected corrosion or degradation of the materials will compromise the integrity of the racks, the applicant has committed to conduct a long-term fuel storage cell surveillance program. Surveillance samples consist of removable stainless steel cladding Boraflex sheets, which are prototypical of the fuel storage cell walls. These specimens will be removed and examined periodically.

On the basis of its evaluation above, the staff concludes that the corrosion that will occur in the spent fuel storage pool environment should be of little significance during the 40-year life of the plant. Components in the spent fuel storage pool are constructed of alloys that have a low differential galvanic potential between them and have a high resistance to general, localized, and galvanic corrosion. Tests under irradation and at elevated temperatures in borated water indicate that the Boraflex material will not undergo significant degradation during the expected service life of 40 years.

The staff further concludes that the environmental compatibility and stability of the materials used in the spent fuel storage pool are adequate on the basis of the test data cited above and actual service experience in operating reactors.

The staff has reviewed the surveillance program and concludes that the monitoring of the materials in the spent fuel storage pool, as proposed by the applicant, will provide reasonable assurance that the Boraflex material will continue to perform its function for the design life of the pool. The materials surveillance program delineated by the applicant will reveal any instances of deterioration of the Boraflex material that might lead to the loss of neutron-absorbing power during the life of the spent fuel racks. The staff does not anticipate that such deterioration will occur. This monitoring program will ensure that, in the unlikely event that the Boraflex material will deteriorate in this environment, the applicant and the NRC will be aware of it in sufficient time to take corrective action.

The staff, therefore, finds that the implementation of a monitoring program and the selection of appropriate materials of construction by the applicant meet the requirements of GDC 61 by providing a capability to permit appropriate periodic inspection and testing of components, and GDC 62, by preventing criticality by maintaining the structural integrity of components and of the boron poison and are, therefore, acceptable.

9.1.3 Spent Fuel Pool Cooling and Cleanup System

The spent fuel pool cooling and cleanup system was reviewed in accordance with SRP Section 9.1.3 (NUREG-0800).

The spent fuel pool cooling and cleanup system is designed to maintain water quality and clarity and to remove decay heat generated by spent fuel assemblies in the fuel pool. The cleanup train is also designed to purify water in the refueling cavity and refueling water storage tank. The system includes all components and piping from the inlet to the exit from the spent fuel pool and refueling cavity, and piping used for fuel pool makeup, from the refueling water storage tank and the cleanup filter/demineralizers to the point of discharge to the radwaste system. The fuel pool cooling system consists of two full-capacity fuel pool cooling pumps and two full-capacity pool coolers with associated piping and valves. Cooling for the fuel pool coolers is provided by the reactor plant component cooling water system. This system is discussed in SER Section 9.2.2. The fuel pool cleanup system consists of two purification pumps, one demineralizer, one coarse filter, two prefilters, and one postfilter.

The fuel pool cooling system is housed in the seismic Category I, flood- and tornado-protected fuel building and containment (see Sections 3.4.1 and 3.5.2 of this SER). The cooling system is completely separate from the cleanup train and is designed to Quality Group C and seismic Category I requirements, as is the reactor plant component cooling water system. The cleanup train is designed to Quality Group D and nonseismic requirements. Its location ensures that failure in any portion of the train would have no adverse effect on safety-related equipment.

The various components of the fuel pool cooling systems are located in shielded cubicles or are separated from other moderate- and high-energy piping systems and are thus protected against the effects of internally generated missiles, pipe whip, and jets (see Sections 3.5.1.1 and 3.6.1 of this SER). This design satisfies the requirements of GDC 2 and 4 by showing compliance with the guidelines of RGs 1.13 (Positions C.1 and C.2), 1.26 (Position C.2), and 1.29 (Positions C.1 and C.2).

Millstone Unit 3 is provided with a separate spent fuel pool cooling and cleanup system, and there is no sharing with Millstone Units 1 or 2; thus, the requirements of GDC 5 are not applicable.

Provisions have been made for routine visual inspection of the spent fuel pool cooling system components. One fuel pool cooling pump and one fuel pool cooler are in operation at all times. The spare fuel pool cooling pump and fuel pool cooler will be operated periodically. Thus, the requirements of GDC 45 and 46 are satisfied.

The spent fuel pool cooling system is designed to handle the accumulated heat load from 1,836 fuel assemblies, including a full-core offload of 193 fresh fuel assemblies. The applicant stated that under maximum normal heat load, the pool temperature would be maintained at or below 125°F. This is acceptable because it is less than the staff's acceptance criterion of a maximum temperature of 140°F for a normal heat load. The applicant heat load calculations assumed single failure.

The maximum abnormal heat load is based on a full-core offload 10 days after the last normal refueling outage and a maximum storage load of 1,836 fuel assemblies. Under these conditions, the maximum pool temperature would reach 149°F. Following a design-basis accident with loss of power, the reactor plant component cooling water will not be available for 4 hours because of loading considerations on the emergency generators. Under this condition, the spent fuel pool water temperature will go above 149°F and reach 165°F in 4 hours, 200°F in 12.5 hours.

The applicant stated that decay heat removal is based on Westinghouse-generated curves for the fuel that will actually be supplied to Millstone Unit 3 and these curves are more appropriate than the analysis recommended in BTP ASB 9-2. The staff performed the heat load calculations for storage of 1,836 fuel assemblies from Millstone Unit 3, and concluded that the spent fuel pool cooling system meets the requirements of GDC 44, "Cooling Water," and BTP ASB 9-2 regarding decay heat removal capacity.

No connections are provided to the spent fuel pool that may cause the pool water to be drained below a safe shielding level. All lines that connect to the pool and extend below the safe level of the pool water are equipped with syphon breakers, check valves, or other means to prevent inadvertent pool drainage. Normal makeup water to the fuel pool is provided from the primary grade water system (see SER Section 9.2.8) or, as a backup, from the seismic Category I refueling water storage tank. Water from the safety-related service water system is also available. To prevent contamination of the pool from the service water, a spool piece is included at the pool end of the service water piping with a blind flange in place. Thus, the requirements of GDC 61, "Fuel Storage and Handling and Radioactivity Control," and the guidelines of RG 1.13 concerning fuel pool design are satisfied.

The system incorporates control room alarmed pool water level, water temperature, and building radiation level monitoring systems, thus satisfying the requirements of GDC 63.

On the basis of its review, the staff concludes that the spent fuel pool cooling and cleanup system is in conformance with the requirements of GDC 2, 4, 44, 45, 46, 61, and 63 and the guidelines of RGs 1.13, 1.26, 1.29, and BTP ASB 9-2 with respect to protection against natural phenomena, missiles, pipe break effects, decay heat removal, inservice inspection, functional testing, radiation protection, performance monitoring, and quality group classification. The spent fuel pool cooling and cleanup system complies with the acceptance criteria of SRP Section 9.1.3.

The spent fuel pool cleanup system is designed to maintain optical clarity and to remove corrosion products, fission products, and impurities from the spent fuel pool water. Water purity and clarity in the spent fuel pool, refueling cavity, and refueling water storage tank (RWST) are maintained by filtering and demineralizing the pool water through filters and a demineralizer. The spent fuel pool cleanup system consists of two purification pumps, two purification prefilters, one coarse filter, one purification demineralizer, and one post-filter. Either pump can be used with the prefilter to filter the water in the RWST, the refueling cavity, or the fuel pool. The fuel pool filters and demineralizer are located in shielded cubicles to minimize operator exposure.

The spent fuel pool water is sampled weekly for pH, conductivity, chloride, fluoride, turbidity, and total gamma activity. Chloride and fluoride limits are 0.15 ppm each, and an acceptable pH may vary between 4.2 and 10.5. Boron concentration is monitored before refueling operations.

The staff's review included Amendment 7 of the FSAR.

The spent fuel pool cleanup system has been reviewed in accordance with SRP Section 9.1.3 (NUREG-0800).

The staff has determined that the spent fuel pool cleanup system (1) provides the capability and capacity of removing radioactive materials, corrosion products, and impurities from the pool water and thus meets the requirements of GDC 61 as it relates to appropriate filtering systems for fuel storage; (2) is capable of reducing occupational exposure to radiation by removing radioactive products from the pool water and thus meets the requirements of 10 CFR 20.1(c) as it relates to maintaining radiation exposures as low as is reasonably achievable; (3) confines radioactive materials in the pool water by means of the demineralizer and filters and thus meets Position C.2.f(2) of RG 8.8 as it relates to reducing the spread of contaminants from the source; and (4) removes suspended impurities from the pool water by filters and thus meets Position C.2.f(3) of RG 8.8 as it relates to removing crud through physical actions.

On the basis of the above evaluation, the staff concludes that the spent fuel pool cleanup system meets GDC 61, 10 CFR 20.1(c), and the appropriate sections of RG 8.8 and, therefore, is acceptable.

9.1.4 Light Load Handling Systems

The light load handling system was reviewed in accordance with SRP Section 9.1.4 (NUREG-0800). Conformance with the acceptance criteria, except as noted below, formed the basis for the staff's evaluation of the light load handling system with respect to the applicable regulations of 10 CFR 50.

The acceptance criteria for the light load handling systems include the guidelines of American National Standards Institute/American Nuclear Society (ANSI/ANS) 57.1. The guidelines contained in the "Review Procedures" portion of the SRP were used instead of ANSI/ANS 57.1.

Any load that weighs less than one fuel assembly and its associated handling tool is defined as a light load. The light load handling system is related to refueling and consists of all components and equipment used from the handling of new fuel from the receiving station to the loading of spent fuel into the shipping cask. The system includes the equipment designed to facilitate the periodic refueling of the reactor and includes the refueling machine, spent fuel pool bridge and hoist, new fuel elevator, new fuel receiving and transfer cranes, fuel transfer system, and associated handling tools and devices. The handling of fuel during refueling is controlled by a series of interlocks to ensure that fuel-handling procedures are maintained. The design ensures that no failure will result in release of radioactivity in excess of that assumed in the design-basis fuel-handling accident.

The refueling machine is a rectilinear bridge and trolley system with a vertical mast extending down into the water. The machine is used to handle new and spent fuel assemblies within the reactor vessel and refueling cavity inside the containment. Electrical interlocks and limit switches on the bridge and trolley drives prevent damage to the fuel assemblies. Redundant limit switches on the mast winch prevent a fuel assembly from being raised above a safe shielding water depth.

The spent fuel pool bridge and hoist is wheel mounted with a monorail hoist. It is used exclusively for handling fuel assemblies within the spent fuel storage area, including the spent fuel racks and the spent fuel cask.

The new fuel elevator is used to lower new fuel assemblies, one at a time, to the bottom of the fuel storage area where they can be transported by the spent fuel pool bridge crane to the storage racks.

The new fuel receiving crane is used to unload the new fuel shipping containers from the truck to a storage location. The new fuel transfer crane is used to remove fuel assemblies from the shipping container to the inspection area and then to the new fuel elevator for subsequent storage in the fuel pool.

The fuel transfer system includes an underwater electric motor-driven transfer car that runs on tracks extending from the refueling cavity through the transfer tube and into the refueling canal. Fuel assemblies are placed on the car in the refueling cavity by the refueling machine; they are removed in the spent fuel pool by the spent fuel pool bridge crane after passing through the transfer tube.

The entire light load handling system is housed within the fuel handling building and the containment, which are seismic Category I, flood- and tornado-protected structures. Although fuel-handling-system components are not required to function after a safe shutdown earthquake, critical components of the fuel-handling system are designed to seismic Category I requirements so that they will not fail in a manner that results in unacceptable consequences such as fuel damage or damage to safety-related equipment. The spent fuel pool bridge and hoist that travels over the spent fuel storage racks, as described above, is designed to seismic Category I requirements. The design thus satisfies the requirements of GDC 2 and the guidelines of RGs 1.29 (Positions C.1 and C.2) and 1.13 (Positions C.1 and C.6).

Millstone Unit 3 has its own fuel-handling system, and there is no sharing with Millstone Units 1 or 2. Therefore, the requirements of GDC 5 are not applicable.

The applicant has stated that when light loads are dropped over the fuel pool or reactor vessel from their maximum normal elevation, they will not result in greater fuel damage than that assumed for a dropped fuel assembly in the design-basis fuel-handling accident. Hence, the resulting radiological releases would be less than those assumed in the fuel-handling accident. Thus, the staff concludes that the requirements of GDC 61 and 62 and the guidelines of RG 1.13 (Position C.3) are satisfied.

On the basis of its review, the staff concludes that the light load handling system is in conformance with the requirements of GDC 2, 5, 61, and 62 as they relate to protection against natural phenomena, shared systems, and safe fuel handling, including prevention of criticality, and with the guidelines of RGs 1.13 (Positions C.1, C.3, and C.6) and 1.29 (Positions C.1 and C.2) with respect to overhead crane interlocks, prevention of unacceptable releases in fuel-handling accidents, and maintaining plant safety in a seismic event. It is, therefore, acceptable. The light load handling system meets the acceptance criteria of SRP Section 9.1.4.

9.1.5 Overhead Heavy Load Handling System

The applicant has recently provided information on the overhead heavy load handling system. This information is under review by the staff and its consultants.

NUREG-0612 was transmitted to the applicant for action by NRC generic letters dated December 22, 1980, and February 3, 1981. NUREG-0612 resolved Generic Task A-36 and provides guidelines for necessary changes to ensure safe handling of heavy loads once the plant becomes operational. Enclosure 2, attached to the December 22, 1980 generic letter, identified a number of interim measures dealing with safe load paths, procedures, operator training, and crane inspections, testing, and maintenance.

The staff is imposing the following licensing conditions to resolve this issue.

(1) Before initiating the first refueling, the applicant shall have implemented the guidelines of Sections 5.1.1 and 5.3 of NUREG-0612 (Phase I).

(2) Before startup following the first refueling outage, the applicant shall have made commitments acceptable to the NRC regarding the guidelines of Sections 5.1.2 through 5.1.6 of NUREG-0612 (Phase II).

9.2 Water Systems

9.2.1 Station Service Water System

The service water system was reviewed in accordance with SRP Section 9.2.1 (NUREG-0800).

The service water system (SWS) performs both safety-and nonsafety-related functions and supplies cooling water to the plant from the ultimate heat sink, Long Island Sound, which is discussed in detail in Section 9.2.5 of this SER.

The SWS for Millstone Unit 3 consists of two trains. These trains provide cooling water to reactor component cooling water heat exchangers, containment recirculation coolers, emergency diesel engine coolers, charging pumps cooler, safety injection pump cooler, control building air conditioning (A/C) water chiller, residual heat removal pump room ventilation unit, containment recirculation pump room ventilation unit, service water strainer backwash, service water pump lubrication, motor control center and rod control area A/C unit, and postaccident liquid sample cooler. The SWS also provides cooling water to nonessential turbine plant component cooling heat exchangers and circulating water lubrication. The SWS provides an emergency source of makeup water to the fuel pool and an emergency backup source of water to the auxiliary feedwater system and control building chilled water system.

Each train of the SWS contains two half-capacity service water pumps, two service water self-cleaning strainers, two booster pumps, piping, and valves. The booster pumps provide the additional head required and pump service water to the control room A/C chiller. The SWS discharge line from the containment recirculation coolers contains radiation monitors, and the SWS can be isolated on a high radiation alarm. Although there are single isolation valves on the inlet and discharge side of the service water side of the containment coolers, this is acceptable because of the isolation valves on the containment recirculation system side which can also be closed on a high radiation alarm. Normally, one service water pump in each train will operate with the other on standby. The SWS pumps are located in the safety-related portion of the circulating and service water pumphouse building. Net positive suction head will be maintained at the pump suction because of its location at el -13 ft (MSL), 5 ft below the design low water level. Each service water header has connections to and from the chemical feed chlorination system for the addition of chlorine to the SWS to inhibit biological fouling. The SWS discharges into a quarry through the safety-related circulating water discharge tunnel. The quarry is connected to the Long Island Sound by an open channel.

All safety-related components of the SWS are housed in seismic Category I, flood- and tornado-protected structures. The service water pumps are located in pairs in watertight cubicles in the intake structure. The applicant has shown that the possibility of water entering the pumphouse through the pump shaft is not credible. Service water cubicle floor drains are discussed in Section 9.3.3. Underground piping of the SWS is protected from natural phenomena. The system itself is designed to seismic Category I, Quality Group C requirements. Thus, the requirements of GDC 2 and the guidelines of RG 1.29 are satisfied.

The SWS is designed to meet the single-failure criterion. Power is supplied to redundant SWS pumps from separate emergency buses. Each service water pump can supply the minimum cooling water requirements during a design-basis accident (DBA) with loss of power and during cold shutdown with loss of power. Therefore, the staff concludes that GDC 44 is satisfied.

The SWS is not shared with the other Millstone units with the exception of the ultimate heat sink. Therefore, the staff concludes that the requirements of GDC 5 are not applicable.

The SWS design incorporates provisions for functional testing and inspection. The service water pumps in each train normally will be operated alternately. The service water flow and temperature data for the reactor plant component cooling water heat exchanger will be taken periodically to indicate possible biological fouling problems. Therefore, the staff concludes that the requirements of GDC 46 are satisfied.

On the basis of the above, the staff concludes that the SWS meets the requirements of GDC 2, 5, 44, 45, and 46 with respect to the system's protection against natural phenomena, sharing of systems, capability for transferring the required heat loads, inservice inspection, and testing and meets the guidelines of RG 1.29 (Positions C.1 and C.2) with respect to the system's seismic classification. The SWS meets the requirements of SRP Section 9.2.1.

9.2.2 Cooling Water Systems

The reactor auxiliary cooling water systems were reviewed in accordance with SRP Section 9.2.2 (NUREG-0800).

The reactor auxiliary cooling water systems consist of the reactor plant component cooling water, chilled water, neutron shield tank cooling, charging pumps cooling, safety injection pumps cooling, and condensate demineralizer component cooling water systems. These systems are used individually or in combination to provide cooling water for heat removal from reactor plant components. Part of the reactor plant component cooling water system and the entire charging and safety injection cooling water systems are safety related.

9.2.2.1 Reactor Plant Component Cooling Water System

The reactor plant component cooling water (RPCCW) system is a closed-loop cooling water system that transfers heat from reactor auxiliary systems to the service water system during plant operation and normal and emergency shutdown. It provides an intermediate barrier between radioactive or potentially radioactive heat sources and the service water system.

The RPCCW system includes three half-capacity motor-driven cooling water pumps, three half-capacity heat exchangers, a surge tank, a chemical addition tank, and associated piping and valves. The system is designed to reduce the reactor coolant temperature from 350°F to 120°F in 20 hours with service water available at 75°F.

The system provides cooling water to the safety-related seal water heat exchangers, letdown heat exchanger, fuel pool coolers, residual heat removal (RHR) heat exchangers and RHR pump seal coolers, reactor coolant pumps thermal barriers and bearing oil coolers, and excess letdown heat exchanger. It provides cooling water to the containment air recirculation cooling coils and neutron shield tank cooler on loss of power and containment isolation signal. It provides cooling water to nonsafety-related components such as the thermal regeneration chiller, radioactive liquid and gaseous waste system, chilled water system refrigeration units, containment penetration coolers, instrument air compressor coolers, reactor plant sampling system, and auxiliary condensate system cooler. It provides makeup water to the safety injection and charging pumps cooling water surge tanks, thermal regeneration chiller surge tanks, and fuel transfer system.

All safety-related portions of the RPCCW system are located inside seismic Category I, tornado-, missile-, and flood-protected buildings. Safety-related piping and equipment are designed to seismic Category I and Quality Group C requirements. Thus, the requirements of GDC 2 and the guidelines of RG 1.29 (Positions C.1 and C.2) are satisfied.

No portion of this system is shared with other Millstone units. Failure of the nonsafety-related portion of the system components will not affect the performance or reliability of the safety-related components. Thus, the requirements of GDC 5 are satisfied.

The system is designed to meet the single-failure criterion with two redundant trains to serve those components essential for safe shutdown. During normal operation, two RPCCW pumps and two heat exchangers accommodate the heat removal load. A spare pump and heat exchanger are provided to allow for pump or heat exchanger maintenance. One RPCCW pump is fed by one emergency bus, and the second pump is fed by the second emergency bus. The spare pump can be manually connected to either emergency bus. During accident conditions, one pump and one heat exchanger train are sufficient to accommodate the heat removal load. The surge tank has sufficient capacity to accommodate 30-day system leakage. The tank is partitioned so that loss of water from one compartment will affect only one pump. Makeup water is provided by the condensate system.

The staff identified a concern regarding loss of component cooling water flow to the reactor coolant pumps (RCPs) as a result of a single failure in the common supply line, which might result in an unacceptable locked rotor condition. In response to this concern the applicant has indicated that the RCPs can function satisfactorily for 20 min without component cooling water flow. Low component cooling water flow alarms are provided in the control room to indicate a loss of component cooling water supply. This allows the operator sufficient time to trip the pumps before unacceptable damage occurs. Redundant high oil cooler temperature alarms and high bearing temperature alarms are also provided in the control room. The containment supply and return headers are cross-connected so that if one RPCCW pump fails, all four RCPs are supplied with cooling water from the unaffected pump. This, therefore, resolves the staff's concern about single failure. Thus, the requirements of GDC 44 are satisfied.

During normal operation, all portions of the RPCCW system are either in continuous or intermittent operation. Availability of the remaining pumps will be ensured by periodic tests and inspection according to the plant Technical Specifications. The system components are located in accessible areas to permit inservice inspection, as required. Thus, the requirements of GDC 45 and 46 are satisfied.

On the basis of its review, the staff concludes that the RPCCW system meets the requirements of GDC 2, 5, 44, 45, and 46 with respect to protection against natural phenomena, shared systems, decay heat removal capability, inservice inspection, and functional testing and meets the guidelines of RG 1.29 (Positions C.1 and C.2) with respect to the system's seismic classification. It is, therefore, acceptable. The RPCCW system meets the acceptance criteria of SRP Section 9.2.2.

9.2.2.2 Chilled Water Systems

The chilled water system is a closed-loop nonnuclear safety class system with the exception of the containment isolation valves and the piping between them, which are Safety Class 2. The system provides cooling water for the refueling water cooler, service building air conditioning (A/C) units, motor control center and rod control area A/C units, containment air recirculation cooling coils, neutron shield tank cooler, and various components inside the containment structure. During loss of power or after receiving a containment isolation phase A signal, cooling water supply to two of the three containment air recirculation coolers and the neutron shield tank coolers is transferred to the reactor plant component cooling water system.

The chilled water system consists of three half-capacity, self-contained chillers, three circulating pumps, a surge tank, and associated piping and valves. Makeup water to the surge tank and heat sink for the chiller condenser is provided by the RPCCW system described in SER Section 9.2.2.1. A failure in the chilled water system will not affect the safety-related portions of the RPCCW system.

On the basis of the above, the staff concludes that the chilled water system design is acceptable and meets the acceptance criteria of SRP Section 9.2.2.

9.2.2.3 Neutron Shield Tank Cooling System

Tne neutron shield tank cooling system is a nonsafety-related closed cooling water system. It provides cooling water to the neutron shield tank, which is heated by neutron and gamma radiation from the reactor. It is a natural circulation system and consists of two full-capacity neutron shield tank coolers, a surge tank, and associated piping and valves. Makeup water to the system is provided from the nonsafety primary grade water system. Heat is rejected in the cooler to the chilled water system or the reactor plant component cooling water system on loss of power or containment isolation phase A signal.

On the basis of the above, the staff concludes that the design of the neutron shield tank cooling system is acceptable and meets the acceptance criteria of SRP Section 9.2.2.

9.2.2.4 Charging Pumps Cooling System

The charging pumps cooling system is a safety-related closed cooling water system that transfers heat load from the charging pumps lubricating oil coolers to the service water system. This system consists of two full-capacity pumps, two full-capacity charging pump coolers, a charging pump cooling surge tank, and associated piping and valves. Makeup water to the system is provided by the reactor plant component cooling water system described in Section 9.2.2.1.

This system is not shared with any other Millstone units. All components are designed to seismic Category I and Quality Group C requirements. The system components are located inside the seismic Category I, tornado-, missile-, and flood-protected auxiliary building. Thus, the requirements of GDC 2 and 5 with respect to protection against natural phenomena and shared systems and the guidelines of RG 1.29 (Positions C.1 and C.2) with respect to seismic classification are met.

Redundant components and piping are used throughout the charging pumps cooling system. The pumps are powered from redundant emergency buses. A single failure of any component in either cooling loop will not affect operation of the redundant train. The charging pump coolers are connected to separate service water supply and return lines. Failure of one cooling pump will automatically start the standby pump.

Redundant suction and discharge pump headers which are cross-connected so that either pump can supply cooling water to any charging pump lube oil cooler are protected by double automatic isolation valves. The surge tank capacity (1,000 gal) is adequate for 30 days for any system leakage in the event of a design-basis loss-of-coolant accident (LOCA). Thus, the requirements of GDC 44 are satisfied.

The charging pumps cooling system will be in continuous operation with essential system parameters continuously monitored and indicated in the control room. The system pumps will be alternated in service on a scheduled basis. All components are accessible for inservice inspection. Therefore, the requirements of GDC 45 and 46 are satisfied.

On the basis of the above, the staff concludes that the charging pumps cooling system meets the requirements of GDC 2, 5, 44, 45, and 46 with respect to protection against natural phenomena, shared systems, decay heat removal capability, inservice inspection, and functional testing and meets the guidelines of RG 1.29 (Positions C.1 and C.2) with respect to the system's seismic classification. It is, therefore, acceptable. The charging pumps cooling system meets the acceptance criteria of SRP Section 9.2.2.

9.2.2.5 Safety Injection Pumps Cooling System

The safety injection pumps cooling system is a safety-related closed cooling water system that transfers heat load from the safety injection pumps bearing oil to the service water system. This system consists of two full-capacity pumps, two full-capacity safety injection pump coolers, a surge tank, and associated piping and valves. Makeup water to the system is provided by the safety-related portion of the reactor plant component cooling water system.

This system is not shared with any other Millstone units. All components are designed to seismic Category I and Quality Group C requirements. The system components are located inside a seismic Category I, tornado-, missile-, and flood-protected building.

Thus, the requirements of GDC 2 and 5 with respect to protection against natural phenomena and shared systems and the guidelines of RG 1.29 with respect to seismic classification are met.

Redundant components and piping are used throughout the safety injection pumps cooling system. The cooling pump for each safety injection pump is powered from the same emergency bus as its associated safety injection pump. The use of redundant components, piping, and emergency buses ensures that the system can withstand a single failure. Each of two compartments in the surge tank has a capacity of 500 gal, which is adequate for 30 days should any system leakage take place during a design-basis LOCA when makeup water may not be available. The coolers are connected to separate service water supply and return lines. Thus, the requirements of GDC 44 are satisfied.

The cooling loops of both safety injection pumps will be operated periodically with essential system parameters monitored and indicated in the control room. All components are accessible for inservice inspection. Tests and inspection will be performed in accordance with SER Sections 3.9.6 and 6.6. Therefore, the requirement of GDC 45 and 46 are satisfied.

On the basis of its review, the staff concludes that the safety injection pumps cooling system meets the requirements of GDC 2, 5, 44, 45, and 46 with respect to protection against natural phenomena, shared systems, decay heat removal capability, inservice inspection, and functional testing and meets the guidelines of RG 1.29 (Positions C.1 and C.2) with respect to the system's seismic classification. It is, therefore, acceptable. The safety injection pumps cooling system meets the acceptance criteria of SRP Section 9.2.2.

9.2.2.6 Condensate Demineralizer Component Cooling Water System

The condensate demineralizer component cooling water (CDCCW) system is a nonsafety-related closed cooling water system. It provides cooling water to the nonsafety-related components such as the regenerated system distillate cooler, evaporator, and bottom sample cooler (See Section 11.2). The system consists of one CDCCW pump, one heat exchanger, and associated piping and valves. The heat from the CDCCW system is removed by the circulating water traveling screen wash and disposal system. This system shares its surge tank with the Millstone Unit 2 CDCCW system. Failure of any portion of this system will not damage any safety-related components or system.

On the basis of its review, the staff concludes that the design of the CDCCW system is acceptable and meets the acceptance criteria of SRP Section 9.2.2.

9.2.3 Demineralized Water Makeup System

The demineralized water makeup system was reviewed in accordance with SRP Section 9.2.3 (NUREG-0800). Conformance with the acceptance criteria formed the basis for the staff's evaluation of the demineralized water makeup system with respect to the applicable regulations of 10 CFR 50.

The nonsafety-related, Quality Group D, nonseismic Category I demineralized water makeup system provides demineralized water, as required, to the 300,000-gal condensate storage tank (see Section 9.2.6), 150,000-gal condensate surge tank (see Section 9.2.6), 360,000-gal demineralized water storage tank (see Section 10.4.9), or, after additional treatment in the primary grade water deaerator, to the two 100,000-gal-capacity primary grade water storage tanks (see Section 9.2.8). Water to the demineralized makeup system is provided through the potable water system from the Waterford Public Water Supply.

The demineralized water makeup system includes the supply water treating system and the waste water treating system. The supply water treating system consists of a water treatment storage tank and two trains of filtration and demineralization components that can produce demineralized water at the rate of 248 gpm. The waste water treating system is designed to accept chemical wastes from the demineralization trains and, after treatment and neutralization, to discharge the waste water into the circulating water discharge tunnel.

This system has no safety-related functions. It is designed to meet BTPs ASB 3-1 and MEB 3-1 (Section 3.6) as they relate to breaks in high- and moderate-energy-piping systems outside containment. Instrumentation, including alarms, has been provided at the water treatment control panel to prevent delivery of offspecification water to all systems. Failure of the system does not affect the capability to safely shut down the plant as described above; thus, the requirements of GDC 2 and 5 and the guidelines of RG 1.29 (Positions C.1 and C.2) are met.

On the basis of its review, the staff concludes that the demineralized water makeup system meets the requirements of GDC 2 and 5 with respect to protection against natural phenomena and shared systems because its failure does not affect the functions of safety-related systems and meets the guidance of RG 1.29 (Positions C.1 and C.2) concerning its seismic classification. It is, therefore, acceptable. The demineralized water makeup system meets the acceptance criteria of SRP Section 9.2.3.

9.2.4 Potable and Sanitary Water Systems

The potable and sanitary water systems were reviewed in accordance with SRP Section 9.2.4 (NUREG-0800).

The nonsafety-related, Quality Group D, nonseismic Category I potable water system provides clean water for drinking and sanitary purposes and makeup water to the demineralized water makeup system and to components such as the control building chiller, the instrument air compressor, and the seal water for vacuum priming pumps. It can also provide makeup water to the demineralized water storage tank through a temporary spool piece, as additional backup for cooling water for cold shutdown. The potable and sanitary water systems include all components from the connections to the town of Waterford's public water supply main to the point of discharge. The sanitary water system is also nonsafety related, Quality Group D, nonseismic Category I.

The potable water system uses backflow preventers, an air gap, and vacuum breakers throughout the system to prevent any possible contamination from radioactivity, chlorine, or other flushing activities conducted on systems throughout the plant. The failure of the potable or sanitary water systems will not affect plant safety. Thus, the requirements of GDC 60 are met.

On the basis of its review, the staff concludes that the potable and sanitary water systems meet the requirements of GDC 60 with respect to prevention of release of potentially radioactive water and are, therefore, acceptable. The potable and sanitary water systems meet the acceptance criteria of SRP Section 9.2.4.

9.2.5 Ultimate Heat Sink

The ultimate heat sink (UHS) was reviewed in accordance with SRP Section 9.2.5 (NUREG-0800).

The UHS for Millstone Unit 3 is Long Island Sound. Sensible heat removed from both safety- and nonsafety-related cooling systems during normal operation, shutdown, and accident conditions is discharged by the service water and circulating water systems.

The safety-related portions of the circulating and service water pumphouse are designed to meet seismic Category I requirements and to withstand the effects of all natural phenomena and missiles. The service water pump compartments and the pumps are designed to the high and low water conditions described in SER Section 2.4.11. The circulating water discharge tunnel is also designed to seismic Category I requirements.

Millstone Units 1, 2, and 3 all discharge their cooling water into the quarry, which is connected to the Long Island Sound, through an open channel. If the quarry outlet should become blocked because of a seismic event or debris clogging, the service water from all three units would flood the quarry discharge area and eventually drain off into the Long Island Sound without restricting the system's heat removal capability in the three units or flooding any safety-related structures of Millstone Units 1, 2, and 3.

The UHS meets the cooling requirements for a 30-day period including those for core decay heat, sensible heat, and plant auxiliary systems in accordance with the guidelines of RG 1.27.

On the basis of its review, the staff concludes that the UHS meets the requirements of GDC 2 and 44 with respect to protection against natural phenomena and decay heat removal capacity and the guidelines of RGs 1.27 with respect to the design and functional requirements of the UHS, 1.29 with respect to the seismic design classification, 1.102 with respect to protection from flooding, and 1.117 with respect to protection from tornado missiles. It is, therefore, acceptable. The UHS meets the applicable acceptance criteria of SRP Section 9.2.5.

9.2.6 Condensate Storage Facilities

The condensate storage and transfer system was reviewed in accordance with SRP Section 9.2.6 (NUREG-0800).

The nonsafety-related, Quality Group D, nonseismic Category I condensate makeup and drawoff system includes all components and associated piping from the condensate surge tank (150,000 gal) and condensate storage tank (300,000 gal) to the points of connection or interfaces with other systems. The condensate surge tank provides the makeup and drawoff requirements of the main condensate system and the auxiliary boiler condensate system. The condensate storage tank provides makeup water as required to the condensate surge tank, auxiliary feedwater system, reactor plant component cooling water system, turbine plant component cooling system, and other makeup systems requiring condensate quality water. All of these systems, with the exception of the reactor plant component cooling water system, are normally isolated from the condensate storage tank. Flow is provided on demand by operation of either control or manual valves located in the interfacing system. Flow head from the condensate tank is provided by one of the two component cooling water makeup pumps.

The nonsafety-related condensate storage and condensate surge tanks are located in the yard. Freeze protection is provided by heat tracing piping and by circulating water through a pump and heater loop. No portion of the condensate storage facility piping is classified as moderate- or high-energy piping. Protection from flooding for safety-related systems is discussed in Section 3.4.1 of this SER.

The Millstone Unit 3 condensate facilities are not shared with other Millstone units on site. Thus, the requirements of GDC 5 are not applicable.

The system was evaluated and found to have no function necessary for achieving safe reactor shutdown conditions or for accident prevention or mitigation. The safety-related systems that receive water from the condensate storage tank have either a safety class primary water supply or have sufficient storage capacity to perform their safety functions without additional makeup. Thus, the requirements of GDC 44, 45, and 46 are not applicable.

On the basis of this review, the staff concludes that the condensate storage and drawoff system is acceptable and meets the acceptance criteria of SRP Section 9.2.6.

9.2.7 Turbine Plant Component Cooling Water System

The nonsafety-related turbine plant component cooling water system was reviewed in accordance with the guidelines of SRP Section 9.2.3 (NUREG-0800).

The nonsafety-related, Quality Group D, nonseismic Category I turbine plant component cooling water system removes heat from various nonsafety-related turbine plant components such as turbine oil coolers, electrohydraulic control fluid coolers, air compressors, after coolers, generator hydrogen coolers, excitor air coolers, and feedwater and condensate pump coolers.

The turbine plant component cooling water is designed as a closed cooling system and consists of a surge tank, a chemical addition tank, three half-capacity circulating pumps, three half-capacity heat exchangers, and associated piping and valves. The service water provides the heat sink for this system. Makeup water to the surge tank is provided by the condensate storage tank. System water chemistry for corrosion inhibition is maintained by chemical addition.

The components of this system are located in the turbine building, auxiliary boiler enclosure, and warehouse facilities. Protection from flooding of safety-related systems is discussed in Sections 3.4.1 and 9.3.3 of this SER.

The system was evaluated and found to have no function necessary for achieving safe reactor shutdown conditions or for accident prevention or mitigation. Thus, the requirements of GDC 44, 45, and 46 are not applicable.

On the basis of this review, the staff concludes that the turbine plant component cooling water system is acceptable and meets the acceptance criteria of SRP Section 9.2.2.

9.2.8 Primary Grade Water System

The nonsafety-related primary grade water system was reviewed in accordance with the guidelines of SRP Section 9.2.6 (NUREG-0800).

The primary grade water system is a nonnuclear safety-related system with the exception of the containment isolation valves and the piping between them, which are Safety Class 2.

This system provides a reliable source of water for the reactor coolant makeup, cooling water for the pressurizer relief tank, and miscellaneous services such as mixing water for the boric acid batching tank and flushing resin from the ion exchangers. It consists of two 100,000-gal primary grade water storage tanks, two supply pumps, two heating loops with electric heaters and circulating pumps, a deaerator with supply and effluent pumps, and associated piping and valves. Makeup water is received from the demineralized water system and the recovered water from the boron recovery system (Section 9.3.5) and the liquid waste system (Section 11.2.7). The use of two tanks ensures that the entire supply of primary grade water does not become contaminated. Water from the boron recovery system or radioactive liquid waste system is received in one tank. In the event of contamination, water in that tank is processed.

The primary grade water system lines penetrating the containment are isolated on a containment isolation phase A signal. These lines and associated valves are designed to seismic Category I and Safety Class 2 requirements and are inside tornado-, missile-, and flood-protected buildings. Thus, the requirements of GDC 2 and the guidelines of RG 1.29 are satisfied.

No portion of this system is shared with other Millstone units. Thus, the requirements of GDC 5 are not applicable.

Dissolved oxygen content in the water is maintained within limits by the deaerator. In the event of high boron concentration in one tank, all the water in the tank is reprocessed by the boron recovery system, and the system makeup is supplied by the other tank.

The primary grade water storage tanks are located in the yard. Freeze protection is provided by a heater and recirculation loop. Protection from flooding for safety-related systems is discussed in Sections 3.4.1 and 9.3.3 of this SER.

The system was evaluated and found to have no function necessary for achieving safe reactor shutdown conditions or for accident prevention or mitigation. The safety-related systems that receive water from the primary grade water storage tank have either a safety class primary water supply or have sufficient storage capacity to perform their safety functions without additional makeup. Thus, the requirements of GDC 44, 45, and 46 are not applicable.

On the basis of this review, the staff concludes that the primary grade water system is acceptable and meets the acceptance criteria of SRP Section 9.2.6.

9.3 Process Auxiliaries

9.3.1 Compressed Air Systems

The compressed air system was reviewed in accordance with SRP Section 9.3.1 (NUREG-0800).

The compressed air system contains four compressed air subsystems. These are a service air system, the instrument air system, and the backup cold shutdown instrument air system, which are outside containment, and the containment instrument air system. Compressed air systems are not safety related except for the portion of the compressed air system that penetrates the containment between the containment isolation valves.

The instrument and service air systems are designed to provide compressed air of suitable quality and pressure to all instrumentation and controls and to pneumatically operated tools. The service air system consists of one oil-free compressor (capacity 750 scfm) located in the turbine building. The main instrument air system outside the containment consists of two redundant oil-free compressors (capacity 750 scfm) with two redundant air dryers and air filters also located in the turbine building. The service air compressor can provide compressed air to the instrument air header, upstream of the air dryers and the air filters, through a normally closed valve.

The backup cold shutdown instrument air system consists of two redundant oil-free compressors (capacity 105 scfm) with two redundant air dryers and air filters, and is located in the auxiliary building. These compressors are powered from Class 1E buses and operate on loss of offsite power. The system is connected to the instrument air header, which supplies compressed air to all air-operated valves required for cold shutdown that are outside the containment, and to the instrument air line, which penetrates the containment before the containment isolation valves.

The containment instrument air system contains redundant oil-free air compressors (capacity 102 scfm), with redundant air dryers, and air filters located inside the containment. It provides compressed air to all reactor containment instrumentation. The compressed air is automatically supplied from the air system outside the containment on a low-pressure signal and from the cold shutdown air compressors during loss of offsite power. During normal operation, there is no transfer of compressed air into or out of containment. All instrument and control equipment fails in the safe mode upon loss of air (see SER Sections 7.3, 7.4, and 7.6 for details).

The instrument and service air systems are nonsafety related except for the containment isolation valves and the piping in between. The isolation valves and associated piping are designed to seismic Category I, Quality Group B requirements and are located in the tornado-, missile-, and flood-protected building. Thus, the requirements of GDC 2 and the guidelines of RG 1.29 related to seismic classification are met.

Instrument and service air systems are not shared with other Millstone units. Thus, the requirements of GDC 5 are not applicable.

The redundant air dryers and air filters maintain the instrument air dewpoint at 40°F and remove all particles larger than 1 micron. Oil-free compressors minimize the oil content in the air. Inline moisture and pressure differential indicators and annunciators are provided to allow operator action to correct any air quality deviation. All safety-related controlled components can be tested to verify that upon loss of air they will respond by assuming their designated fail-safe positions. Thus, the instrument and service air system meets GDC 1 with respect to quality standards by meeting ANSI MC II-1-1976 as it relates to minimum instrument air quality standards, and meeting RG 1.68.3 as it relates to testing of instrument air systems.

On the basis of this review, the staff concludes that the compressed air systems meet the requirements of GDC 1 regarding instrument air quality and GDC 2 regarding protection from missiles, floods, and natural phenomena and the guidelines of RG 1.68.3 concerning testing of instrument air systems. GDC 5 regarding shared systems and components is not applicable. Therefore, the compressed air system meets the acceptance criteria of SRP Section 9.3.1.

9.3.2 Process and Postaccident Sampling Systems

9.3.2.1 Process Sampling System

The process sampling system is designed to provide representative liquid and gaseous samples drawn from the primary and secondary coolant systems, the associated auxiliary system process streams, and the spent fuel pool cleanup system. Provisions are made to ensure that representative samples are obtained from well-mixed streams or volumes of effluent by the selection of proper sampling equipment and location of sampling points as well as proper sampling procedures. In the event of an accident, all sample lines that pass through the containment will be automatically isolated by fail-closed valves on either side of the containment.

The staff's review of the FSAR included all amendments up to Amendment 7. The information provided by the applicant has been reviewed in accordance with SRP Section 9.3.2 (NUREG-0800).

The process sampling system includes piping and other components associated with the system from the point of sample withdrawal from a fluid system up to the analyzing station, sampling station, or local sampling point. The staff's review included the provisions proposed to sample all principal fluid process streams associated with plant operation and the applicant's proposed design of these systems, including the location of sampling points, as shown on piping and instrumentation diagrams.

The staff determined that the proposed process sampling system meets (1) the requirements of GDC 13 to monitor variables that can affect the fission process for normal operation, anticipated operational occurrences, and accident conditions, by sampling the reactor coolant, the emergency core cooling system core flooding tank, the refueling water storage tank, the boric acid mix tank, and the boron injection tank for boron concentration; (2) the requirements of GDC 14 to ensure a low probability of abnormal leakage, rapidly propagating failure, and gross rupture by sampling the reactor coolant and the secondary coolant for chemical impurities that can affect the reactor coolant pressure boundary material integrity; (3) the requirements of GDC 26 to control the rate of reactivity changes by sampling the reactor coolant, the refueling water storage tank, and the boric acid mix tank for boron concentration; and (4) the requirements of GDC 63 and 64 to monitor for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents by sampling the reactor coolant, the pressurizer tank, the steam generator blowdown, the secondary coolant condensate treatment waste, the sump inside containment, the containment atmosphere, the spent fuel pool, and the gaseous radwaste storage tank for radioactivity.

The staff further determined that the proposed process sampling system meets (1) the standards of ANSI N13.1-1969 for obtaining airborne radioactive samples; (2) the requirements of 10 CFR 20.1(c) and Positions 2.d(2), 2.f(3), 2.f(8), and 2.i(6) of RG 8.8, Revision 3, to maintain radiation exposures to as low as is reasonably achievable by providing (a) ventilation systems and the gaseous radwaste treatment system to contain airborne radioactive materials, (b) the liquid radwaste treatment system to contain radioactive material in fluids, (c) the spent fuel pool cleanup system to remove radioactive contaminants in the spent fuel pool water, and (d) remotely operated containment isolation valves to limit reactor coolant loss in the event of rupture of a sampling line; (3) the requirements of GDC 60 to control the release of radioactive materials to the environment by providing isolation valves that will fail in the closed position; and (4) Positions C.1, C.2, and C.3 of RG 1.26, Revision 3, and Positions C.1, C.2, C.3, and C.4 of RG 1.29, Revision 3, by designing the sampling lines and components of the process sampling system to conform to the classification of the system to which each sampling line and component is connected. The system thus meets the quality standards requirements of GDC 1 and the seismic requirements of GDC 2.

On the basis of this evaluation, the staff concludes that the proposed process sampling system meets the relevant requirements of 10 CFR 20.1(c) and GDC 1, 2, 13, 14, 26, 60, 63, and 64 and the apropriate sections in RGs 8.8, 1.26, and 1.29 and, therefore, is acceptable.

9.3.2.2 Postaccident Sampling System (NUREG-0737, II.B.3)

Subsequent to the TMI-2 incident, the need was recognized for an improved postaccident sampling system (PASS) to determine the extent of core degradation following a severe reactor accident. Criteria for an acceptable sampling and analysis system are specified in NUREG-0737, Item II.B.3. The system should have the capability to obtain and quantitatively analyze reactor coolant and containment atmosphere samples without radiation exposure to any individual exceeding 5 rems to the whole body or 75 rems to the extremities (GDC 19) during and following an accident in which there is core degradation. Materials to be analyzed and quantified include certain radionuclides that are indicators of severity of core damage (e.g., noble gases, isotopes of iodine and cesium, and nonvolatile isotopes), hydrogen in the containment atmosphere, and total dissolved gases or hydrogen, boron, and chloride in reactor coolant samples.

To comply with NUREG-0737, Item II.B.3, the applicant should (1) review and modify his sampling, chemical analysis, and radionuclide determination capabilities as necessary and (2) provide the staff with information pertaining to system design, analytical capabilities, and procedures in sufficient detail to demonstrate that the criteria have been met.

The staff's review included Amendment 7 and letters from the applicant dated April 9 and May 10, 1984.

The PASS is capable of obtaining and analyzing reactor coolant samples and containment atmosphere samples within 3 hours from the time a decision is made to take a sample. An alternate power source will be provided during a loss of offsite power to meet the 3-hour sampling and analysis time limit. These provisions meet Criterion (1) of NUREG-0737, Item II.B.3, and are, therefore, acceptable.

The staff finds that the applicant meets Criterion (2) of NUREG-0737, Item II.B.3, by establishing an onsite radiological and chemical analysis capability. The applicant has committed to provide a plant-specific procedure that will include other physical parameters in addition to fission product activities to provide a realistic estimate of core damage by September 1, 1984. Confirmation should be provided, before September 1, 1984, that a plant-specific procedure to estimate the extent of core damage is in place. On the basis of this commitment, the staff finds that these provisions meet Criterion (2) and are, therefore, acceptable.

Reactor coolant and containment atmosphere sampling during post-accident conditions does not require an isolated auxiliary system to be placed in operation so that the sampling function can be performed. The PASS can obtain samples from the reactor coolant hot and cold legs, the containment recirculation, auxiliary building, and containment sumps, and the containment atmosphere, without using an isolated auxiliary system. The PASS valves, which are not accessible after an accident, are environmentally qualified for the conditions in which they need to operate. The staff finds that these provisions meet Criterion (3) of NUREG-0737, Item II.B.3, and are, therefore, acceptable.

Pressurized reactor coolant samples are cooled and degassed to obtain representative dissolved gas samples. If chlorides exceed 0.15 ppm, verification that dissolved oxygen is less than 0.1 ppm is possible. Dissolved oxygen will be monitored directly using an Orbisphere. These provisions meet Criterion (4) of NUREG-0737, Item II.B.3, and are, therefore, acceptable.

The chloride analysis is performed on the reactor coolant by a grab sample within the 96-hour time limit specified in Criterion (5) of NUREG-0737, Item II.B.3, with a measurement range of 0.1 to 20 ppm by polarographic analysis. The staff determined that these provisions meet Criterion (5) and are, therefore, acceptable.

The applicant has performed a shielding analysis to ensure that exposure to operators while they are obtaining and analyzing a PASS sample is within the acceptable limits. This exposure can be incurred by the operators when they enter and leave the sample panel area, operate sample panel manual valves, position the grab sample into the shielded transfer carts, and perform manual sampling dilutions, if required, for isotopic analysis. PASS personnel radiation exposures incurred from reactor coolant and containment atmosphere sampling and analysis are within 5 rems for the whole body and 75 rems for the extremities, which meet the requirements of GDC 19 and Criterion (6) of NUREG-0737, Item II.B.3, and are, therefore, acceptable.

The PASS has the capability to analyze coolant boron concentrations in the range of 1 to 3,000 ppm. At concentrations below 1,000 ppm the tolerance is ±50 ppm, and above 1,000 ppm the tolerance is ±5%. The staff finds that this provision meets Criterion (7) of NUREG-0737, Item II.B.3, and is, therefore, acceptable.

The PASS provides in-line analysis as well as backup grab samples. The grab samples can either be diluted or undiluted. Backup chemical and radiochemical analyses will be performed in the postaccident sampling facility on site in the chemistry laboratory for Millstone Units 1 and 2. A licensed shipping container therefore is not needed. These provisions meet Criterion (8) of NUREG-0737, Item II.B.3, and are, therefore, acceptable.

The radionuclides in both the primary coolant and the containment atmosphere will be identified and quantified. Provisions are available for diluted reactor coolant samples to minimize personnel exposure. The PASS can perform radioisotope analyses at the levels corresponding to the source term in RGs 1.3 and 1.7. Radiation background levels will be restricted by shielding between the counting room and sampling equipment; ventilation in the counting room is such that analytical results can be obtained within an acceptably small error (approximately a factor of 2). The staff finds that these provisions meet Criterion (9) of NUREG-0737, Item II.B.3, and are, therefore, acceptable.

The accuracy, range, and sensitivity of the PASS instruments and analytical procedures are consistent with the recommendations of RG 1.97, Revision 2, and the clarifications of NUREG-0737, Item II.B.3, transmitted to the applicant on September 2, 1982. Therefore, they are adequate for describing the radiological and chemical status of the reactor coolant. The analytical methods and instrumentation were selected for their ability to operate in the postaccident sampling environment. The standard test matrix and radiation effect evaluation indicated no interference in the PASS analyses. Training for PASS operators will be conducted semiannually, and PASS system testing will be conducted, as a minimum, annually. The staff finds that these provisions meet Criterion (10) and are, therefore, acceptable.

The applicant has addressed provisions for (1) purging to ensure samples are representative, (2) size of sample line to limit reactor coolant loss from a rupture of the sample line, and (3) ventilation exhaust from the PASS.

The containment atmosphere sample takes suction from the hydrogen recombiner supply lines, which are heat traced to minimize iodine plateout. These provisions meet Criterion (11) and are, therefore, acceptable.

On the basis of its evaluation, the staff concludes that the postaccident sampling system meets all 11 criteria of Item II.B.3 in NUREG-0737, and is therefore, acceptable, pending confirmation that a plant-specific procedure to estimate the extent of core damage will be in place before September 1, 1984. Before 5% power operation is exceeded, the applicant shall have installed and have operational the postaccident sampling system.

9.3.3 Reactor Plant Vent and Drain Systems (Equipment and Floor Drainage System)

The reactor plant vent and drain systems were reviewed in accordance with SRP Section 9.3.3 (NUREG-0800).

The nonsafety-related, Quality Group D, nonseismic Category I reactor plant vent and drain systems include all piping from equipment or floor drains to the sump, sump pumps, and piping necessary to carry potentially radioactive and nonradioactive effluent through separate subsystems. Potentially radioactive drainage is collected in floor and equipment drain sumps in each building and discharged to the gaseous or liquid waste system for treatment and/or disposal. Drainage from the turbine building is monitored for radioactivity and is pumped to the yard drainage system or to the liquid waste system on a predetermined radioactivity level. Drainage from nonradioactive sources is discharged to the yard storm sewer system. Therefore, the requirements of GDC 60 are satisfied.

Containment penetrations for the equipment and floor drainage system are designed to seismic Category I and Quality Group B requirements and are located in seismic Category I, flood- and tornado-protected structures. Therefore, the requirements of GDC 2 and RG 1.29 (Positions C.1 and C.5) are satisfied.

The safety-related equipment is protected from flooding damage by physical location within the buildings, or by Class 1E level instruments, which provide warning protection. The areas of the plant where nonseismically designed piping and safety-related equipment are located together are in the auxiliary building, engineered safety features building, control building, fuel building, and the emergency generator enclosure.

The auxiliary building is divided into two areas, the motor control center (MCC)/rod drive area and the remainder of the building. The MCC/rod drive area does not contain piping that could be considered as a source of flooding. All safety-related equipment in the other area is elevated above the floor on concrete pads to prevent any damage by flooding from a pipe rupture. No significant accumulation of water would occur on floors above the lowest elevation because of the stairwells, gratings, pipe sleeves, and duct penetrations. The auxiliary building pipe tunnel sump, located in the basement of the building, is provided with safety-related level instrumentation to alert the control room operator of a flooding condition within the area.

In the engineered safety features building safety-related equipment is located in separate cubicled areas. These cubicles are designed to prevent water intrusion from sources both internal and external to the building and have watertight walls to el 21 ft 6 in., which protect the redundant trains of safety-related equipment from a single passive failure of piping. The sumps all contain safety-related level instrumentation, which provides the control room operator with indication of flooding in the area. The applicant performed a flooding analysis assuming operator action within 30 min after the postulated piping failure and verified that safety-related alarms would provide sufficient warning for the operator to take appropriate action.

In the emergency generator enclosure, safety-related equipment is in cubicles to prevent any simultaneous flooding. The diesel generator systems and auxiliary equipment are designed to start and maintain full-load output with the fire protection sprinkler system operating. There would be no significant accumulation of water within the building to affect safety-related equipment because the building is located on grade elevation and water could drain to the outside through the doors and floor drains.

In the fuel building, safety-related equipment is in cubicles to prevent a nonseismic pipe rupture in the adjacent areas from flooding this equipment. There would be no significant accumulation of water in the area of the safety-related equipment because stairs, grating, and pipe sleeves would drain any water into the basement pipe tunnel. From there, the water would drain into the auxiliary builing pipe tunnel sump where there is safety-related level instrumentation to alert the control room operator to a flooding condition in the area.

In the control building main steam valve building and hydrogen recombiner building, there is no nonseismic piping that could cause flooding damage to any safety-related equipment.

In the basement of the service building, there are redundant safety-related cable tunnels (FSAR Figure 3.8-65). These tunnels are separated from any source of water by a concrete wall that will have all penetrations through it sealed to prevent any water leakage. The remainder of the service building does not contain any safety-related equipment.

The circulating and service water pumphouse is divided into two areas: the safety-related service water pump cubicles and the remainder of the building. These areas are separated from each other by a floodproof concrete wall that will have all penetrations through it sealed to prevent any water leakage. The safety-related service water cubicles contain no non-ASME Code piping capable of flooding the cubicle, and the remainder of the building contains no safety-related equipment. A separate floor drainage system in each service water pump cubicle directs any leakage to a sump that discharges at el 36.5 ft (MSL) outside the cubicle into the circulating water pumphouse.

The remainder of the plant, the turbine building, waste disposal building, auxiliary boiler room, and condensate polishing area do not contain any safety-related equipment in areas of potential flooding. Also, a pipe rupture and flooding of any of these buildings will not affect any safety-related equipment in other plant areas. The flooding resulting from nonseismic Category I tanks is discussed in SER Section 3.4.1 and the circulating water expansion joint in SER Section 10.4.5.

On the basis of its review, the staff concludes that the reactor vent and drain systems meet the requirements of GDC 2 and the acceptance criteria of SRP Section 9.3.3.

9.3.4 Chemical and Volume Control System

The chemical and volume control system (CVCS) is designed to control and maintain reactor coolant inventory and to control the boron concentration in the reactor coolant through the process of charging (makeup) and letdown (drawing off). The system is also designed to provide seal water injection flow to the reactor coolant pumps and to control the reactor coolant water chemistry conditions and activity level by ion exchange, soluble chemical neutron absorber concentration, and makeup water. An essential portion of the system consists of the three charging pumps. These pumps are used during normal operation. The centrifugal charging pumps are also used for high pressure safety injection when the emergency core cooling system (ECCS) is required to function. (ECCS is evaluated in Section 6.3.2.2).

The volume control tank serves as a volume surge for the reactor coolant letdown system to provide for control of hydrogen concentration in the reactor coolant and provide a reservoir of makeup for the charging pumps. The boric acid makeup system provides for boron additions to compensate for reactivity changes and provide shutdown margin for maintenance and refueling operations or emergencies. The charging and letdown portions of the system are designed to seismic Category I requirements and contain redundant active components and an alternate flow path to meet the single-failure criterion.

The staff's review included Amendment 7 of the FSAR. The CVCS description and piping and instrumentation diagrams have been reviewed in accordance with SRP Section 9.3.4 (NUREG-0800). This system (including the boron recovery system) includes components and piping associated with the system from the letdown line of the primary system to the charging lines that provide makeup to the primary system and the reactor coolant pump seal water system.

The basis for acceptance in the staff's review has been conformance of the applicant's design of the CVCS with the following regulations and RGs: (1) the requirements of GDC 1 and the guidelines of RG 1.26 by assigning quality group classifications to system components in accordance with the importance of the safety function to be performed; (2) the requirements of GDC 2 and the guidelines of RG 1.29 by designing safety-related portions of the system to seismic Category 1 requirements; (3) the requirements of GDC 14 by maintaining reactor coolant purity and material compatibility to reduce corrosion and thus reduce the probability of abnormal leakage, rapid propagating failure, or gross rupture of the reactor coolant pressure boundary; (4) the requirements of GDC 29 as it relates to the reliability of the CVCS to provide negative reactivity to the reactor by supplying borated water to the reactor coolant system in the event of anticipated operational occurrences; (5) the requirements of GDC 33 and 35 by designing the CVCS with the capability to supply reactor coolant makeup in the event of small breaks or leaks in the reactor coolant pressure boundary and to function as part of the ECCS assuming a single failure coincident with loss of offsite power; and (6) the requirements of GDC 60 and 61 with respect to confining radioactivity by venting and collecting drainage from the CVCS components through closed systems.

On the basis of the review of the CVCS and the requirements for system performance of necessary functions during normal, abnormal, and accident conditions, the staff concludes that the design of the CVCS and supporting systems meets the requirements of GDC 1, 2, 14, 29, 33, 60, and 61 and is, therefore, acceptable.

9.4 Heating, Ventilation, and Air Conditioning Systems

9.4.1 Control Building Ventilation System

The control building ventilation system was reviewed in accordance with SRP Section 9.4.1 (NUREG-0800).

The control building ventilation system includes the control room air conditioning system, the instrument rack and computer room air conditioning system, the control room emergency ventilation and pressurization system, the switchgear area air conditioning system, the battery room area ventilation system, the chiller equipment space ventilation system, control room toilet and kitchenette exhaust system, and the control building purge ventilation system. The switchgear area air conditioning system provides ventilation to the cable spreading area. The control room air conditioning system also includes a safety-related chilled water system, which provides chilled water as the cooling medium to the control area air conditioning units. The air conditioning and ventilation systems and the chilled water system consist of two 100% capacity redundant trains. The control room is normally maintained at a slightly positive pressure relative to the outdoors by taking makeup air through a tornado-, missile-, and flood-water-protected air intake louver.

The control room area ventilation system is designed to maintain a suitable environment for equipment operation and safe occupancy of the control room under all plant operating conditions. The control room area ventilation system serves the control room, the instrument rack and computer room, the switchgear rooms, battery rooms, chiller equipment room, cable spreading area, and control room toilet and kitchenette (see Section 6.4 of this SER for further discussion of control room habitability).

The control room air conditioning and ventilation pressure boundary is maintained by redundant isolation valves or dampers on all inlet and exhaust openings. Redundant radiation monitors and chlorine gas detectors located on the control room air intake automatically isolate the control room building on high radiation or chlorine content. The system is isolated on control building isolation (CBI) signal. The control room emergency ventilation and pressurization system provides compressed air to the control room for 1 hour after an accident (60 sec after the isolation signal); the redundant emergency filtration train, which contains a fan, prefilter, carbon absorber, and high-efficiency particulate air (HEPA) filter, provides breathable air in the control room, instrument racks, and computer room. Once the compressed air is depleted, the emergency filtration subsystem is started. A positive pressure is maintained in the control room by admission of some outside air through the filter. The emergency control room ventilation system is designed to meet RG 1.52 (Position C.2). Further discussion of compliance of this position is contained in Section 6.5.1.

The control room air intake is also provided with a smoke detector alarm system. The control building purge ventilation system removes smoke or carbon dioxide from the instrument rack and computer room, the cable spreading area, switchgear rooms, and the mechanical equipment room. The control room is purged by the purge ventilation in the adjacent areas. The cable spreading area and switchgear rooms are protected by a carbon dioxide fire extinguishing system. The applicant has provided annunciators in the control room to provide information regarding the capability of the battery room exhaust fans to prevent accumulation of hydrogen.

The control building air conditioning, ventilation, and chilled water systems are located in the seismic Category I, missile-, flood-, and tornado-protected structure. Essential portions of the system itself are seismic Category I, Quality Group C, and are physically separated from the high-energy system. The redundant chlorine detectors are not designed to seismic Category I criteria, nor are they electrical Class 1E. These detectors are designed to fail-safe criteria so that in the event of a detector failure, the control room envelope is automatically isolated. This fail-safe design would not subject the control room environment to the toxic gases and as such is acceptable. The control room pressurization system, which is used during the first hour after an accident, was not initially designed to seismic Category I requirements. Because this was not acceptable to the staff, the applicant has committed to redesign the system to seismic Category I requirements. The air bottles are designed to ASME Code, Section VIII criteria, and the piping and valves are designed to ANSI B-31.1 criteria. Thus, the staff concludes that the control room ventilation system meets the requirements of GDC 2 and the guidelines of RG 1.29 (Positions C.1 and C.2).

No portion of this system is shared with other Millstone units. Thus, the requirements of GDC 5 are not applicable.

All seismic Category I electrically powered motors and controls associated with the control building air conditioning and ventilation system and the chilled water system are redundant and are powered from separate Class 1E power systems to ensure operability of at least one train of the control room ventilation system in the event of any single active failure. The control room ventilation components are accessible to permit inservice inspection and testing as required.

The above design meets the requirements of GDC 4 and 19 and the guidelines of RGs 1.78 (Positions C.3, C.7, and C.14) and 1.95 (Positions 4a and 4d) with respect to the uninterrupted safe occupancy of the control room and associated required manned areas under all normal and accident conditions including LOCA conditions.

Because the control room is not a source of radioactivity and the emergency filtration system only functions following an accident, the requirements of GDC 60 and the guidelines of RG 1.140 are not applicable.

On the basis of this review, the staff concludes that the control room area ventilation system is in conformance with the requirements of GDC 2, 4, 5, and 19 related to protection from natural phenomena, maintaining proper environmental limits for equipment operation, shared systems, and protection to permit access for occupancy of the control room under normal and accident conditions, and the guidelines of RGs 1.29 (Positions C.1 and C.2), 1.52 (Position C.2), 1.78 (Positions C.3, C.7, and C.14), and 1.95 (Positions C.4a and C.4d) related to seismic design qualifications, system testing and maintenance, protection against hazardous chemical release, protection of personnel against chlorine gas release, and design for normal operation. The system is therefore acceptable. The control building ventilation system meets the acceptance criteria of SRP Section 9.4.1.

9.4.2 Fuel Building Ventilation System (Spent Fuel Pool Area Ventilation System)

The fuel building ventilation system was reviewed in accordance with SRP Section 9.4.2 (NUREG-0800).

The fuel building ventilation system, which serves the entire fuel building, is designed to maintain a suitable environment for equipment operation and to limit potential radioactive release to the atmosphere during normal operation and postulated fuel-handling-accident conditions. The system is not required for safe shutdown of the plant in the event of a LOCA but is only required to mitigate the consequences of a fuel-handling accident.

The fuel building ventilation system consists of a nonsafety-related air supply system and one safety-related and one nonsafety-related exhaust system. The supply air system consists of three 50% capacity heating and ventilating units shared between the fuel building and the waste disposal building ventilation system (see SER Section 9.4.3 for details). Each unit consists of a prefilter, a hot water preheat coil, a hot water reheat coil, and a fan that draws air through an air-operated damper from outside. Two safety-related wall-mounted backdraft dampers provide makeup air in the event of loss of the nonsafety-related supply air system or isolation following a failure of one of the two redundant special filter assemblies. The safety-related exhaust system consists of two redundant 100% capacity special filter assemblies, consisting of prefilters, absolute filters, and carbon filters, and associated fans and dampers. The nonsafety-related exhaust system consists of one 100% capacity exhaust fan with inlet and outlet dampers. The exhaust air is discharged through a radiation-monitored ventilation vent.

All essential parts of the fuel building ventilation system are seismic Category I, Quality Group C, thereby satisfying the guidelines of Position C.1 of RG 1.29. The system is located in the fuel building, which is seismic Category I, flood and tornado protected. This satisfies the requirements of GDC 2. There are no high- or moderate-energy systems located near the fuel building ventilation system. Adequate protection against internally generated missiles and the effects of pipe whip and fluid jets is provided by separated equipment locations (see Sections 3.5.1.1 and 3.6.1 of this SER).

Millstone Unit 3 has its own fuel building and fuel building ventilation system. There is no sharing of ventilation system functions with the other Millstone units. Therefore, the requirements of GDC 5 are not applicable.

The exhaust subsystem of the fuel ventilation system is an engineered safety feature (see Section 6.5). Each of the two redundant sets of exhaust filter train fans and motor-operated dampers is served from separate trains of the emergency Class 1E standby power and thus meets the single-failure criterion. Thus, the fuel building ventilation system meets the requirements of GDC 60 and the guidelines of RGs 1.52 (Position C.2) and 1.140 (Positions C.1 and C.2) for system design, testing, and maintenance.

The flow of air within the fuel building is directed from areas of low potential for airborne contamination to areas of greater potential for airborne contamination. The nonsafety exhaust system operates during normal operation, while the safety-related exhaust system operates during the fuel-handling operation and accident conditions (containment isolation signal) or high exhaust contamination. Outleakage from the fuel building is prevented by maintaining a negative pressure relative to the outside atmosphere.

Thus, the system meets the requirements of GDC 61 and the guidelines of RG 1.13 (Position C.4) regarding controlled leakage (see Section 6.5).

On the basis of this review, the staff concludes that the fuel building ventilation system is in conformance with the requirements of GDC 2, 60, and 61 as they relate to protection against natural phenomena, control of releases of radioactive materials, and radioactivity control and the guidelines of RGs 1.13 (Position C.4), 1.29 (Positions C.1 and C.2), 1.52 (Position C.2), and 1.140 (Positions C.1 and C.2) as they relate to protection against radioactive releases, seismic classification, and system design for emergency and normal operation. The system is, therefore, acceptable. The spent fuel building ventilation system meets the acceptance criteria of SRP Section 9.4.2.

9.4.3 Auxiliary and Waste Disposal (Radwaste) Area Ventilation System

The auxiliary and waste disposal area ventilation system was reviewed in accordance with SRP Section 9.4.3 (NUREG-0800).

The auxiliary and waste disposal area ventilation system serves the waste disposal building and the auxiliary building with separate systems for each area.

These systems are designed to maintain a suitable environment for equipment operation and personnel access and to limit potential radioactive releases to the environment during all modes of operation.

The auxiliary building ventilation system (ABVS) consists of both safety- and nonsafety-related subsystems and serves all areas of the auxiliary building, including all engineered safety features within the building.

The nonsafety-related, nonseismic Category I, Quality Group D, general ventilation supply and unfiltered exhaust subsystems normally operate in conjunction with the safety-related filtered exhaust subsystems.

The safety-related charging pump, component cooling water pump, and heat exchanger ventilation system; the motor control center (MCC), rod control, and cable vault ventilation system; the auxiliary building filtration units including fans and dampers; and the auxiliary building isolation dampers are all seismic Category I and Quality Group C.

The auxiliary building ventilation system is actuated manually. The charging pump, component cooling pump, and heat exchanger area ventilation supply and exhaust fan and dampers are actuated by the operation of these pumps. The auxiliary buildin, ventilation isolation dampers are actuated by a safety injection signal (S S).

The general ventilation air supply portion of the system includes two 50% capacity air-handling units. Each air-handling unit includes a prefilter, a preheat coil, a fan, and a heating coil. The exhaust fans maintain the building at a negative pressure. General ventilation air is supplied to both clean and potentially contaminated areas of the auxiliary building. Control of airborne activity is accomplished by exhausting air supplied to clean areas through the potentially contaminated areas. This air in turn is processed by the filtered exhaust subsystem. The remaining air supplied to clean areas is exhausted by the unfiltered exhaust subsystem. All air exhausted from the auxiliary building by the filtered exhaust subsystem and the unfiltered exhaust subsystem is directed to the unit vent where it is monitored by the unit vent radiation monitor before it is released to the atmosphere. Radioactivity in this case is monitored upstream of the filtration units. On high radiation alarm, exhaust air is diverted through one of the two filter units. In the event of an SIS or loss of power (LOP), all auxiliary building ventilation system components automatically shut down. The filtered exhaust subsystem is then automatically operated. All areas of the auxiliary building with the exception of the ECCS pump rooms and the MCC, rod control, and cable vault ventilation systems are automatically isolated from the filtered exhaust subsystem.

The charging pump, component cooling water pump, and heat exchanger area redundant supply and exhaust fans continue to operate, venting through the turbine building vent after filtration.

The MCC, rod control, and cable vault ventilation subsystem includes two redundant supply units consisting of a prefilter, service water cooling coil, chilled water cooling coil, and fan. Each unit continuously recirculates conditioned air through the electrical space to maintain design temperature.

The auxiliary building ventilation system is located in the auxiliary building, which is a seismic Category I, flood- and tornado-protected structure (see Sections 3.4.1 and 3.5.2). The system is arranged so that both essential and nonessential equipment and areas are cooled normally by nonsafety-related equipment with an entirely separate safety-related subsystem. The filtered exhaust subsystem is brought into service under emergency conditions. The failure of any nonsafety-related equipment will not affect the essential functions of safety-related equipment. Essential (safety-related) portions of the system itself are seismic Category I, Quality Group C, and are physically separated from high-energy systems. The outside air intakes are tornado missile protected. Thus, the requirements of GDC 2 and the guidelines of RG 1.29 (Position C.1) for safety-related and Position C.2 for nonsafety-related portions, are met.

The seismic Category I, Quality Group C, auxiliary building filtered exhaust subsystem consists of two redundant, 100% capacity trains. The exhaust filters consist of a preheater/demister section and carbon and absolute filters section (see Section 6.5). The filtered exhaust subsystem performs both a safety and nonsafety-related function. The two preheater/demister sections, filter trains, centrifugal fans, and associated isolation and inlet vane dampers are connected to separate trains of the Class 1E emergency standby power. Thus, the requirements of GDC 60 and the guidelines of RGs 1.52 (Position C.2) and 1.140 (Position C.1) are satisfied.

The waste disposal area ventilation system is classified as nonsafety related, nonseismic Category I, Quality Group D. It is a once-through system using outside air as the ventilating and cooling medium and consisting of separate supply and exhaust subsystems. The supply air is provided by the three nonsafety-related 50% capacity heating and ventilating units described for the fuel building ventilating system in SER Section 9.4.2.

The exhaust subsystem consists of two axial fans with one operating and the other on standby. The exhaust duct is arranged in such a manner that cell exhaust air flow is from areas of low potential airborne contamination to areas of higher potential airborne contamination. The exhaust air is monitored for radiation by the radiation monitor located in the plant vent stack and locally in the exhaust duct from the waste disposal building. During normal plant operation, the exhaust air is directed to the ventilation vent stack located on the turbine building. Upon detecting high radiation, the exhaust air is diverted to the auxiliary building filter system through a set of two air-operated dampers before being discharged to the atmosphere through the ventilation vent stack. The filter inlet dampers from the waste disposal building exhaust close automatically on receipt of an SIS or LOP signal.

The waste disposal area ventilation system is separated from safety-related systems; therefore, its failure will not compromise plant safety. Thus, the requirements of GDC 2 and guidelines of RG 1.29 (Position C.2) are met. This system operates only during normal conditions and performs no safety functions and is not shared with other Millstone units. Thus, the requirements of GDC 5 are met.

On the basis of the above review, the staff concludes that the auxiliary building ventilation system and waste disposal area ventilation system are in conformance with the requirements of GDC 2, 5, and 60, as they relate to protection against natural phenomena, assurance of proper operating environment for essential equipment, shared systems, and control of releases of radioactive materials to the environment, and the guidelines of RGs 1.29 (Positions C.1 and C.2), 1.52 (Position C.2), and 1.140 (Positions C.1 and C.2) as they relate to seismic classification and system design for emergency and normal operation. The auxiliary building ventilation system and waste disposal area ventilation system meet the acceptance criteria of SRP Section 9.4.3.

9.4.4 Turbine Building Ventilation System

The turbine building ventilation system was reviewed in accordance with SRP Section 9.4.4 (NUREG-0800).

The turbine building ventilation system is a nonsafety-related system, which removes the heat dissipated from equipment, piping, and lighting. The supply portion of the system consists of four axial flow fans each with associated ductwork, intake louvers, and dampers. Six transfer fans transfer air from the lower level and the battery room to the upper level of the turbine building.

The exhaust portion of the system consists of 12 axial flow fans located below the turbine building roof with an associated backdraft damper and a weatherproof hood. The storage area, condensate polishing area, lubricating oil storage room, sample sink areas, and elevator machinery room have separate ventilation subsystems.

The turbine building ventilation system is classified as nonsafety related, nonseismic Category I, Quality Group D. The system maintains an acceptable environment for personnel and the nonessential equipment served during normal plant operation and has no safety functions. It is separated from safety-related plant systems and potentially radioactive areas; therefore, failure of the system will not compromise the operation of any essential plant systems or result in an unacceptable release of radioactivity. Therefore, it meets the requirements of GDC 2 and the guidelines of RG 1.29, Position C.2.

On the basis of this review, the staff concludes that the turbine building ventilation system meets the requirements of GDC 2 with respect to the need for protection against natural phenomena because its failure does not affect safety system functions or result in release of radioactive material, and meets the guidelines of RG 1.29 (Position C.2) concerning its seismic classification. The system is, therefore, acceptable. The turbine area ventilation system meets the acceptance criteria of SRP Section 9.4.4.

9.4.5 Engineered Safety Features Ventilation Systems

The engineered safety features (ESF) ventilation systems were reviewed in accordance with SRP Section 9.4.5 (NUREG-0800).

The ESF building ventilation system consists of both safety- and nonsafety-related ventilation systems and is designed to provide a suitable environment for personnel and equipment operation and to prevent or minimize the spread or release of airborne radioactive contamination to the atmosphere.

The normal ventilation system is nonsafety related and consists of three sets of supply and exhaust fans. One set serves the ventilation mechanical rooms, the second serves the following areas:

(1) safety injection pump and quench spray pump areas and residual heat removal pump and heat exchanger areas

(2) containment recirculation pump and cooler areas

(3) refueling water recirculation pump area

(4) motor-driven auxiliary feedwater pump areas

(5) turbine-driven auxiliary feedwater pump area

(6) main steam piping penetration area

The third set of fans ventilates the piping cubicles to provide an air exchange during occupancy.

The ESF building normal ventilation system exhaust is monitored for radiation releases during normal plant operation and will not operate during or after postulated accidents.

The ESF building emergency ventilation system contains the following safety-related ventilation subsystems:

(1) two self-contained air-conditioning units for the residual heat exchanger area, residual heat removal pump area, safety injection and quench spray pump area

(2) two self-contained air-conditioning units for the containment recirculation pump and cooler areas

(3) one supply and exhaust fans system for the mechanical room and auxiliary feedwater pump areas

All of the safety-related ESF building ventilation subsystems are located in a seismic Category I structure that is tornado, missile, and flood protected.

These subsystems are classified seismic Category I and are supplied with Class 1E electric power. Thus, they meet the requirements of GDC 2 and the guidelines of RG 1.29 (Positions C.1 and C.2) for seismic design classifications. The self-contained air conditioning units start automatically whenever any of the safety-related pumps within their respective areas start and supply air throughout the equipment area and return the air to the units.

The mechanical room and the auxiliary feedwater pump area ventilation system consists of two redundant trains of 100% capacity supply and exhaust fans. The design of this system permits the use of an outside air supply during the summer and the recirculation of air during the winter. The train A supply and exhaust fans start when either of the motor-driven auxiliary feedwater pumps starts or the steam flows to the turbine-driven auxiliary feedwater pump. The train B supply and exhaust fans start on failure of train A. A single train is capable of maintaining design temperature conditions.

The redundant components are connected to redundant Class 1E buses and can function as required in the event of loss of offsite power. The safety-related ESF building ventilation system can withstand a single active component failure without degrading the performance of the safety function.

Upon receipt of an SIS, the dampers within the normal ventilation system close, isolating the safety injection pump, quench spray pump, RHR pump, and heat exchanger areas. At this time, the supplementary leak collection and release system (see SER Section 6.2.3 for details) starts and maintains a negative pressure within the interior cubicles. The safety-related air conditioning units start and cool their respective areas.

All areas in which safety-related equipment is located are monitored for high temperature. Thus, this system meets the requirements of GDC 4 for maintaining proper environmental conditions in essential areas within the design limits for normal, transient, or accident conditions.

No portion of this system is shared with other Millstone units; thus, the requirements of GDC 5 are not applicable.

Outside air supply for the auxiliary feedwater pump areas is provided through an inlet filter, which prevents dust accumulation. It thus meets the requirements of GDC 17 by meeting the guidelines of NUREG/CR-0660 related to accumulation of dust particles.

On the basis of the above review, the staff concludes that the ESF ventilation system is in conformance with the requirements of GDC 2, 4, 5, and 17 as they relate to protection against natural phenomena, assurance of proper environment for essential equipment, shared systems, and proper functioning by meeting the guidelines of RGs 1.29 (Positions C.1 and C.2) and 1.26 and NUREG/CR-0660 as they relate to accumulation of dust particles. Thus, the ESF ventilation system meets the acceptance criteria of SRP Section 9.4.5.

9.4.6 Emergency Generator Enclosure Ventilation System

The safety-related emergency generator enclosure ventilation system was reviewed in accordance with SRP Section 9.4.5 (NUREG-0800).

The emergency generator enclosure has two safety-related and two nonsafety-related ventilation systems, one for each generator enclosure. The respective safety-related system automatically starts upon start of the emergency generator diesel engine. Each safety-related ventilation system consists of two 50% capacity supply fans and electrohydraulically operated inlet air, recirculating air, and exhaust air dampers.

The supply fan introduces outside air into the enclosure and forces air out of the enclosure through the exhaust dampers and then through the muffler enclosure to the outdoors.

This flow carries away the heat rejected by the emergency generator combustion air exhaust muffler. All equipment and ductwork in this ventilation system are seismically designed and supported. All ventilating systems are provided with tornado dampers.

Each enclosure also has a nonsafety-related ventilation system, which consists of one exhaust fan ductwork and a backdraft damper. This system operates when the emergency generator is not operating. Each of these ventilation systems consists of an exhaust fan that draws air into the enclosure through the safety-related air inlet damper. The air is discharged to the outdoors through a backdraft damper. Although the system is nonsafety related, it is seismically supported to prevent damage to safety-related equipment during a seismic event.

Heat is provided to the emergency diesel generator enclosure by three electric unit heaters. Space temperature is maintained above 50°F when outside temperature is at a minimum design condition.

The diesel generator enclosure ventilation system is located in a seismic Category I structure that is tornado, missile, and flood protected. All ventilating inlets and outlets are provided with concrete missile-protected hoods. All essential system components are designed to seismic Category I and Quality Group C requirements. Therefore, the system meets the requirements of GDC 2 and the guidelines of RG 1.29 (Positions C.1 and C.2).

No portion of this system is shared with other Millstone units. Thus, the requirements of GDC 5 are not applicable.

The applicant has provided protection from dust accumulation (see SER Section 9.5.8 for details); the staff, therefore, concludes that the requirements of GDC 17 regarding protection from unacceptable dust collection in accordance with the guidelines of NUREG/CR-0660 have been met.

Each safety-related emergency generator enclosure ventilation system is powered from a separate emergency electrical power train. In the event of damper control failure, the electrohydraulically operated air inlet and/or exhaust dampers fail in the open position. An enclosure temperature switch alarms in the control room when that enclosure temperature exceeds 120°F or falls below 50°F. Thus, it meets the requirements of GDC 4 with respect to maintaining proper design environmental conditions.

On the basis of this review, the staff concludes that the emergency generator enclosure ventilation system is in conformance with the requirements of GDC 2, 4, and 5 as they relate to protection against natural phenomena and assurance of proper environment for essential equipment and shared systems and meets the guidelines of RG 1.29 (Positions C.1 and C.2) concerning seismic classifications. The staff concludes that it meets the requirements of GDC 17 concerning protection from unacceptable dust collection in accordance with the guidelines of NUREG/CR-0660. The system, therefore, is acceptable and meets the acceptance criteria of SRP Section 9.4.5.

9.4.7 Circulating and Service Water Pumphouse and Other Yard Structures Ventilation Systems

The safety-related circulating and service water pumphouse ventilation system was reviewed in accordance with SRP Section 9.4.5 (NUREG-0800).

The ventilation system for the service water portion of the pumphouse is safety related. The ventilation system for the circulating portion of the pumphouse and other yard structures is nonsafety related and does not affect any safety-related equipment. The service water portion of the pumphouse contains four service water pumps with two pumps in each of two cubicles. Each cubicle contains its own safety-related seismic Category I ventilation system. Each service water pump cubicle ventilation system supplies and exhausts air through air inlets and discharges located on the roof. The air inlet and exhaust ductwork is seismically supported and designed and is provided with sound attenuators to reduce noise emission from the pumphouse. All air-operated dampers fail in the open position. Emergency power is supplied to each service water pump cubicle by a separate independent train. Each ventilation system exhaust fan is operated by means of a temperature control switch, which maintains the service water pump cubicle at the desired temperature.

The two cubicles are separated from each other and from the circulating water pump section of the pumphouse by missile and flood barriers.

The air intake and exhaust hoods for each service water cubicle ventilation system are protected from tornados and floods and are designed to withstand a safe shutdown earthquake.

Heating is provided by an electric heater in each cubicle to maintain space temperature above 40°F. Heating of the service water pump cubicles is not essential. Heaters in the service water pump cubicle are seismically supported to prevent damage to essential portions of the service water pump cubicle from a seismic event.

On the basis of this review, the staff concludes that the service water pumphouse ventilation system meets the requirements of GDC 2 and 4 with respect to providing protection from natural phenomena, floods, and missiles and maintaining proper environmental conditions and meets the guidelines of RG 1.29 (Positions C.1 and C.2). It meets the acceptance criteria of SRP Section 9.4.5.

9.4.8 Main Steam Valve Building Ventilation System

The safety-related main steam valve building ventilation system was reviewed in accordance with SRP Section 9.4.5 (NUREG-0800).

The main steam valve building ventilation system provides a suitable environment for personnel, equipment operations, and controls during normal operation, and for safety-related equipment during loss-of-offsite-power (LOP) transients and upon safety injection signal (SIS) initiation.

The main steam valve building ventilation system consists of four axial flow fans and two intakes with dampers and associated ductwork. Two fans are nonsafety related and are powered from the normal power supply. The other two fans are safety related and are powered from the Class 1E power supply. Each safety-related fan has a discharge backdraft damper arranged in series with an inlet emergency-powered motor-operated damper. Each nonsafety-related fan has a discharge backdraft damper arranged in series with an inlet Class 1E-powered air-operated damper. Each intake assembly consists of two Class 1E-powered motor-operated dampers in series and associated ductwork.

The main steam valve building ventilation system is located in a seismic Category I structure that is tornado, missile, and flood protected. All ventilating inlets and outlets are provided with concrete missile-protected hoods. All essential system components are designed to seismic Category I and Quality Group C requirements. Therefore, the system meets the requirements of GDC 2 and the guidelines of RG 1.29 (Positions C.1 and C.2).

No portion of this system is shared with other Millstone units. Thus, the requirements of GDC 5 are not applicable.

The main steam valve building ventilation system exhaust fans powered by the normal ac power supply are not safety related. The failure of these fans will not interfere with operation of other safety-related systems.

The other two ventilation exhaust fans and their associated components are connected to Class 1E power supplies to permit their continued operation to mitigate the consequences of LOP transients. These components are designed and manufactured according to QA Category I requirements to maintain the integrity of the Class 1E power system.

When an LOP occurs, both normally powered exhaust fans are shut down. To maintain the space temperature below the design limit, the Class 1E-powered fans and the outdoor intake motor-operated dampers remain in the open position. Upon receiving an SIS, all exhaust fans are shut down, their associated Class 1E-powered dampers are closed, and the Class 1E-powered motor-operated dampers in the supplementary leak collection and release system (see SER Section 6.2.3 for details) are opened, thus creating a slight negative pressure within the main steam valve building. In the event of loss of power to one of the Class 1E-powered trains, the motor-operated damper will fail open and the air-operated damper will fail closed.

Thus, this system meets the requirements of GDC 4 for maintaining proper environmental conditions during normal, transient, or accident conditions.

On the basis of this review, the staff concludes that the main steam valve building ventilation system is in conformance with the requirements of GDC 2, 4, and 5 as they relate to protection against natural phenomena and assurance of proper environment and shared systems and meets the guidelines of RG 1.29 (Positions C.1 and C.2) concerning seismic classifications. The main steam valve building ventilation system meets the acceptance criteria of SRP Section 9.4.5.

9.4.9 Waste Disposal Building Ventilation System

The waste disposal building ventilation system is reviewed in Section 9.4.3 of this SER.

9.4.10 Hydrogen Recombiner Building Ventilation System

The hydrogen recombiner building ventilation system was reviewed in accordance with SRP Section 9.4.5 (NUREG-0800).

The hydrogen recombiner building ventilation system provides a suitable environment for personnel and equipment operation and mitigates the potential for a release of airborne radioactive material to the atmosphere. It consists of both safety- and nonsafety-related subsystems. The hydrogen recombiner building postaccident exhaust system and the hydrogen recombiner portion of the ventilation system are safety related. The hydrogen recombiner control room air conditioning unit, the HVAC equipment room ventilation system, and the recombiner building unit heaters are nonsafety related and are controlled by locally mounted temperature control switches.

The safety-related hydrogen recombiner ventilation system provides cooling to the hydrogen recombiners located in separate cubicles and which are required to operate 1 hour after a postulated accident. This ventilation system consists of two redundant supply and exhaust ducts with redundant safety-related fans, radiation monitors, and electrohydraulic isolation dampers. After a postulated accident, when one of two hydrogen recombiners operates, the ventilation system to that cubicle is manually activated from the control room. This ventilation system will shut down on receipt of a high radiation signal from the radiation monitor located in the discharge duct.

The hydrogen recombiner cubicles and the hydrogen recombiner sample room can be maintained at a negative pressure by exhausting air through the hydrogen recombiner postaccident exhaust system ductwork connected to the supplementary leak collection and release system (SRP Section 6.2.3).

The hydrogen recombiner building ventilation system is located in a seismic Category I structure that is tornado, missile, and flood protected. All ventilating inlets and outlets are provided with concrete missile-protected hoods. All essential system components are designed to seismic Category I requirements.

Therefore, the system meets the requirements of GDC 2 and the guidelines of RG 1.29 (Positions C.1 and C.2).

No portion of this system is shared with other Millstone units. Thus, the requirements of GDC 5 are not applicable.

The electrical components (electrohydraulic dampers, fans, and radiation monitors) of the safety-related ventilation system are powered from the Class 1E power supplies to permit their continued operation to mitigate the consequences of LOP transients. These components are designed and manufactured according to QA Category I requirements to maintain the integrity of the Class 1E power system. In the event of loss of power to one of the Class 1E-powered trains, the electrohydraulic dampers will close to prevent the release of radioactivity to the environment. Thus, this system meets the requirements of GDC 4 for maintaining proper environmental conditions during normal, transient, or accident conditions.

On the basis of this review, the staff concludes that the hydrogen recombiner building ventilation system is in conformance with the requirements of GDC 2, 4, and 5 as they relate to protection against natural phenomena and assurance of proper environment and shared systems and meets the guidelines of RG 1.29 (Positions C.1 and C.2) concerning seismic classifications. The hydrogen recombiner building ventilation system meets the acceptance criteria of SRP Section 9.4.5.

9.5 Other Auxiliary Systems

9.5.1 Fire Protection

The staff has reviewed the fire protection program in accordance with SRP Section 9.5-1 (NUREG-0800), which contains, in BTP CMEB 9.5-1, the technical requirements of Appendix A to BTP APCSB 9.5-1 and Appendix R to 10 CFR 50. Because the applicant has compared his program to the latter guidelines, this report also references these guidelines.

As part of its review, the staff will visit the plant site to examine the relationship of safety-related components, systems, and structures in specific plant areas to both combustible materials and to associated fire detection and suppression systems. The site visit has not been conducted because the construction of the plant has not progressed to the level where such a visit would be meaningful.

The staff review included an evaluation of the automatically and manually operated water and gas fire suppression systems, fire detection systems, fire barriers, fire doors and dampers, fire protection administrative controls, and fire brigade size and training. The objective of the staff review is to ensure that in the event of a fire, personnel and plant equipment would be adequate to safely shut down the reactor, maintain the plant in a safe shutdown condition, and minimize the release of radioactive material to the environment.

There are two other operating nuclear power plant units on the same site. Some services are shared between the units.

The staff's consultant, Rolf Jensen & Associates, Inc., participated in the review of the fire protection program and concurs with the staff's findings.

9.5.1.1 Fire Protection Program Requirements

Fire Protection Program

The fire protection program described in the applicant's Fire Protection Evaluation Report establishes policy for the protection of structures, systems, and components important to safety. The program conforms to the technical requirements in BTP CMEB 9.5-1, Section C.1. The staff finds that the fire protection program meets the guidelines of BTP CMEB 9.5-1, Section C.1, and is, therefore, acceptable.

Fire Hazards Analysis

The applicant's fire hazards analysis identified combustible materials present in fire areas, identified safety-related equipment, determined the consequences of a fire on safe shutdown capability, and summarized available fire protection in accordance with BTP CMEB 9.5-1, Section C.1.b.

GDC 3 requires that fire fighting systems be designed to ensure that rupture or inadvertent operation does not significantly impair the safety capability of structures, systems, and components important to safety. The applicant has not indicated that components required for hot shutdown are so designed that rupture or inadvertent operation of fire suppression systems will not adversely affect the operability of these components.

The staff is concerned whether the mechanisms by which fire and fire fighting systems may cause the simultaneous failure of redundant or diverse trains have been adequately considered in the design. The staff will require the applicant to identify the mechanisms that were considered in the fire hazards analysis and the measures taken to preclude the fire or fire-suppressant-induced failure of redundant or diverse safety trains and to document the procedures.

Alternate Shutdown

Alternate shutdown capability has been provided for the control room and cable spreading room and is evaluated in Section 9.5.1.4 of this report.

Implementation of Fire Protection Program

The fire protection program should be operational before initial fuel loading.

9.5.1.2 Administrative Controls

The administrative controls for fire protection consist of the fire protection program and organization, the fire brigade training, the controls over combustible materials and ignition sources, the prefire plans and procedures for fighting fires, and quality assurance. The applicant has committed to implement the fire protection program and administrative controls delineated in Appendix R to 10 CFR 50, Section III.K (BTP CMEB 9.5-1, Section C.2).

On the basis of the applicant's commitments, the staff concludes that administrative controls meet the guidance of BTP CMEB 9.51, Section C.3, and, therefore, are acceptable.

9.5.1.3 Fire Bridgade and Fire Brigade Training

The applicant has committed to provide a fire brigade that will consist of at least five members per shift.

To provide proper coverage during all phases of operation, members of each shift crew will be trained in fire protection in accordance with the staff's guidance including RG 1.101. The applicant has committed to implement the requirements of Appendix R to 10 CFR 50, Section III.H (BTP CMEB 9.5-1, Section C.3) concerning the fire brigade and the fire protection program contained in the staff supplemental guidance, "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance," dated August 29, 1977, including fire brigade training and fire fighting procedures.

On the basis of the applicant's commitments, the staff concludes that the fire brigade and the training for the fire brigade meet the guidelines of BTP CMEB 9.5-1, Section C.4, and, therefore, are acceptable.

9.5.1.4 General Plant Guidelines

Building Design

The walls that separate buildings and walls and floor/ceiling assemblies used to enclose rooms containing safe shutdown systems are 3-hour-fire-rated assemblies. The applicant has stated that all fire-rated assemblies are designed in accordance with fire-barrier designs for 3 hours obtained from the Fire Resistance Directory pubished by Underwriters Laboratory (UL), or they are constructed of 7-in.-thick reinforced concrete in accordance with the National Fire Protection Handbook (National Fire Protection Association, 1981) for a minimum fire resistance rating of 3 hours. On the basis of its review, the staff concludes that the fire-rated walls and floor/ceiling assemblies are provided in accordance with the guidelines of BTP CMEB 9.5-1, Section C.5.a, and are, therefore, acceptable.

Three-hour-fire-rated penetration seals are provided for all penetrations of fire-rated walls of floors/ceilings tested in accordance with BTP CMEB 9.5-1, Section C.5.a(3). On the basis of its review, the staff concludes that the fire-barrier penetration seals meet the guidelines of BTP CMEB 9.5-1, Section C.5.a(3), and, therefore, are acceptable.

The applicant's Fire Protection Evaluation Report indicates that door openings in fire-rated barriers are provided with fire door assemblies that have ratings commensurate with the fire ratings of the walls in which they are installed, but does not state that these will be UL-labeled fire door assemblies. Suitability of fire doors is determined by test by nationally recognized testing laboratories, and doors not tested and not labeled cannot be relied on to provide effective protection. Therefore, the staff will require that the applicant test and label all fire door assemblies in accordance with National Fire Protection Assocation (NFPA) 252. "Fire Tests of Door Assemblies."

The heating, ventilation, and air conditioning (HVAC) penetrations of fire-rated barriers are provided with UL-labeled fire damper assemblies that have ratings commensurate with the fire ratings of the walls in which they are installed. The fire dampers are installed according to the manufacturer's directions. Fire dampers with 3-hour-fire ratings are provided in all 3-hour-fire-rated barriers with the exception of the 3-hour-fire-rated barrier that separates redundant file rooms on el 79 ft 8 in. of the auxiliary building. Because of physical constraints, the fire damper could not be installed in the barrier. The applicant has installed the damper in the HVAC ductwork approximately 15 ft from the wall and enclosed the intervening ductwork and supports in a 2-hour-fire-rated barrier. The in situ fuel load in the filter room is 52,975 Btu/ft^2, which if totally consumed would correspond to a 40-min fire on the American Society for Testing and Materials (ASTM) E-119 time temperature curve. Because the in situ fuel load produces a fire severity of less than 2 hours, the staff finds that the described damper installation is an acceptable deviation from BTP CMEB 9.5.1, Section C.5.a. On the basis of its evaluation, the staff concludes that the fire dampers with the approved deviation are provided in accordance with the guidelines of BTP CMEB 9.5-1, Section C.5.a, and are, therefore, acceptable.

Access and escape routes are provided for each fire area. Stairwells outside primary containment serving as access and egress routes are enclosed with fire barriers having 2-hour-fire ratings with 1-1/2-hour UL-labeled fire door assemblies at all openings into the stairwell. Fire exit routes will be clearly marked and established by prefire plans. On the basis of its review, the staff concludes that the applicant's fire protection program concerning access and egress routes meets the guidelines of BTP CMEB 9.5-1, Sections C.5(6) and (7) and is, therefore, acceptable.

Metal deck roof construction is either noncombustible or listed as Class I by Factory Mutual. Suspended ceilings and their supports are made of noncombustible materials, and concealed spaces above the suspended ceilings are devoid of combustible materials. The staff finds this to be in accordance with the guidelines of BTP CMEB 9.5-1, Sections C.5.a (10) and (11), and, therefore, acceptable.

High voltage-high amperage transformers installed inside buildings are of the dry type.

No oil-filled transformers are installed in buildings containing safetyrelated equipment, or within 50 ft or less from buildings containing safety-related equipment. Each oil-filled transformer is protected with a water spray extinguishing system that is automatically actuated by heat detectors and surrounded by moat-type construction designed to prevent the spread of oil or fire.

On the basis of its evaluation, the staff concludes that the installation of the transformers meets the guidelines of BTP CMEB 9.1-1, Sections C.5.a(12) and (13), and is, therefore, acceptable.

Fire Protection for Safe Shutdown Capability

The applicant has not provided the staff with the information necessary to perform an independent evaluation of the fire protection features that ensure safe shutdown capability. The staff will require the applicant to follow the fire protection guidance for safe shutdown contained in BTP CMEB 9.5-1, C.5.b.

Alternate Shutdown Capability

Alternate shutdown capability is provided for the control room and cable spreading room by a remote shutdown panel located in the west switchgear area on el 4 ft 6 in. of the control building. The staff has not completed the review of the alternate shutdown capability. It will report the results of its evaluation in a supplement to the SER.

Control of Combustible Materials

Safety-related systems have been isolated or separated from combustible materials as much as possible. The storage of flammable liquids complies with NFPA 30, "Flammable and Combustible Liquids Code."

The fire protection for the reactor coolant pumps and the diesel generator fuel oil day tanks is discussed in Section 9.5.1.6 of this report.

The turbine building does not contain any circuits or equipment of safe shutdown systems and is separated from such areas by 3-hour-rated fire walls.

On the basis of its evaluation, the staff concludes that combustible materials have been separated from safety-related systems or provided with suppression in accordance with the guidelines of BTP CMEB 9.5-1, Section C.5.d(1), and are, therefore, acceptable.

Except for the hydrogen seal oil unit in the turbine building, all flammable gas storage containers are stored outside buildings. All hydrogen piping located inside buildings in areas containing safety-related equipment is enclosed in steel guard piping that is vented to the atmosphere to prevent hydrogen buildup in the event of a hydrogen pipe break.

On the basis of its evaluation, the staff concludes that the hydrogen storage and piping meet the guidelines of BTP CMEB 9.5-1, Sections C.5.d(2) and (5), and are, therefore, acceptable.

Electrical Cable Construction, Cable Trays, and Cable Penetrations

All cable trays are of steel construction. Electrical cable construction passes the IEEE Std. 383-1974 flame test. The cables are designed to allow wetting down with fire suppression water without electrical faulting.

Cable tray penetrations will have fire ratings that are at least equal to the ratings of fire barriers in which they are installed.

Automatic water suppression systems are not provided for cables in heavily cabled areas, including the cable tunnels, motor control center area, and rod control areas of the plant. The applicant has not provided adequate information to indicate that such areas are protected in accordance with staff guidelines. The staff will require the applicant to meet the guidelines of BTP CMEB 9.5-1, Section C.5.e(2), for the protection of cables outside the cable spreading room.

Ventilation

The cable tunnels and cable spreading rooms have dedicated ventilation systems designed specifically to exhaust smoke or other products of combustion. Normal plant ventilation systems will be used in other areas of the plant for this purpose. Portable smoke ejectors will be provided to assist in removal of the products of combustion should the normal ventilation systems be unavailable because of damper closures or other failures. The staff finds this acceptable.

Where total flooding gas extinguishing systems are used, air intake and exhaust ventilation dampers are provided with mechanisms that will close them on initiation of gas flow. The staff finds this acceptable. Fire-barrier ventilation openings are provided with fire dampers that will close if a fire should cause room temperature to exceed a set value.

Fresh air supply intakes to areas containing safety-related equipment or systems are remote from exhaust air outlets of other fire areas. Stairwells are designed to minimize smoke infiltration during a fire.

On the basis of its evaluation, the staff finds that the ventilation system meets the guidelines of BTP CMEB 9.5-1 Section C.5.f, and is, therefore, acceptable.

Lighting and Communication

Emergency lighting will be installed in all areas of the plant that may have to be manned for safe shutdown operations and in access and egress routes to and from all areas. The emergency lighting consists of fixed, self-contained fluorescent or sealed-beam units with individual 8-hour minimum battery power supplies. The emergency lighting system provides illumination at all points where equipment operation is needed for shutdown as well as at all points on the floor, including angles and intersections of corridors, passageways, and stairways, of not less than 3.0 ft candles measured at the floor. The staff concludes that the emergency lighting meets the guidelines of BTP CMEB 9.5-1, Section C.5.g(1), and is, therefore, acceptable.

The applicant has provided a fixed emergency communication system that is independent of the normal plant communication system at preselected stations. A portable radio communications system has been provided for use by the fire brigade. Fixed repeaters have been installed for portable radio communication. If a fire should damage the Unit 3 fixed repeater station, the plant's portable radios have been equipped with multiband frequency capability. This multiband frequency capability will allow plant personnel to continue communications using the base station as backup communication center or the capability to change frequency bands and operate through either adjacent plant's fixed repeater system. The staff concludes that two-way voice communication meets the guidelines in BTP CMEB 9.5-1, Section C.5.g(4), and is, therefore, acceptable.

9.5.1.5 Fire Detection and Suppression

Fire Detection

The fire detection system consists of the detectors, associated electrical circuitry, and electrical power supplies. The types of detectors used are products-of-combustion, rate-of-rise, and fixed temperature detectors. The systems provide distinctive audible and visual alarms locally and in the control room. Detection devices are installed in all areas containing or presenting fire exposure to safety-related equipment.

The fire detection system complies with the requirements of NFPA 72D for a Class A system.

Primary and secondary power supplies for the fire detection system satisfy the provisions of Section 2220 of NFPA 72D. The staff finds this acceptable.

The applicant's Fire Protection Evaluation Report does not indicate that fire detectors have been selected and installed in accordance with NFPA 72E. The staff will require the applicant to select and install early warning fire detectors as a minimum in accordance with NFPA 72E.

Fire Protection Water Supply System

An underground yard fire main loop has been installed to furnish anticipated water requirements. This loop is an extension of the yard fire main loop serving Millstone Units 1 and 2 and is provided with post-indicator-type sectional control valves that permit isolating portions of the loop and maintaining independence of the individual loop around each unit, except that the fire water supply (storage tanks and fire pumps) is not independent of the Unit 1 and 2 loops.

Valves are installed to permit isolation of outside hydrants from the fire main. The fire main system piping is independent of the service water and sanitary water system piping.

The water supply system consists of three fire pumps; two pumps are electrically driven and one is diesel engine driven. One of the electric pumps is in the Millstone Unit 2 pumphouse; the other pumps are in the Millstone Unit 1 pumphouse. Each fire pump is separately connected to a buried [illegible]-in. water main loop around the plant. Each fire pump has a rated capacity of 2,000 gpm. The fire water demand can be met with one fire pump out of service.

The fire protection water supply system is kept pressurized by a fire service jockey pump. The fire pumps are automatically started when low pressure is sensed in the pump discharge header. Each pump can be stopped manually at its local control panel. Separate audible and visual alarms are provided in the Unit 1 and Unit 2 control rooms to monitor the status of the fire pumps, prime mover availability, power failure, and failure of the fire pumps to start.

The water supply for fire protection is taken from two 245,000-gal water storage tanks located at Millstone Units 1 and 2. The suction piping to the three fire pumps is arranged to permit suction from either or both of the tanks. The tanks are provided with valves so that a leak in one tank or its associated piping would not cause both tanks to drain. Water supply to the tanks is through a 12-in.-diameter city water main, which can refill a tank in 8 hours or less. Well water also is available through a separate connection (normally disconnected).

The greatest demand for fire water is from the Millstone Unit 3 turbine building sprinkler system, approximately 1,500 gpm. This demand and an additional

500 gpm for hose streams can be met with one pump out of service from one tank for the required 2 hours (2 hr x 2,000 gpm x 60 min/hr = 240,000 gal). On the basis of this volume and the automatic makeup from the city water supply, the staff finds that the size of these existing tanks is an acceptable deviation from the guidelines of BTP CMEB C.6.b(11), which require a minimum volume of 300,000 gal per tank.

Yard hydrants are provided at intervals of 250 ft along the fire protection water supply loop. The lateral to each yard hydrant is provided with an isolation valve to facilitate hydrant maintenance and repairs without shutting down any part of the fire water supply system. Standard hose houses are provided. Sectional control valves are provided to isolate portions of the underground main for maintenance or repair without shutting off the supply to primary and backup fire suppression systems that serve areas containing safety-related systems or in which safety-related systems are exposed.

By letter dated March 9, 1984, the applicant stated that all valves in the fire protection water supply system are supervised in accordance with the guidelines in BTP CMEB 9.5-1, Section C.6.c(2).

On the basis of its review and the applicant's statement, the staff concludes that the fire protection water supply system meets the guidelines in Section C.6.c of BTP CMEB 9.5-1 and is, therefore, acceptable.

Water Sprinkler and Hose Standpipe Systems

All sprinkler and hose station standpipe systems have independent yard fire main connections, except for the emergency generator enclosure, service building, waste disposal building, containment building, and the auxiliary boiler room. Thus, a single active failure or a crack in a moderate-energy line could impair both the primary and backup fire suppression systems in these areas. Headers in the turbine building are arranged to prevent single failure from impairing either primary or backup water supply by using three physically separate fire-water supply mains internally cross-connected. The staff will require the applicant to provide a fire protection water supply for the emergency generator enclosure, service building, waste disposal building, containment, and the auxiliary boiler room so that a single break or failure in the supply piping will not result in the loss of both the primary and secondary water supplies.

The wet pipe sprinkler systems meet the provisions of NFPA 13, "Standard for the Installation of Sprinkler Systems," and NFPA 15, "Standard for Water Spray Fixed Systems for Fire Protection." The areas equipped with water suppression systems are listed in Section 9.5-1.2.1 of the FSAR.

Manual hose stations are located throughout the plant in accordance with NFPA 14, "Standard for the Installation of Standpipe and Hose Systems." There are no hose stations in the control building. To meet the guidelines of BTP CMEB 9.5-1, Section C.1.c, the staff will require the applicant to install manual hose stations so that all areas of the control building can be reached with an effective hose stream.

BTP CMEB 9.5-1, Section C.6.c, recommends that standpipes be sized 4 in. in diameter for multiple hose station supplies and 2½ in. in diameter for single hose station supplies. The applicant has provided standpipe of a smaller size. The staff will require the applicant to either verify that the smaller sized standpipe is capable of providing the 500-gpm hose streams at adequate pressure for manual fire fighting operations or increase the size of the piping in the standpipe system.

The applicant has not identified the seismic design of standpipe systems, which is recommended in BTP CMEB 9.5-1, Item C.6.c.(1). For plants with construction permits issued before July 30, 1976, the guidelines contained in Appendix A to BTP APCSB 9.5-1 do not contain a requirement for seismic design of standpipe systems. Therefore, this is an acceptable deviation from the guidelines of BTP CMEB 9.5-1, Item C.6.c.(1).

Carbon Dioxide Suppression System

Low-pressure carbon dioxide, automatic total flooding systems are provided for primary protection in the control building (cable spreading area and emergency switchgear areas), service building (switchgear area and cable tunnels), auxiliary building (motor control center and rod control areas), turbine building (front standard and alternate/exciter housing), and emergency generator fuel oil tank vault areas.

The acceptability of the carbon dioxide system in the cable spreading room is discussed in Section 9.5.1.6. The systems are activated by heat detectors that alarm and annunciate locally and in the control room. The carbon dioxide systems may also be activated manually. The systems are designed and installed in accordance with NFPA 12, "Carbon Dioxide Extinguishing Systems."

On the basis of its evaluation, the staff concludes that the carbon dioxide extinguishing systems meet the guidelines of BTP CMEB 9.5-1, Section C.6.d, and are, therefore, acceptable.

Halon Suppression Systems

Total flooding Halon 1301 systems are provided for the computer room and instrument rack room underfloor area and the records file room of the warehouse. The systems are designed to provide an initial concentration of 6% to 7% by volume of Halon 1301 within 10 sec of initiation. The Halon 1301 suppression systems are to be manually initiated on receipt of fire alarms or automatically discharged on operation of the heat detectors in the area.

On the basis of its review, the staff concludes that the Halon 1301 extinguishing systems meet the guidelines of BTP CMEB 9.5-1, Section C.6.e, and are, therefore, acceptable.

Portable Extinguishers

Portable fire extinguishers are provided to conform with the guidelines of NFPA 10. On the basis of its review, the staff concludes that the fire extinguishers meet the guidelines of BTP CMEB 9.5-1, Section C.6.f, and are, therefore, acceptable.

9.5.1.6 Fire Protection of Specific Plant Areas

Containment

The containment building is separated from adjacent buildings by 3-hour-fire-rated barriers. The containment building fire protection features include hose stations, ionization smoke detectors, and portable fire extinguishers.

The applicant committed to provide oil collection systems for each reactor coolant pump with a collection tank for each pump sized to contain the entire pump oil inventory in accordance with Section III.O of Appendix R. The staff finds this acceptable.

On the basis of its review, the staff concludes that the fire protection for containment with the commitments meets the guidelines of BTP CMEB 9.5-1, Section C.7.a, and is, therefore, acceptable.

Control Room

The control room complex is separated from all other areas of the plant by 3-hour-fire-rated assemblies. Smoke detectors have been installed in the control room and the main control room console. All smoke detector alarms are annunciated on the control room panel. Portable fire extinguishers inside the control room and hose stations outside the control room are provided in accordance with Section C.7.b of BTP CMEB 9.5-1.

The applicant has provided an alternate shutdown system for the control room. The alternate shutdown system is reviewed in Section 9.5.1.4 of this report.

The outside air intakes for the control room ventilation systems are equipped with smoke detectors that alarm in the control room.

On the basis of its review, the staff concludes that the fire protection for the control room complex is in accordance with BTP CMEB 9.5-1, Section C.7.b, and is, therefore, acceptable.

Cable Spreading Room

The cable spreading room is separated from the balance of the plant by 3-hour-fire-rated barriers. Both safe shutdown divisions are installed in the room. The applicant has provided an alternate shutdown system for the cable spreading room. The alternate shutdown system is reviewed in Section 9.5.1.4 of this report.

A separate ventilation system has been provided to automatically exhaust the cable spreading room in the event of a fire. Portable blowers will be used to manually remove the smoke.

Smoke detectors have been installed to provide early-warning fire detection. Portable fire extinguishers provide manual fire fighting capability.

The primary means of fire suppression in the cable spreading room is a total flooding automatic carbon dioxide system. The staff will require the applicant to provide a fixed water suppression system as a backup to the carbon dioxide system to meet the guidelines of BTP CMEB 9.5-1, Section C.7.c.

Switchgear Rooms

The switchgear rooms are separated from each other and from other plant areas by 3-hour-fire-rated walls and floor/ceiling assemblies. Automatic fire detection is provided by heat and smoke detectors. Automatic fire suppression is provided by carbon dioxide extinguishing systems. Manual protection is provided by standpipe hose stations and portable extinguishers.

Floor drains have not been provided in the switchgear rooms to prevent damage to equipment from water used for fire fighting purposes.

By letter dated March 9, 1984, the applicant committed to install 4-in. high watertight curbs at all door openings between the switchgear rooms and adjacent fire areas to prevent water from entering the switchgear rooms. The staff finds the applicant's commitment acceptable.

On the basis of its review, the staff concludes that the fire protection for the switchgear rooms is in accordance with BTP CMEB 9.5-1, Section C.7.e, and is, therefore, acceptable.

Remote Safety-Related Panels

The remote shutdown panels are separated from the remainder of the plant by walls and floor/ceiling assemblies with fire ratings of 3 hours. Heat and smoke detectors are located at various points in the area. Manual fire suppression capability is provided by portable fire extinguishers. The staff finds that the fire protection for this area is in accordance with the guidelines of BTP CMEB 9.5-1, Section C.7.f, and is, therefore, acceptable.

Safety-Related Battery Rooms

The battery rooms are separated from each other and from the balance of the plant by 3-hour-fire-rated barriers. Smoke detection systems are provided in each battery room. Hose stations and portable fire extinguishers are available in the areas for manual fire suppression. The ventilation system is designed to maintain the hydrogen levels below 2%. Air flow monitors that alarm in the control room to monitor the loss of ventilation have been provided in each battery room.

On the basis of the above evaluation, the staff concludes that the fire protection for the battery rooms meets the guidelines of BTP CMEB 9.5-1, Section C.7.g, and is, therefore, acceptable.

Emergency Diesel Generator Rooms

Each diesel generator is located in a different fire area separated by 3-hour-fire-rated barriers. All cable and piping penetrations through the fire-rated barriers are fitted with 3-hour-fire-rated penetration seals.

A 550-gal diesel fuel oil day tank is located in each diesel generator room. Each fuel oil day tank is provided with an oil collection system which will be connected by hard piping to an underground storage oil separator tank. The total capacity of the collection/draining system will be 110% of the day tank capacity. The staff finds this acceptable.

Each diesel generator room is protected by an automatic preaction sprinkler system with separate heat detectors. Manual fire suppression capability is provided by hydrants and portable fire extinguishers.

On the basis of its review, the staff concludes that the protection provided for the diesel generators meets the guidelines of Section C.7.i of BTP CMEB 9.5-1, and is, therefore, acceptable.

Other Plant Areas

The applicant's fire hazards analysis addressed other plant areas not specifically discussed in this report. The staff finds that the fire protection for these areas is in accordance with the guidelines of BTP CMEB 9.5-1 and is, therefore, acceptable.

9.5.1.7 Summary of Deviations From BTP CMEB 9.5-1

The following deviations from the guidelines of BTP CMEB 9.5-1 have been identified and approved:

(1) installation of a 3-hour-rated damper in the ductwork rather than in the wall (Section 9.5.1.4)

(2) fire water supply tank size (Section 9.5.1.5)

(3) no connection of the standpipe system to a seismic Category I water system (Section 9.5.1.5)

(4) no floor drains in the switchgear rooms (Section 9.5.1.6)

9.5.1.8 Conclusions

The following are the unresolved fire protection items:

(1) potential systems interaction (Section 9.5.1.1)
(2) qualification of fire doors (Section 9.5.1.4)
(3) safe shutdown capability (Section 9.5.1.4)
(4) alternate shutdown capability (Section 9.5.1.4)
(5) protection of cables outside cable spreading room (Section 9.5.1.4)
(6) installation of fire detectors (Section 9.5.1.5)
(7) independent sprinkler and hose station connections (Section 9.5.1.5)
(8) manual hose coverage (Section 9.5.1.5)
(9) hose station standpipe diameters (Section 9.5.1.5)
(10) cable spreading room protection (Section 9.5.1.6)

9.5.2 Communication Systems

The communication system is designed to provide reliable intraplant and interplant (or plant-to-offsite) communications under both normal plant operation and accident conditions.

9.5.2.1 Intraplant Systems

The intraplant communication systems provide sufficient equipment of various types so that the plant has adequate communications to start up, continue safe operation, or safely shut down. The intraplant systems include:

(1) Voice Paging (Public Address) System

The intraplant voice paging (public address (PA)) system provides communications from the control room to all buildings and control areas within the unit. In addition, through interconnections with the telephone system, the PA system provides communication from one control area to any other. Isolation is provided between the two systems, which have different operating voltages and impedances. The PA system is an independent system using separate amplifiers and speakers at each paging station. PA loudspeaker stations are provided in all buildings that compose the plant and in the outside areas surrounding the plant. Access to voice paging speakers is provided and initiated by dialing a code number from any plant dial telephone. The control room has priority access to the PA system. This access bypasses the plant telephone system.

The PA system consists of loudspeaker stations, amplifiers, a telephone interface, two page override handsets, and a multitone generator. The PA system is powered from a nonvital bus, which ultimately is fed from a Class 1E motor control center via a battery charger. The battery charger is seismically qualified and mounted and is located in the control building.

The loudspeaker stations are suitable for operation in conjunction with the loudspeaker amplifiers. Horn-type speakers have accessories suitable for mounting on horizontal or vertical structural surfaces. The amplifiers are suitable for operation on a 120-V, 60-Hz, single-phase supply. Rated output of unit loudspeaker amplifiers is not less than 12 W.

Each handset station includes a handset, a hookswitch, amplifier, terminal facilities, page/party spring-loaded selector switch, and 6 ft of self-coiling cord. The handsets include a magnetic receiver and a low impedance noise-cancelling transmitter. These handsets are located in the control room and at the auxiliary shutdown panel and include an override control for paging.

The multitone generator that is used in the evacuation alarm system provides a signal source to the paging system producing five distinctive tones. These tones are steady, pulse, siren, warble, and yelp. The tone generator transmits the designated evacuation. The alarm tone overrides the paging system to ensure that it is audible throughout the plant.

(2) Telephone (Plant Switching Network) System

The telephone (plant switching network) system is supplied by Southern New England Telephone Company (SNETCo) and consists of standard telephones, multiline telephones, and a Dimension 2000 switch, which is capable of handling 1,500 to 2,000 lines.

Dimension 2000 and its associated telephones allow communication throughout the plant by dialing the appropriate four-digit extension number. Communication on site, off site, or with the Emergency Operations Facility (EOF) is accomplished by dialing the appropriate tie-line code(s). Currently, there are five tie lines to the EOF.

The plant switching network is powered from the normal ac system. Emergency power is available through standby batteries and a dc to ac inverter. The plant switching network is directly coupled to the telephone company's message network and the voice paging system.

The message network is the entire communications system established and operated by SNETCo and the neighboring telephone companies. The message network connects both public and private facilities. It is tied directly to the plant switching network with multiple Bell System central office tie lines. The plant telephone system is also tied remotely to the message network through Bell System dial repeating tie trunks and dedicated microwave system dial repeating tie trunk.

(3) Maintenance Jack System

The maintenance jack system, which is used for calibration and maintenance, consists of amplifiers, headsets, handsets, and a network of plug-in jack stations with five-party selector switches located throughout the plant. Its power source is the same as that of the PA system.

Jack stations are mounted on control panels or in separate enclosures. Each station contains a six-position selector switch (position for each of the five channels and an off position) and a receptacle to receive the plug unit of the headset or handset. Those jack stations that are mounted in separate enclosures have a provision to cover the receptacle when the station is not in use.

Headsets and handsets contain speaker(s), a microphone assembly, 6 ft of retractable cord, and a plug suitable to mate with the receptacle of the jack stations.

A system amplifier located in the emergency switchgear room 2 consists of five independent amplifiers, each capable of driving a channel.

The maintenance jack system does not interface with any other communication system.

(4) Reactor Fuel Handling Carrier Phone System

The fuel-handling carrier phone system consists of an amplifier, jack plug stations, and handsets. Its power source is the same as that of the PA system.

Jack stations are mounted in separate enclosures. Each station contains a receptacle to receive the plug unit of a handset, as well as provisions to cover the receptacle when the station is not in use. The jack stations are of single-channel design.

Handsets include a speaker, a microphone assembly, 6 ft of retractable cord, and a plug to mate with the receptacle of the jack stations.

The amplifier, located in the auxiliary building, is a single-party-type component, capable of driving the single channel.

The jack stations are located on the spent fuel pool bridge, manipulator crane, five locations in the containment at various elevations, and four locations in the fuel building.

The fuel-handling carrier phone system does not interface with any other communication system.

(5) Sound-Powered Telephone System

The sound-powered telephone system consists of a master station, a switchbox, and eight substations with handsets. The system is self powered.

Each substation includes a hand-held telephone with a push-to-operate button located on the handset, a handset holder, and a wall-mounted aluminum case containing a manually operated magneto generator for call signaling and an audible call-signal device. The master station, in addition to the equipment furnished with a substation, includes a selector switch (for calling substations individually) and a switchbox containing eight 6-pole switches for disconnecting any faulted substation cable in the system.

The master station is located in the auxiliary shutdown panel area. Substations are located in the emergency generator enclosures, the emergency switchgear rooms, the main control room, the charging pump control cubicle, engineered safety features building, and the service water pumphouse.

The sound-powered telephone system does not interface with any other communication system.

(6) Intraplant Radio Systems

The intraplant radio systems include the following:

(a) Radio Remote Control Console

A dedicated radio remote control console is provided in the Millstone Unit 3 control room for communications with all associated onsite as well as offsite radio facilities as described in Section 9.5.2.2. Its power source is the same as that of the PA system. Normally, all radio systems, except the unit's operations and maintenance (O&M) system, are quiet to the unit operator unless selected by the operator for monitoring or operation. Tone alert, except on the O&M system, is provided to enable remotely located radio dispatchers to contact the control room operator.

(b) Multifrequency UHF Repeater System

An ultra high frequency (UHF) repeater system is dedicated to plant operations and maintenance activities. In the event of repeater failure, a "talk-around" feature can be accessed from the control room radio console and will allow communications to continue without the repeater. The radio console is able to access similar but separate UHF radio repeater systems at Millstone Units 1 and 2, as well as site security.

(c) Onsite Paging, Hand-Held, and Mobile UHF Repeater O&M Radio System

The O&M radio system is used for two-way communications by station operating and maintenance personnel and is controlled by the consoles in Units 1, 2, and 3. This system consists of a control/base station and a repeater relay station. The control/base station is installed in the warehouse No. 5 telecommunications room in Millstone Unit 3. It is fully solid state, incorporating integrated circuitry located on plug-in modules or independent printed circuit boards.

The repeater relay station is installed in the Millstone Unit 3 model shop. It is fully solid state, incorporating integrated circuits located on modular plug-in circuit boards, and is protected against overcurrent conditions and power surges.

(d) Onsite Hand-Held and Mobile UHF Repeater Security Radio System

The security radio system consists of one repeater/relay station and three control/base stations. System control is from the radio consoles in Units 1, 2, and 3, the security central and secondary alarm stations, and the EOF.

The repeater/relay station is fully solid state, incorporating integrated circuits located on modular plug-in circuit boards. The station is protected against lighting, overcurrent, and power surges. The security control/base station is a compact two-way radio suitable for desk top mounting and fully utilizes the advantages of solid-state circuits (i.e., reliability, small size, ruggedness, and low maintenance requirements).

The O&M and security radio systems are powered from the normal 120 Vac. The portable units are powered by rechargeable batteries.

Actual demonstration of the installed intraplant communication systems will check for effective communication between plant personnel in all vital areas during maximum potential noise levels. The outcome of these high noise level tests may lead to some modifications of the installation. Preoperational testing of the intraplant radio systems will identify those locations on site where hand-held radio communication devices will be prohibited because of radio frequency interference with control and instrumentation; administrative procedures will prevent the use of hand-held UHF radios in those locations from affecting the solid-state reactor protection and/or ESF systems.

The applicant has shown that the integrated design of the communication system provides adequate communication, from onsite power sources, for fire fighting purposes and control and maintenance of safety-related equipment. The integrated design consists of separate and independent systems. Component failures in one of these systems will have limited effect on that system and no effect on other systems. Loss of offsite power would have limited effect on communication capabilities because of the interconnection with the Class 1E motor control center. Severing of one communication line or trunk in a localized plant area will have limited effect on that system and no effect on the other systems.

However, a fire or accident in the purple switchgear room or the purple cable tunnel may impact the capability of the voice paging system, the maintenance jack system, the sound-powered telephone system, and portions of the plant switching network. In the event of a fire or accident in either of these two areas, cold shutdown can be achieved by operator action from the main control room. If communication is required, hand-held portable radios will be used.

9.5.2.2 Interplant (Plant-to-Offsite) Communication System

The design basis for interplant communications is to provide dependable communication for reliable operation. The interplant communication systems include:

(1) Telephone Communication Systems

Telephone communication systems are discussed in Section 9.5.2.1 of this SER.

(2) Evacuation Alarm Systems

The evacuation alarm system is discussed in Section 9.5.2.1 of this SER.

(3) Microwave System

The Northeast Utilities (NU) microwave system provides all three generating units at the Millstone site with a reliable telecommunications medium. The microwave system links the Millstone site to other key facilities within the NU-franchised service area as well as other utility companies throughout New England.

The microwave system uses frequency-modulated low-power radio signals that operate in the 2,000- and 6,000-MHz industrial microwave frequency bands established for industrial users by the Federal Communications Commission. The site system is powered from batteries backed up by a nonvital diesel generator at the Millstone site.

The type of telecommunications traffic that is placed on the microwave system is the same type that would normally be placed on a dedicated, four-wire, data-grade telephone circuit. This would include the following:

(a) Dial-repeating tie trunks or tie lines that connect the telephone private branch exchange (PBX) at one location within the NU system to a similar PBX at another location.

(b) Automatic ringdown circuits for use as hotline-dedicated phones, where lifting a phone at one end will cause the phone on the other end of the circuit to ring.

(c) Radio control circuits that provide control of remotely located radio transmitters from key areas within the Millstone complex. This includes radio control circuits that provide one-way control as required by radio paging transmitters as well as control circuits that provide two-way control for standard mobile radio operation.

(d) Data circuits that connect one computer with another or allow data-gathering equipment to communicate with a central host computer. These circuits use data rates up to and including 9,600 baud with a very high degree of reliability.

(e) Data circuits that carry analog data. These circuits also can benefit from the greater reliability offered by the microwave system. This type of telecommunications traffic includes telemetering of important analog quantities and reporting alarms that are remote from the Millstone site.

(f) Data circuits used for protective relaying signals. These circuits provide the electric generating and transmission system with protection from catastrophic failure.

The microwave system provides the Millstone site with an additional telecommunications network that is completely separate from the offsite telephone system. The use of two diverse systems to share the telecommunications requirements of the Millstone site results in enhanced telecommunications reliability because a failure of either system will not completely interrupt offsite telecommunications traffic. The microwave system will also allow Millstone to access a modern telephone PBX located approximately 50 mi from the site at the NU headquarters in Berlin, Connecticut. In an emergency situation, NU personnel would be able to displace less critical microwave channels with the additional traffic from the Millstone site.

(4) Emergency Notification System

This system is a direct NRC hotline telephone that will connect the Millstone Unit 3 control room with the NRC operations monitoring facility in King of Prussia, Pennsylvania. The system consists of an automatic ringdown phone in the control room at Millstone Unit 3. This phone is connected to a dedicated AT&T long line and is independent of the station PBX. A dedicated ac power supply with backup power capabilities provides signaling and ringing power. The system is equipped with a failure lamp for indications of circuit problems.

(5) Multiple Dedicated Automatic Ringdown Telephones

This system consists of automatic ringdown telephones from the Millstone Unit 3 control room to the Connecticut State Police, the Waterford Police, the Berlin Emergency Operations Center, the site Emergency Operations Facility, and the site Technical Support Center. All of these telephones receive their power for signaling and ringing from SNETCo's New London office via individual hard-wire pairs. All circuits are independent of the station PBX.

(6) Connecticut Valley Electric Exchange (CONVEX) Dispatch Loop

The CONVEX dispatch loop (also known as the State Wide Dispatch Loop and the Full Period Phone System) is a dedicated party line system provided by the telephone company. The CONVEX dispatch loop provides telecommunications service from the load control dispatch center (CONVEX) to the generation and substation facilities. Each location that is served by the CONVEX dispatch loop uses the following telephone components:

(a) A speaker amplifier with volume control is used to monitor voice traffic on the loop.

(b) A signaling tone operated by a pushbutton is provided to gain the attention of the person monitoring the loop.

(c) A telephone-style handset is used to provide two-way communications.

The telephone equipment used in this system is powered by reliable dc sources such as lead calcium station batteries. This power is supplied at some locations with telephone battery power and at other locations with NU station battery power.

(7) Control Room Intercom System

The control room intercom system provides a communications link between the control rooms of Units 1, 2, and 3. The intercom operates independently of the plant switching network and voice paging systems.

(8) Interplant Radio Systems

The interplant radio systems include the following:

(a) CONVEX Command Control Network (CCN)

The CONVEX CCN is a two-way radio system using tone alert signaling to provide communications between the control room, the CONVEX load dispatcher, and other key NU operations facilities.

This system is controlled by the radio console in Units 1, 2, and 3. The transmitter/receiver base station is installed in the warehouse no. 5 telecommunication room at Millstone Unit 3 and is powered from the normal ac power source. The base station is fully solid state, incorporating integrated circuitry located on plug-in modules or independent printed circuit boards.

(b) Waterford Police Radio System

The Waterford Police Department two-way radio system provides communications between the Waterford police radio dispatcher and control room.

Base station description and location are the same as that for the CCN.

(c) Tri-Town UHF Radio System

The Tri-Town UHF radio system is an administrative two-way radio system used by three towns in the Millstone area. Each of these towns has the ability to call the control room using tone alert signaling.

The system is controlled by the consoles in Units 1, 2, and 3 and the base/control station and repeater relay station. The base/control station is located in the warehouse no. 3 receiving office. It contains two transmitting frequencies; the second frequency is "talk-around" in the event of a repeater relay station failure. The primary power source is 120 Vac with dual 13.8-Vdc battery backup. The station is fully solid state.

The repeater relay station is installed at the base of the Millstone stack and is enclosed in a weatherproof cabinet. The repeater is fully solid state and its primary power source is 120 Vac with a 13.8-Vdc battery backup that will energize 1 min after primary power interruption.

(d) State Police Radio System

The State Police two-way radio system operates on two frequencies. One frequency is used for radio tests and short-duration communications. The other frequency is used for communications over extended periods of time. Tone alert signaling allows the State Police radio dispatcher to call the control room.

The system is controlled by the consoles in Units 1, 2, and 3. The station is a desk-top style and is located in the Unit 1 control room. The unit is fully solid state and uses dedicated lines for control. The primary power source is 120 Vac.

(e) Connecticut Light & Power Company (CL&P) Radio System

The CL&P two-way radio system allows the control room to access a utility radio system equipped with a large number of radio-equipped vehicles.

This system is controlled by the radio console in Units 1, 2, and 3. The transmitter/receiver base station is installed in the warehouse no. 5 telecommunications room at Millstone Unit 3 and is powered from the normal ac power source. The base station is fully solid state, incorporating integrated circuitry located on plug-in modules or independent printed circuit boards.

(f) Very High Frequency (VHF) Radio Paging System

An automated VHF radio paging system is provided so that control room personnel can rapidly notify state, local, plant, and NU personnel in the event of any abnormal condition at the site. This system controls multiple base station radios located throughout NU's franchised operating area, and is controlled by the consoles in Units 1, 2, and 3. Each console contains an auto-page encoder module.

The scope of review of the plant communication systems included the assessment of the number and types of communication systems provided and of the adequacy of the power sources and verification of the functional capability of the communication systems under all conditions of operation.

The basis for acceptance in the staff review was conformance of the design criteria and bases and design of the installed communication systems to the acceptance criteria and guidance of SRP Section 9.5.2. Other bases for acceptance were conformance to industry standards and the ability of the systems to provide effective communication from diverse means within Millstone Unit 3 under maximum potential noise levels.

On the basis of its review, the staff concludes that the installed communication systems at Millstone Unit 3 conform to the above-cited standards, criteria, and design bases and can perform their design functions. They are, therefore, acceptable.

Special communication system requirements for fire protection are addressed in Section 9.5.1 of this SER.

9.5.3 Lighting System

The lighting system for Millstone Unit 3 is designed to provide adequate lighting in all areas of the station and consists of a normal and two standby (essential) ac lighting systems and an emergency dc (battery-pack) lighting system. The design is based on illumination levels that equal or exceed those recommended by the Illuminating Engineering Society for central stations and NUREG-0700, "Guidelines for Control Room Design Review."

9.5.3.1 AC Normal Lighting System

The normal lighting system provides general illumination for the station and is fed from local non-Class 1E motor control centers through dry-type 480/208-120 Vac lighting transformers.

9.5.3.2 AC Standby (Essential) Lighting System

The essential lighting system supplements the normal lighting system and provides illumination for operation in the control room, the emergency switchgear rooms including the auxiliary shutdown panel, the diesel generator rooms, emergency feedwater pump room, and other safety-related and vital areas required to bring the plant to a safe shutdown. In addition, access and egress paths for plant evacuation are provided with lighting from this system.

The essential ac lighting system is supplied from the emergency "orange" or "purple" 480/208 Vac, three-phase, dry-type voltage-regulating transformers. The output of the transformers, although "black," is run exclusively in conduit and does not share raceways with normal "black" power, emergency power, or with "black" power that originates from transformer supplies from the opposite emergency bus. The output of the transformer is protected by an air circuit breaker. The essential ac lighting system operates continuously, with the exception of the lighting in the containment. On loss of offsite ac power, the essential ac lighting system is automatically energized via the emergency ac power source (i.e., emergency generator).

9.5.3.3 DC Emergency Lighting System

The dc lighting system consists of 8-hour self-contained, sealed-beam battery packs. These battery packs are supplied with a trickle charge via the Class 1E ac power system which, in the event of a loss of offsite power, is supplied automatically from the emergency generator. The dc lighting system operates on loss of the normal or essential ac lighting system depending on the area of the plant. Once the essential ac lighting system is energized, the dc lighting system will shut off. The dc lighting system provides lighting for the control room, emergency switchgear rooms including the auxiliary shutdown panel, other safety-related and vital areas, and access and egress paths for personnel evacuation throughout the station.

The applicant was requested to identify the emergency lighting in the safety-related areas of the plant and to show that adequate illumination is provided in those areas for fire-fighting purposes, and control and maintenance of safety-related equipment. In a letter dated May 17, 1984, the applicant provided the information on the lighting system. The staff has evaluated this information and has determined that the emergency lighting system is unacceptable in the auxiliary shutdown panel area and that inadequate illumination is provided in the safety-related areas.

The auxiliary shutdown panel is located in the purple division switchgear room. Emergency lighting is provided by the purple division ac essential lighting system and the dc emergency lighting system. The ac emergency lighting system provides average illumination of 10 ft-candles, and the dc lighting system provides 4 ft-candles of illumination at the panel. The staff has determined that a minimum of 10 ft-candles at the work station is required to perform the functions necessary to shut down the plant. Use of the auxiliary shutdown panel along with the failure of the purple train ac lighting system will result in degraded and inadequate illumination at the panel area. Should such a condition occur, the applicant states that sufficient portable lighting packs will supplement the dc emergency lighting system to provide an average of 10 ft-candle light intensity at the shutdown panel to enable the operator to perform the necessary functions. The portable lighting packs, at best, will last about 8 hours before replacements must be brought in. The staff believes that, under such postulated conditions, it is quite likely that the shutdown panel area may be manned for a period of days and the use of portable lighting packs to supplement the dc emergency lighting system does not appear to be prudent or feasible, and is, therefore, not acceptable.

Under station blackout events, work may have to be performed in the purple switchgear room to facilitate restoration of ac power. This would require adequate dc illumination (minimum 10 ft-candles) at the work stations - switchgear panels. Therefore, for the auxiliary shutdown panel area/purple switchgear room, the staff requires the following:

(1) The illumination level provided by the purple emergency ac lighting system should be increased to a minimum of 10 ft-candles over the work area.

(2) Adequate ac lighting (a minimum of 10 ft-candles) should be provided in the auxiliary shutdown area from the other train of the ac lighting system.

(3) The illumination level provided by the dc lighting system should be increased to a minimum of 10 ft-candles in those areas of the purple switchgear room where work may be performed to restore ac power.

The other safety-related areas of the plant are inadequately illuminated by the ac emergency lighting system and/or the dc lighting system. The illumination levels range from 3 and 4 ft-candles from the dc lighting system in the control room and diesel generator control panel areas, respectively, to 0.75 and 0.5 ft-candles from the emergency ac and dc lighting systems, respectively, for the safety-related areas and access and egress paths to those areas.

Therefore, the staff requires the following:

(1) Since the control room, the orange switchgear room, and the diesel generator room may require access during certain events so that ac power can be restored, the dc emergency lighting system illumination intensity shall be increased to a minimum of 10 ft-candles at those work stations where work may be performed to restore ac power. In addition, the ac emergency lighting system illumination intensity shall be increased to a minimum of 10 ft-candles at the work station instead of an average of 10 ft-candles.

(2) For the other safety-related areas, the illumination intensity shall be increased to a minimum of 10 ft-candles at the panel surfaces and at the work stations and 2 to 5 ft-candles on the basis of the activity level for access and egress to safety-related plant areas.

The applicant has been informed of these positions.

The plant lighting systems are designed so that a single failure cannot degrade the essential lighting below a safe level, except as noted above. In addition, battery-powered portable lighting will be provided for emergency use by the fire brigade and operations personnel required to achieve safe plant shutdown. The plant ac lighting systems are tested at installation and provisions are provided for testing the dc emergency system. In addition, the staff requires that the battery-powered portable lighting stored on site be inspected and tested. The staff requires that the inservice inspection and testing of the dc emergency lighting system and the portable lighting stored on site be included in the operating procedures and include as a minimum the following:

(1) At least once every 18 months the installed emergency dc lighting packs and the stored onsite portable dc lighting packs shall be verified as operable and shown to provide rated illumination and any other surveillance testing shall be performed as recommended by the manufacturer.

(2) On a periodic basis, as defined by the manufacturer, the capability of the dc lighting packs to perform the design safety function shall be verified and any other surveillance testing shall be performed as recommended by the manufacturer.

The scope of the review of the lighting system for Millstone Unit 3 included assessment of all components necessary to provide adequate lighting during both normal and emergency operating conditions, the adequacy of the power sources for the normal and emergency lighting systems, and verification of functional capability of the lighting system under all conditions of operation.

The basis for acceptance in the staff review was conformance of the design bases, criteria and the design of the lighting systems and necessary auxiliary supporting systems to the acceptance criteria and guidance of SRP Section 9.5.3. Other bases for acceptance were conformance to industry standards, NUREG-0700, and the ability of the systems to provide effective lighting under all conditions of operations.

On the basis of its review, the staff concludes that the various lighting systems provided at Millstone Unit 3 are not in conformance with the above-cited standards, criteria, and design basis, cannot perform their design function, and, therefore, are not acceptable.

Special lighting system requirements for fire protection are addressed in Section 9.5.1 of the SER.

9.5.4 Emergency Diesel Engine Fuel Oil Storage and Transfer System

9.5.4.1 Emergency Diesel Engine Auxiliary Support Systems (General)

There are two emergency diesel generators for each unit at Millstone Unit 3, and each diesel engine has the following auxiliary systems, which are discussed in detail in the SER sections indicated:

(1) fuel oil storage and transfer system (Section 9.5.4.2)
(2) cooling water system (Section 9.5.5)
(3) starting system (Section 9.5.5)
(4) lubrication system (Section 9.5.7)
(5) combustion air intake and exhaust system (Section 9.5.8)

This section of the SER applies to all of the above systems.

With several exceptions, the diesel generator and its auxiliary support systems are housed in a seismic Category I diesel generator building that provides protection from the effects of tornados, tornado missiles, and floods. The exceptions are portions of the diesel generator exhaust stacks, the diesel fuel oil storage tanks, and the fuel oil fill and vent lines. The diesel generator fuel oil storage tanks are housed in an underground seismic Category I concrete vault located adjacent to the diesel generator building that provides protection from the effects of tornados, tornado missiles, and floods; the fuel oil day and storage tank vents are located in enclosures and are protected from the effects of tornados, tornado missiles, and floods caused by natural phenomena. Therefore, the requirements of GDC 2 and 4 with regard to missiles and the recommendations and guidance of RGs 1.115 and 1.117 are met. Protection from the effects of tornados, tornado missiles, and floods is evaluated in Section 3 of this report. Tornado-missile protection of the diesel generator fuel oil fill lines and exhaust stacks is discussed in Sections 9.5.4.2 and 9.5.8, respectively.

The diesel generators and their auxiliary systems for Millstone Unit 3 are independent of the diesel generators at Millstone Units 1 and 2. Thus, the requirements of GDC 5 are met. However, the diesel generators do share the seismic Category I fuel oil storage tanks and one fuel oil transfer pump per fuel oil tank to meet the 7-day fuel oil storage requirement for one diesel generator. There is a cross-tie between the two trains with two normally locked closed valves to facilitate fuel oil transfer between systems. Section 9.5.4.2 of this SER discusses this aspect of the fuel oil storage and transfer system.

The diesel engine and its engine-mounted and separately skid-mounted portions of the auxiliary support systems piping and components normally furnished with the diesel generator package are designed to seismic Category I requirements and follow the guidelines of the Diesel Engine Manufacturers Association (DEMA) standards. The diesel engine and its mounted auxiliary support systems piping and components conform to the requirements of IEEE Std. 387-1977, "Standard Criteria for Diesel-Generator Units Applied as Standby Power Supplies for Nuclear Power Generating Stations," which endorses the DEMA standard and the guidelines of RG 1.9, "Selection, Design and Qualification of Diesel-Generator Units Used as Onsite Electric Power Systems at Nuclear Plants." The diesel engine and its auxiliary support systems meet the quality control requirements of 10 CFR 50, Appendix B. The quality assurance program is evaluated in Section 17 of this report.

Maintenance and engineering personnel responsible for the diesel generators will receive vendor training, which will be incorporated into maintenance department training. New personnel will receive equivalent training. Maintenance on diesel generators will be performed or directly supervised by personnel who have received this training. Ongoing training will include the requalification training program required by 10 CFR 55 for operations personnel, as well as maintenance departmental training for maintenance and engineering personnel, which will be equivalent to the vendor training program.

The preventive maintenance program at Millstone Unit 3 encompasses investigation of components that have a history of repeated malfunctioning and require constant attention and repair. The applicant will also be reviewing operating experiences from other utilities through vendor programs, licensee event reports, and plant incident reports to aid in identifying problems.

On completion of repairs or maintenance and before an actual start, run, and load test, a final equipment check is made to ensure that all electrical circuits are functional (i.e., fuses are in place, switches and circuit breakers are in their proper position, there are no loose wires, all test leads have been removed, and all valves are in the proper position to permit a manual start of the equipment). After the unit has been satisfactorily started and load tested, it is returned to automatic standby service and is under the control of the control room operator.

The applicant has discussed the procedures for no-load and light-load operation of the diesel generator and committed to implement the following procedures before startup:

(1) During extended no-load and light-load operation (less than 20% full load), the diesel generators will be loaded to a minimum of 50% of full load for 1 hour following each 24 hours of continuous no-load or light-load operation.

(2) During periodic testing, the diesel will be loaded to a minimum of 20% of full load as recommended by the manufacturer.

(3) During troubleshooting operations, no-load operation will be minimized. If the troubleshooting operation takes place over an extended period of time (i.e., up to 24 hours), the engine will be cleared by loading the diesel in accordance with Item (1) above.

Experience at some operating plants has shown that diesel engines have failed to start because of accumulation of dust and other deleterious material on electrical equipment associated with starting the diesel generators (e.g., auxiliary relay contacts and control switches). On the basis of meetings with the applicant as well as on preliminary responses to requests for additional information, the staff has determined that the accumulation of dust - excluding dust generated from concrete floors and walls - on the electrical equipment associated with starting of the diesel generators (e.g., auxiliary relay contacts and control switches) is limited by the diesel generator building ventilation system design and operation panel designs, administrative procedures, and weekly cleaning procedures. In a letter dated May 17, 1984, the applicant confirmed this information; therefore, the staff finds that the recommendations of NUREG/CR-0660 with regard to limiting the accumulation of dust on electrical equipment is being met. However, the applicant has not fully alleviated the staff's concern about the accumulation of concrete dust from floors and walls. This is addressed later in this section.

The applicant will perform preoperational and startup tests of the diesel engine auxiliary support systems in accordance with the recommendations and guidelines of RG 1.68, "Initial Test Programs for Water Cooled Reactor Power Plants." The adequacy of the test program is evaluated in Section 14.1 of this report.

The design of the diesel engine auxiliary support systems has not been fully evaluated with respect to the recommendations and guidelines of BTPs ASB 3-1, "Prevention Against Postulated Piping Failures in Fluid System Piping Outside Containment," and MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment." The applicant has not evaluated the failure of the high-energy diesel generator air starting system. Therefore, the systems are not in conformance with GDC 4. Protection against dynamic effects associated with the postulated pipe system failures other than the air starting system is evaluated in Section 3.6 of this report. The high-energy failure analysis of the air starting system is discussed in Section 9.5.6.

The adequacy of the fire protection for the emergency diesel generator and associated auxiliary support systems with respect to the recommendations and guidelines of BTP CMEB 9.5.1, "Guidelines for Fire Protection for Nuclear Power Plants," is evaluated in Section 9.5.1 of this report.

The design of the diesel generator auxiliary support systems also has been evaluated with respect to the recommendations of NUREG/CR-0660, "Enhancement of Onsite Emergency Diesel Generator Reliability." This report made specific recommendations on increasing the reliability of nuclear plant emergency diesel generators. Information requests concerning these recommendations were transmitted to the applicant during the review process. The applicant responded in the amendments to the FSAR stating how the recommendations of NUREG/CR-0660 have been or will be met.

The staff has reviewed these responses and determined that the applicant's conformance to the recommendations is as shown in Table 9.1.

On the basis of its review, the staff has concluded that there is sufficient assurance of diesel generator reliability. However, to ensure long-term reliability of the diesel generator installation, the staff requires that the following issues be resolved and their solutions implemented before initial startup or as stated below.

(1) Moisture in the Air Start System

This is discussed in Section 9.5.6 of this report.

(2) Concrete Dust Control

Even though the applicant as described above has minimized the accumulation of dust and dirt brought into the diesel generator rooms from the outside through the ventilation systems, doorways, and other openings, the applicant has not alleviated the staff concern on concrete dust generated as a result of moving equipment, personnel movement, and other conditions that induce generation of dust from concrete floors and walls. It is the staff's position that, before initial startup, the concrete floor and walls shall be painted with an appropriate paint or treated to minimize the generation of concrete dust. In a letter dated May 17, 1984, the applicant has committed to treat the floor slab with an appropriate sealant to preclude generation of concrete dust. The staff requires that sealant be applied before initial startup.

(3) No-Load and Light-Load Operation

The applicant has stated that each diesel generator unit is capable of operating at its maximum rated output under the following outdoor service conditions and for the durations indicated during the following weather disturbances:

(a) Outdoor Service Conditions

Ambient air intake: -17°F to 102°F
Humidity: maximum 100%

(b) Weather Disturbances

A tornado pressure transient causing an atmospheric pressure reduction of 3 psi in 3 sec followed by a rise to normal pressure in 3 sec; a shorter transient (1.5 sec) will not affect engine operation and output.

A hurricane or northeastern storm pressure of 26 in. of mercury for a duration of 1 hour; the engine is capable of continued operation for up to 14 hours at 26 in. of mercury with no effect on operation and output.

In a meeting in Philadelphia on June 9 and 10, 1982, the diesel engine manufacturer (Colt Industries) stated that no-load and light-load operation of the diesel engines at low ambient temperatures is an unacceptable operating condition for Colt engines. The manufacturer stated that, under these environmental conditions, the diesel engine would fail to operate properly because there would be insufficient turbocharger preheating of the combustion air and potential fuel oil degradation. The diesel engine could fail within a short period of time.

Failure under these environmental conditions could possibly prevent diesel engine restart on a subsequent loss of offsite power. To alleviate this condition, the manufacturer recommends a minimum loading of the engine based on the outside ambient temperatures (e.g., at -10%°F the diesel would have to be loaded to between 60% to 66% of full load to prevent engine failure). This would require the paralleling of the onsite (diesel generator) power source with the offsite power source for extended periods of time. This is unacceptable to the staff and would violate the independence requirement of GDC 17. In letters dated February 1 and 14, 1983, Public Service of New Hampshire, which has similar diesel engines at its Seabrook plant stated that the diesel engine manufacturer had advised them that an air temperature of 50°F or greater at the turbocharger inlet would allow continuous no-load and light-load operation of the diesel generators. Operation with inlet air temperatures below 50°F would require preheating of the combustion air.

On the basis of preliminary information supplied by the applicant, the manufacturer now states that the diesel engines can operate at no-load, light-load, and rated-load conditions with no degradation of the engine's operating characteristics or ability to accept and carry load when operated at stated ambient service conditions. The applicant has provided formal documentation to substantiate the manufacturer's new statement but has not shown that the diesels can accept full load within the required accident load sequence following such operation as stated in Section 8.3.11 of this SER. The staff is pursuing this issue with the applicant.

(4) Vibration of Instruments and Controls

The applicant stated that three control panels are furnished with the diesel engine: (1) the relay and terminal box is mounted at the generator end of the diesel generator skid; (2) an engine gauge board, including pressure switches, is located at the engine end of the skid and mounted on vibration isolation devices; and (3) a diesel generator control panel is mounted on a vibration-free floor area. The applicant also stated that the diesel generator package (including the control panels) is seismically qualified.

Mounting the panels on vibration isolation devices and seismically testing them as part of the diesel generator skid package does not qualify this equipment with the associated controls and monitoring instrumentation for continuous operation under severe vibrational stresses, unless the skid-mounted panels and equipment have been specifically developed, tested, and qualified for these conditions.

The staff requires the applicant to either provide test results and results of analyses that validate that the skid-mounted control panels and mounted equipment have been developed, tested, and qualified for operation under severe vibrational stresses encountered during diesel engine operation or floor mount the control panels currently furnished with the diesel generator separate from the skid on a vibration-free floor area.

In a letter dated May 8, 1984, the applicant has committed to do one of the following:

(a) The engine skid-mounted control panels will be removed from the engine skid and mounted as a freestanding, floor-mounted panel.

(b) Equipment within the panels will undergo vibration qualification during preoperational or qualification testing to confirm that actual equipment vibration is within the tolerances specified as acceptable by the manufacturer.

The staff finds either of the above alternatives acceptable. However, acceptance of alternative (b) is dependent on staff review and approval of test results.

The present diesel generator design meets the requirements of GDC 2 and 4 with regard to tornado- and turbine-missile protection, GDC 5, 17, 18, and 21 and the guidelines of the cited RGs and industry standards, except as noted above. On completion of the above changes and modifications, the design of the diesel generator and its auxiliary systems will also be in conformance with recommendations of NUREG/CR-0660 for enhancement of diesel generator reliability and the related NRC guidelines and criteria. The staff concludes that there is reasonable assurance of diesel generator reliability throughout the design life of the plant.

9.5.4.2 Emergency Diesel Engine Fuel Oil Storage and Transfer System

The design function of the emergency diesel engine fuel oil storage and transfer system is to provide a separate and independent fuel oil supply train for each diesel generator and to permit operation of the diesel generator at ESF load requirements for a minimum of 7 days without replenishment of fuel. The system is designed to meet the requirements of GDC 2, 4, 5, and 17. The meeting of the requirements of GDC 2, 4, and 5 is discussed in Section 9.5.4.1 of this SER.

There are two emergency diesel generators for Millstone Unit 3. Each diesel engine fuel oil storage and transfer system consists of a 550-gal day tank sufficient to power the diesel engine at continuous rated load for approximately 1.5 hours, a 35,000-gal diesel fuel oil storage tank sufficient to power the diesel engine on the basis of the continuous rated load for 3.5 days, two ac motor-driven transfer pumps (one pump powered from the associated diesel generator and the other capable of being powered from either diesel generator), and associated piping, valves, instrumentation, and controls.

Except for the sharing noted in Section 9.5.4.1 between the systems to meet the 7-day fuel oil storage requirements, each diesel engine fuel oil storage and transfer system is independent and physically separated from the other system supplying the redundant diesel generator. Thus, a single failure within any one of the systems will affect only the associated diesel generator. However, because each fuel oil storage tank is sized only for 3.5 days of oil capacity at continuous load and to provide a 7-day fuel oil supply for one diesel generator, the applicant had designed his fuel oil storage and transfer system so that fuel oil can be transferred from either storage tank to both diesel generators. This is accomplished by a cross-tie with two normally locked-closed valves between the two emergency diesel generator fuel oil supply headers and by having one fuel oil transfer pump on each storage tank capable of being powered from either the division A or division B electrical bus. Pump transfer is accomplished by means of a 480-V, seismically qualified, Class 1E, manually operated, administratively controlled transfer switch. Also, the applicant stated that one diesel generator must be shut down to meet the 7-day fuel oil requirements for the other diesel generator. In addition, insufficient information was provided to ensure the staff that the dual-powered fuel oil transfer pumps would not violate the independence criterion of GDC 17 for redundant systems. Thus, the staff found that the design did not meet the requirements of GDC 17 with regard to independence and redundancy and the recommendations of RG 1.137, "Fuel Oil Systems for Standby Diesel Generators," Position C.1; ANSI N195, "Fuel Oil Systems for Standby Diesel Generators," Section 5.2; and SRP Section 9.5.4, Paragraphs I.1.d, and III.6.b, which require 7 days of fuel oil storage for each diesel generator. The staff requested that the applicant justify his design or provide a full 7 days of fuel oil storage for each diesel generator and a detailed description of the operation and procedures for using the fuel oil transfer pumps. In a letter dated May 17, 1984, the applicant described the electrical connections and operations associated with the fuel oil transfer pumps in FSAR Section 8.3.1. With regard to fuel oil storage capacity, the applicant provided the following justification for his design:

> (1) The on-site storage facilities provide a 3 1/2 day fuel supply for each diesel generator assuming both units operate simultaneously at full load. Within 24 hours after notice for fuel has been given, the fuel oil storage tanks can be replenished from geographically diverse off-site fuel suppliers. Such notice will be given within 4 hours of an LOP coincident with a postulated accident per plant procedures. Fuel supplies can be extended to and beyond seven days, as required. In this manner 7 days of fuel can be provided to an operating diesel generator.
>
> (2) Historically the grid supplying the Millstone site has proved very reliable. Since the plants (Millstone 1 and 2) at the site have been operational, offsite power has been restored 95% of the time within 24 hours of first being lost. Consequently, the probability of requiring the diesel generators for any time period exceeding 24 hours is small. Past experience indicates therefore that diesel generator operation in excess of 24 hours is highly unlikely.
>
> A load shedding analysis has been performed which demonstrates that with reduction of loads, the emergency diesel generators will have the capability to be operated continuously for a minimum period of 5 1/2 days with margin that allows slightly over 6 days. In the analysis 8 hours into the worst case accident, which is a DBA coincident with a LOP, the loads may be reduced to approximately 60% of rated capacity on Train A and to approximately 35% on Train B. The load on Train A remains constant where the load on Train B would increase to approximately 40% at 20 hours and fluctuate between 30% and 40% of rated load after 24 hours.

To fully evaluate the load-shedding/fuel capacity analysis, the applicant needs to provide a listing of the loads to be shed or placed onto the bus and when they are to be shed or placed onto the bus. Pending submittal of this information and its acceptance, the staff finds that the design of the system is acceptable and meets GDC 17, RG 1.137, ANSI N195, and SRP Section 9.5.4 with regard to independence, redundancy and fuel oil capacity. However, the staff will require that the following be placed in the

(1) operating procedures

At least once a month, during diesel generator surveillance testing, verify that fuel oil transfer pumps 3EGF*P1A (AO) and 3EGF*P1C(CO) are powered from the "A" Division and 3EGF*P1B(BP) and 3EGF*P1D (DP) are powered from the "B" Division.

(2) plant emergency operating procedures

Within 4 hours of a loss of offsite power (LOP) or a LOP coincident with a design-basis event, fuel oil shall be ordered to be delivered on site within 24 hours.

Therefore, the requirements for GDC 17 as related to the capability of the fuel oil system to meet independence and redundancy criteria are met.

Except for the fuel oil tank fill lines, the flame arrestors, the diesel engine fuel oil storage and transfer system piping and components up to the diesel engine interface (including auxiliary skid-mounted piping) are designed to seismic Category I, ASME Code, Section III, Class 3 (Quality Group C) requirements. They meet the recommendations of RG 1.26, "Quality Group Classifications and Standards for Water-, Steam, and Radioactive Waste Containing Components of Nuclear Power Plants," and RG 1.29, "Seismic Design Classification." The engine-mounted piping and components, from the engine block to the engine interface, are considered part of the engine assembly and are seismically qualified to seismic Category I requirements as part of the diesel engine package. This piping and the associated components - such as valves, fabricated headers, fabricated special fittings, and the like - have not been defined by the applicant. The applicant had been asked to provide the industry standards to which the engine-mounted piping and components are designed.

Each fuel storage tank is filled from a single fuel oil supply line. In addition, the exposed portions of the diesel oil storage tank fill line external to the diesel generator building are not tornado missile protected. The applicant, in a letter dated April 6, 1984, stated that in the event of damage to the fill connections as a result of tornado missiles, floods, and/or seismic events, the fuel oil storage tanks can be filled through a manhole in the top of the tank or through the vent connection. In addition, the fuel oil fill lines, the flame arrestors, piping, and the associated valves and components are designed, manufactured, and inspected in accordance with the guidelines and requirements of ANSI B31.1, "Code for Pressure Piping"; ANSI N45.2, "Quality Assurance Program Requirements for Nuclear Facilities"; and 10 CFR 50, Appendix B. The fuel oil fill line piping and associated components are intentionally overdesigned (subjected to low working stresses for the application), thereby resulting in high operational reliability. The design of the fuel oil fill line piping and components to the cited design philosophy and standards is considered equivalent to a system designed to ASME Code, Section III, Class 3 requirements with regard to system functional operability and inservice reliability.

The design of the emergency diesel engine fuel oil storage and transfer system conforms to ANSI N195, except as stated above. In addition, the fuel oil quality and tests will conform with the guidelines of RG 1.137 (Positions C.2.a through C.2.f) and the requirements will be included in the plant Technical Specifications. Position C.2.g of RG 1.137 (sediment control during refilling) has not been fully addressed. In a letter dated April 6, 1984, the applicant addressed sediment control during refilling operations in the following manner:

> Fuel oil degradation due to the turbulence of sediment in the bottom of the fuel oil storage tank during the addition of new fuel oil is minimized by the following:
>
> (1) Normal fill line strainer (0.10 inch perforation size).
>
> (2) Fuel oil transfer pump discharge strainer (0.062 inch perforation size). The strainer is provided with a pressure differential indicating switch and alarm which activates a high differential pressure alarm on a local panel and a local panel trouble alarm on the main board. If a high pressure differential exists that prevents sufficient fuel oil flow to the day tanks, the redundant fuel oil transfer pump will be automatically started on low-low day tank level.
>
> (3) Engine-mounted duplex fuel oil filter (.00012 to .00020 inch). The filter is provided with a pressure differential indicating switch which activates a high pressure alarm on a local panel and a local panel trouble alarm on the main board. These filters will be frequently monitored, and filter cartridges replaced when necessary.
>
> In addition, the fill line for each fuel oil storage tank is located a sufficient distance from the fuel oil transfer pump to enhance settling of sediment away from the pump suction.

The staff finds the justification unacceptable for the following reasons:

(1) The fill line strainer will not prevent the sediment in the fuel oil storage tank from being stirred up or remove the sediment in the tank during refilling operations.

(2) The distance between the tank fill line and pump suction to allow for sediment settling is also not valid. Once the sediment has been stirred up, it will remain in suspension for several hours. In addition, the concentration of sediment in the fuel oil during refilling is higher at the beginning than at the end of the filling operation, but the amount or quantity of sediment in suspension is relatively the same. Therefore, if the transfer pump is taking suction at the same time as the tank is being filled, large quantities of sediment will pass through the system which could cause damage to components, quickly clog filters and strainers, and result in the unavailability of the engines.

(3) Taking into consideration the strainer and filter clogging resulting from sediment, single failures of pumps, and operator actions, credit cannot be given to the duplex filters and pump strainers.

However, the design of the system as described above allows the diesel generator day tank to be filled from either fuel oil storage tank. Thus, fuel oil can be drawn from one fuel oil storage tank while the other tank is being filled and then allowed to stand for 24 hours while the sediment settles out. This is an acceptable procedure for sediment control in the fuel oil system. Therefore, the staff requires that the plant operating procedures be modified to incorporate this filling procedure.

The scope of review of the diesel engine fuel oil storage and transfer system included layout drawings, piping and instrumentation diagrams, and descriptive information in FSAR Section 9.5.4 for the system and auxiliary support systems essential to its operation.

The basis for acceptance in the staff review was conformance of the design criteria and bases and design of the diesel engine fuel oil storage and transfer system to the requirements of GDC 17 with respect to redundancy and physical independence, to the guidance of the cited RGs, to the recommendations of NUREG/CR-0660, and to industry codes and standards.

On the basis of its review, the staff concludes that the emergency diesel engine fuel oil storage and transfer system meets, except as noted, the requirements of GDC 2, 5, and 17 and meets the recommendations of NUREG/CR-0660, the guidance of the cited RGs and SRP Section 9.5.4, and industry codes and standards. It does not meet the requirements of GDC 4 as noted in Section 9.5.4.1 and this section; thus, it cannot perform its design safety function and, therefore, is unacceptable. On receipt of the required additional information, the staff will report its findings in a supplement to this SER.

9.5.5 Emergency Diesel Engine Cooling Water System

The design function of the emergency diesel engine cooling water system is to maintain the temperature of the diesel engine within a safe operating range under all load conditions and to maintain the engine coolant preheated during standby conditions to improve starting reliability. The system is designed to meet the requirements of GDC 2, 4, 5, 17, 44, 45, and 46. Conformance with requirements of GDC 2, 4, and 5 is discussed in Section 9.5.4.1 of this SER.

The emergency diesel engine cooling water system is a closed-loop system and cools the engine jacket, lube oil cooler, governor lube oil cooler, fuel oil injectors, turbocharger and air coolers, and generator bearing. The emergency diesel engine cooling water system is composed of two subsystems: the jacket water system and the intercooler water system. The major components of the jacket water system for each standby emergency diesel engine include an engine-driven jacket coolant water pump, jacket water heat exchanger, an expansion tank (shared by both subsystems), two turbochargers, a governor lube oil cooler, a lube oil cooler, a motor-driven jacket coolant standby circulation pump, an electric heater, and thermostatic three-way valves, as well as the required instrumentation, controls, and alarms, and the associated piping and valves to connect the equipment. The major components of the intercooler water system for each standby emergency diesel engine include an engine-driven intercooler water pump, an intercooler water heat exchanger, air coolers, and thermostatic three-way valves as well as the required instrumentation, controls, and alarms, and the associated piping and valves to connect the equipment, provide cooling to the fuel injectors and generator bearings, and connect the system with the jacket water system. When the diesel engines are operating, the heat generated is rejected to the service water system by means of the cooling water heat exchangers.

During operation of the standby diesel engine, the temperature of the diesel engine coolant is regulated automatically through the action of temperature-sensing three-way thermostatic valves. When the standby diesel engine is idle, the engine coolant is heated by an electric heater and continuously circulated through the engine. The temperature is controlled by a thermostat to keep the engine warm and ready to accept loads within the prescribed time interval.

The diesel generator is capable of operating fully loaded without secondary cooling for approximately 3 min. The engine and expansion tank contain enough water to absorb the heat generated during this period. This time interval is greater than the time needed to restore essential service water to the diesels in the event of a loss of offsite power. Alarms have been provided to enable the control room operator to monitor the diesel generator cooling while the unit is in the standby mode or in operation.

There are two emergency diesel generators for Millstone Unit 3, and each diesel generator has a physically separate independent cooling water system. Therefore, the requirements of GDC 17 and 44 as related to redundancy and the single-failure criterion are met.

The diesel engine cooling water system piping and components up to the diesel engine interface, including auxiliary skid-mounted piping, are designed to seismic Category I, ASME Code, Section III, Class 3 (Quality Group C) requirements and meet the recommendations of RGs 1.26 and 1.29. The engine-mounted piping and components, from the engine block to the engine interface, are considered part of the engine assembly and are seismically qualified to seismic Category I requirements as part of the diesel engine package. This piping and the associated components - such as valves, fabricated headers, and fabricated special fittings - have not been defined by the applicant. The applicant has been request to provide the industry standards to which the engine-mounted piping and components are designed.

The diesel engine cooling water system conforms with RG 1.9 (Position C.7) as it relates to engine cooling water protective interlocks. The diesel generator system protective interlocks are discussed in Section 8.3 of this report.

The diesel engine cooling water system has provisions to permit periodic inspection and functional testing during standby and normal modes of power plant operation as required by GDC 45 and 46.

The scope of the review of the emergency diesel engine cooling water system included layout drawings, piping and instrumentation diagrams, and descriptive information in FSAR Section 9.5.5 for the system and the auxiliary support systems essential to its operation.

The bases for the acceptance in the staff review were (1) conformance of the design criteria and bases and design of the diesel engine cooling water system to GDC 17 and 44 with respect to redundancy and physical independence, to

GDC 45 and 46 with respect to inspection and testability of the system, to the guidance of the cited regulatory guides, and to the recommendations of NUREG/CR-0660 and industry codes and standards and (2) the ability of the system to maintain stable diesel engine cooling water temperature under all load conditions.

On the basis of its review, the staff concludes that, except as noted, the emergency diesel engine cooling water system meets the requirements of GDC 2, 5, 17, 44, 45, and 46; meets the guidance of the cited regulatory guides and SRP Section 9.5.5; and meets the recommendations of NUREG/CR-0660 and industry codes and standards. It does not meet the requirements of GDC 4 as noted in Section 9.5.4.1; thus, it cannot perform its design function and is, therefore, unacceptable. After receipt of the requested additional information, the staff will report its findings in a supplement to this SER.

9.5.6 Emergency Diesel Engine Starting System

The design function of the emergency diesel engine starting system is to provide a reliable method for automatically starting each diesel generator so that the rated frequency and voltage are achieved and the unit is ready to accept required loads within 10 sec. The system is designed to meet the requirements of GDC 2, 4, 5, and 17. The meeting of the requirements of GDC 2, 4, and 5 is discussed in Section 9.5.4.1 of this SER.

There are two emergency diesel generators for Millstone Unit 3. Each diesel generator has an independent and redundant air starting system consisting of two separate full-capacity air starting subsystems, each with sufficient air capacity to provide a minimum of five consecutive cold engine starts from the low-low pressure alarm setpoint. Redundancy in the starting system is provided by two emergency diesel generators so that a malfunction or failure in one system does not impair the ability of the other system to start its diesel engine. This meets the requirements of GDC 17.

The air starting system for each diesel generator includes two air compressors, two receiver tanks, intake air filters, injection lines and valves, air-to-cylinder control and starting valves, a control air tank, instrumentation, controls, alarms, and the associated piping to connect the equipment. Alarms annunciate on the local panel and in the main control room to enable the operator to monitor the air pressure of the diesel generator starting air system.

Automatic controls are provided to automatically start and stop each air compressor when the pressure in its respective air receivers decreases or increases to predetermined levels.

The diesel engine air starting system piping and components from the air compressor up to the diesel engine interface, including auxiliary skid-mounted piping, are designed to seismic Category I, ASME Code, Section III, Class 3 (Quality Group C) requirements and meet the recommendations of RGs 1.26 and 1.29. The engine-mounted piping and components, from the engine block to the engine interface, are considered part of the engine assembly and are seismically qualified to seismic Category I requirements as part of the diesel engine package. This piping and the associated components - such as valves, fabricated headers, and fabricated special fittings - have not been defined by the applicant.

The diesel generator air starting system is a high-energy system (design pressure 450 psig). The applicant has not provided a high-energy-line-break analysis for the air starting system. He had been asked to provide such an analysis. In addition, commensurate with the safety function performed by the system, the staff requires the following:

(1) All air starting engine-mounted piping and components that are pressurized to high-energy pressures (275 psig or greater) during standby, starting, and/or operation will be designed to seismic Category I, ASME Code, Section III, Class 3 (Quality Group C) requirements.

(2) All high-energy air starting piping will be adequately restrained to prevent damage to other diesel generator piping, components, and equipment from pipe whip. Note: Seismic restraints and seismic supports may not be adequate as pipe whip restraints.

The diesel generator air starting system conforms with RG 1.9 (Position C.7) as it relates to diesel engine air starting system protective interlocks. The diesel generator system protective interlocks are discussed in Section 8.3 of this report.

The present air starting system design does not include air dryers as required by NUREG/CR-0660. In a letter dated May 8, 1984, the applicant committed to install a desiccant-type air dryer plus compressor aftercoolers before the end of the first refueling outage. The applicant stated that estimates show that the material package and engineering will not be completed until December 1984 to support air dryer installation. System modifications initiated at that time would impact startup and testing activities that will occur after system turnover, scheduled for July 1984, as well as other systems dependent on diesel generator operability. Therefore, unless scheduling delays occur, air dryer installation is not anticipated to occur until the first refueling outage. The applicant also stated that system design and operating procedures would minimize water and contaminants in the system during this period, and that the system design characteristics and operating procedures that would allow this delay are:

(1) Air receivers will be blown down each day, once per shift.

(2) The starting air system includes an in-line strainer between the starting air receivers and the air start solenoid valves capable of removing particles greater than 1/32" diameter. The air start solenoids are Circle Seal solenoids with stainless steel bodies and oil resistant seats. They require only a small quantity of low velocity air to function. They can operate even partly plugged and would still permit the air start valves to open.

(3) The smallest passages of the air start valves are well in excess of 1/32". The valves are considerably different than those found on Colt-Fairbanks Morse Type OP engines. This improved type of air start valves is less succeptible to malfunction due to debris in the air being admitted to it.

(4) Debris accumulation downstream of the air start valve but before the air start distributor is precluded by a 80 micron in-line air filter.

(5) The engine contains two completely parallel and independent air supply path from each compressor, air receiver tank,and solenoid start valve, to two fully redundant air start distributors. In the event small debris in either separate receiver tank fouled an air start line, the other redundant air start path would be available for engine starts.

(6) Debris accumulation during the time frame when the air start system is put into service until the first scheduled refueling outage occurs should be minimal. It is minimized by both frequent blowdown and in-line filters, which will be changed out periodically.

The staff had evaluated the justification provided by the applicant for deferral of air dryer installation to the first refueling. Installation of the air dryers before fuel loading could possibly cause a substantial delay in the plant's scheduled startup and testing activities. Therefore, the staff accepts the applicant's justification for delaying the air dryer installation until the first refueling with the following license conditions:

(1) The air dryers shall be installed at the first opportunity but no later than before startup of the first refueling outage.

(2) To ensure diesel generator air starting availability in accordance with GDC 17,

(a) The air receivers shall be blown down at least once per shift.

(b) The in-line filters shall be inspected and changed if necessary at least once (______).*

License conditions (2)(a) and (2)(b) will be removed once the air dryers have been installed.

The air starting system also provides control air to the diesel engine fuel rack, cooling water system three-way valves, and for other functions. Upon loss of air pressure, the controls, valves, and the like go to the fail-safe position and do not degrade or prevent the engine from operating and carrying load.

Operating experience at two nuclear power plants has shown that during periodic surveillance testing of a standby diesel generator, initiation of an emergency start signal (LOCA or LOP) resulted in the failure of the diesel to start and perform its function because of depletion of the starting air supply from repeated activation of the starting relay. This event occurred as the result of inadequate procedures and from a failure in engine starting and control circuit

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*Frequency to be supplied by applicant or as recommended by manufacturer.

logic to address a built-in time delay relay to ensure the engine comes to a complete stop before attempting a restart. During the period that the relay was timing out, fuel to the engine was blocked while the starting air was uninhibited. This condition with repeated start attempts depleted starting air and rendered the diesel generator unavailable until the air system could be repressurized. This is an unacceptable operating condition. The applicant was asked to review his procedures and/or control system logic to ensure that this event will not occur at Millstone Unit 3. Because this request was issued late in the review process, he has not had time to fully evaluate his system. The staff will pursue this item with the applicant.

The scope of review of the emergency diesel engine starting system included layout drawings, piping and instrumentation diagrams, and descriptive information in FSAR Section 9.5.6 for the system and auxiliary support systems essential to its operation.

The bases for acceptance in the staff review were (1) conformance of the design criteria and bases and design of the diesel engine air starting system to the recommendations of NUREG/CR-0660 and industry codes and standards and (2) the ability of the system to start the diesel generator within a specified time period.

On the basis of its review, the staff concludes that, except as noted, the emergency diesel engine air starting system meets the requirements of GDC 2, 5, and 17 and meets the guidance of the cited RGs and SRP Section 9.5.6, the recommendations of NUREG/CR-0660, and industry codes and standards. The system does not meet the requirements of GDC 4 as noted above and in Section 9.5.4.1; thus, it cannot perform its design safety function and is, therefore, unacceptable. After receipt of the requested additional information, the staff will report its findings in a supplement to this SER.

9.5.7 Emergency Diesel Engine Lubricating Oil System

The design safety function of the emergency diesel engine lubricating oil system is to provide a supply of filtered lubrication oil to the various moving parts of the diesel engine including piston and bearings. The system is designed to meet the requirements of GDC 2, 4, 5, and 17. The meeting of the requirements of GDC 2, 4, and 5 is discussed in Section 9.5.4.1 of this SER.

Major components of the emergency diesel engine lubricating oil system include an engine-driven lube oil pump; an engine-driven rocker arm lube oil pump; a moisture detection system; a motor-driven prelube and filter pump; a motor-driven rocker arm prelube pump; a lube oil collection sump, strainers, and filters; a lube oil cooler; an electric heater and thermostatic three-way valve; instrumentation, controls, and alarms; and associated piping and valves to connect the equipment. The diesel engine is equipped with relief ports, as well as a crankcase exhaust system composed of a crankcase vacuum pump, oil separator, and the necessary piping to the outside of the diesel generator building. This system provides protection from crankcase explosion. Alarms and protective devices are provided to alert the control room operator to abnormal conditions in the diesel generator lubrication oil systems during standby, startup, or operating status.

The emergency diesel engine lubrication oil system is an integral part of the diesel engine and thus meets the requirements of GDC 17 with regard to system independence and the single-failure criterion. The engine heat is rejected to the diesel engine jacket water system.

The diesel engine lubrication system is composed of two subsystems: the main engine lube oil system and the rocker arm lube oil system. During engine operation or when the engine is on standby, the main engine lube oil system supplies oil to all main bearings, the camshaft bearings, cam followers, engine wearing parts, and turbocharger. The prelubrication portion of this system is operated when the engine is on standby, at which time the lube oil is heated by an electric heater and circulated through the engine continuously by an ac motor-driven pump to improve the first-try starting reliability.

During engine operation or when the engine is on standby, the rocker arm lube oil system supplies oil to the rocker arm assemblies. The prelubrication portion of this system is operated manually once a week for 5 min as recommended by the manufacturer. The applicant has committed to incorporate the manufacturer's recommendation into the operating procedures. The systems have alarms to indicate heater failure.

FSAR Section 9.5.7.1 states that the temperature of the lubricating oil is automatically maintained above a minimum value by means of an independent recirculation loop, including its own pump and heater, to enhance first-try starting reliability of the engine in the standby condition. The rocker arm lubrication system is an independent subsystem of the diesel lubrication oil system, which is connected to the main system by a float valve in the rocker arm oil reservoir; thus, the lube oil in the rocker arm lubrication system will never be preheated unless the oil level is low enough to open the float valve. The applicant, in a letter dated April 6, 1984, provided adequate justification as to why the rocker arm lubrication does not need to be preheated. He stated:

> The SAE 30 lubricating oil in the rocker arm lubrication system has a pour point of -5°F. The oil is heated by conduction from the standby jacket coolant heating system which has a minimum temperature of jacket coolant heating system which has a minimum temperature of 95°F. This will maintain the operability of the rocker arm lubrication system when room temperatures are within expected ranges. If a failure of either emergency generator enclosure heating system occurs, a low room temperature alarm actuates at 45°F on Ventilation Panel 1 in the control room. In response to this alarm, operator corrective action would be taken. Actions that may be taken include:
>
> (1) bringing in portable space heaters
>
> (2) increasing room temperature by turning on lights or equipment
>
> (3) starting the emergency diesel generator

The staff finds this acceptable; however, it will require that the following be included in the plant procedures to ensure diesel generator availability: Upon actuation of the diesel generator low room temperature alarm, the room air temperature shall be increased to 50°F or greater, or this may result in diesel generator being placed in a limiting condition for operation.

Except for the moisture detector system, the lube oil preheat and diesel engine lubrication oil system piping and components up to the diesel engine interface, including auxiliary skid-mounted piping, are designed to seismic Category I, ASME Code, Section III, Class 3 (Quality Group C) requirements and meet the recommendations of RGs 1.26 and 1.29. The crankcase exhaust system up to the diesel engine interface; the moisture detector system between the locked-closed ASME Code, Section III, Class 3 valves; the preheat and prelube system from the sump up to the ASME Code, Section III, Class 3 valves that isolate the lube oil filter; and the rocker arm lube oil system from the lube oil header up to the diesel engine interface are designed to seismic Category I requirements. These systems are designed, manufactured, and inspected in accordance with the guidelines and requirements of ANSI B31.1 and ANSI N45.2 and 10 CFR 50, Appendix B. These systems' piping and associated components are intentionally overdesigned (subject to low working stresses) for the application, thereby resulting in high operational reliability. The design of the systems' piping and components to the cited design philosophy and standards is considered equivalent to a system designed to ASME Code, Section III, Class 3 requirements with regard to system functional operability and inservice reliability.

The engine-mounted piping and components, from the engine block to the engine interface, are considered part of the engine assembly and are seismically qualified to seismic Category I requirements as part of the diesel engine package. This piping and associated components - such as valves, fabricated headers, and fabricated special fittings - have not been defined by the applicant. The applicant has been requested to provide the industry standards to which the engine-mounted piping and components are designed.

The diesel generator lubricating oil system conforms with RG 1.9 (Position C.7) as it relates to diesel engine lubrication system protective interlocks. The diesel generator system protective interlocks are discussed in Section 8.3 of this report.

A moisture detection subsystem is provided to detect the presence of water in the lube oil system. This system consists of a circulating pump, strainer, moisture detector, piping, and valves. The system will be isolated by locked-closed ASME Code, Section III isolation valves, and will only be used during the monthly diesel engine testing. On detection of moisture, the detector will annunciate a local panel trouble alarm on the main control board and actuate an alarm on the local board located in the emergency generator enclosure.

FSAR Section 9.5.7.2 states that a 1,200-gal capacity lubricating oil sump is provided to supply the engine with an adequate amount of lubricating oil during engine operation. The minimum recommended sump level of approximately 1,000 gal would be reached after 5 days of operation at full-rated load with a normal oil usage rate of 40 gpd. This low level is alarmed in the control room to alert the operators. Once this minimum level is reached, oil will be added to the system without an engine shutdown. Adequate lubricating oil is stored on site to ensure 7 days of operation at rated load. The staff is currently incorporating into the Standard Technical Specifications surveillance requirements to ensure 7 days' supply of lube oil on site at all times. In the interim the staff will require that the following be included as plant-specific Technical Specifications:

(1) Section 3.8.1.1 of the Technical Specifications:

(a) Lubricating oil storage should contain a minimum total volume of 280 gal of lubricating oil per engine.

(b) The capability to transfer lubricating oil from storage to the diesel generator unit for both standby and operating modes should be demonstrated.

(2) Section 4.8.1.1.2.a of the Technical Specifications:

(a) The lubricating oil inventory in storage should be verified.

FSAR Section 9.5.7.5 discusses the level alarms associated with the lubricating oil system. It is stated that the rocker arm lubricating oil reservoir level is monitored for high level and the level is maintained by a level control valve. No mention is made of a reservoir low-level alarm. A failure of the level control valve to maintain lubricating oil level in the rocker arm reservoir could result in inadequate or no lubricating oil for the rocker arms, leading to diesel generator unavailability and/or failure. This is an unacceptable condition. The applicant was requested to justify his design or to provide a low-level alarm for the rocker arm lubricating oil reservoir. He is still evaluating his design.

The scope of review of the diesel generator lubricating oil system included piping and instrumentation diagrams and descriptive information in FSAR Sections 9.5.7 and 9.5.8 for the system and auxiliary support systems essential to its operation.

The basis for acceptance in the staff review was conformance of the design criteria and bases and design of the diesel engine lubricating oil system to the requirements of GDC 17 with respect to redundancy and physical independence, to the guidance and additional acceptance criteria of SRP Section 9.5.7, to the recommendations of NUREG/CR-0660, and to industry codes and standards.

On the basis of its review, the staff concludes that, except as noted, the emergency diesel engine lubricating oil system meets the requirements of GDC 2, 5, and 17, the guidance of the cited RGs and SRP Section 9.5.7, and the recommendations of NUREG/CR-0660 and industry codes and standards. It does not meet the requirements of GDC 4, as noted in Section 9.5.4.1; thus, it cannot perform its design safety function and is, therefore, unacceptable. After receipt of the requested additional information, the staff will report its findings in a supplement to this SER.

9.5.8 Emergency Diesel Engine Combustion Air Intake and Exhaust System

The design function of the emergency diesel engine combustion air intake and exhaust system is to supply filtered air for combustion to the engine and to dispose of the engine exhaust to the atmosphere. The system is designed to meet the requirements of GDC 2, 4, 5, and 17. The meeting of the requirements of GDC 2, 4, and 5 is discussed in Section 9.5.4.1 of this SER.

A separate source of combustion air for each diesel engine is taken from the associated diesel generator air intake structure through an air filter, turbocharger compressor, and combustion air coolers. The path of the exhaust gas discharge is through the turbocharger, exhaust silencer, and exhaust ducting to the outside of the building. This meets the requirements of GDC 17 with regard to system independence, redundancy, and the single-failure criterion.

The location of the air intake structures and design precludes the intake of fire-extinguishing agents and other noxious gases (except exhaust gases discussed below) and dust and other deleterious materials that would affect diesel generator operation.

The diesel generator combustion air intake and exhaust system conforms with RG 1.9 (Position C.7) as it relates to diesel engine combustion air intake and exhaust system protective interlocks. The diesel generator system protective interlocks are discussed in Section 8.3 of this report.

The diesel engine combustion air intake and exhaust system piping and components up to the diesel engine interface are designed to seismic Category I requirements and meet the recommendations of RG 1.29. The engine-mounted piping and components, from the engine block to the engine interface, are considered part of the engine assembly and are seismically qualified to seismic Category I requirements as part of the diesel engine package.

The combustion air intake and exhaust piping and associated components - such as fabricated headers and fabricated special fittings - to the engine interface are designed, to ASME Code, Section III, Class 3 (Quality Group C) requirements and meet the recommendations of RG 1.26. The engine-mounted combustion air intake and exhaust piping and the associated components have not been defined by the applicant. The applicant has been requested to provide the industry standards to which the engine-mounted piping and components are designed.

The applicant states that the diesel generator exhaust pipes located outside the diesel generator building will be exposed to tornado missiles and have not been designed to withstand these missiles. Damage to the diesel exhaust pipes from a tornado missile could result in deformation of or severing of the exhaust pipe. The severing of the exhaust pipe could not affect the operation of the diesel generator. Deformation of the exhaust pipe could result in a decrease in the operational performance of the corresponding diesel generator. To alleviate this problem, the applicant provided an access hatch in the exhaust ductwork, which would be manually opened during tornado alerts and function as an exhaust bypass (secondary exhaust path) in the event of tornado-missile damage to the exhaust system. This is considered an acceptable design. However, FSAR Figure 9.5.4 shows the location of the exhaust access hatch in relation to the exhaust ductwork and the exhaust plenum openings. From the drawings, a tornado missile, entering either the 66-in. x 100-in. or the 48-in. x 168-in. plenum openings, could cause sufficient damage to exhaust ductwork and the access hatch to degrade diesel generator operation or result in the unavailability of the diesel generators. Therefore, the staff required that the exposed portions of the exhaust stacks be tornado missile protected. In addition, in the event of freezing rain, ice storms, and/or a snow storm, because of the location of the openings, clogging or total blockage or the freezing-shut of the access hatch could result. The staff asked the applicant to describe the design feature, inservice inspection procedures, and Technical Specifications that would preclude this event from occurring or alter the design.

In information supplied, the applicant stated that damage to the access hatch from tornado missiles is not credible. No viable missile trajectory targets the access hatch. Once the access hatch is open, exposed portions of the exhaust ductwork are not required. Damage to these exposed parts will not degrade the exhaust. Sufficient information on tornado-missile trajectories has not been provided to substantiate the applicant's justification. On receipt of this information, the staff will report on the adequacy of the tornado-missile protection for the emergency diesel generator exhaust stacks and its conformance to the requirements of GDC 2 in Section 3.5.2 of a supplement to this SER.

The applicant has also provided information on the clogging, total blockage, and/or freezing-shut of the access hatch resulting from freezing rain, ice storms, and/or snow storms. He has stated that plant personnel will be instructed in procedures for opening the access hatch in the event that there is an ice or snow buildup in the exhaust plenum or on the access hatch.

The staff finds the intent of the proposed procedure acceptable; however, the procedure calls for the hatch to be opened after a buildup of ice or snow. Under this procedure the hatch could already be blocked and/or frozen shut before any action is taken. Therefore, the staff will require that the following be included in the plant Technical Specifications in order that the requirements of GDC 2 and 4 will be met:

(1) In the event of (a tornado alert, or)* an ice storm, snow storm or freezing rain storm forecast, the access hatch in the emergency diesel generator combustion exhaust system shall be opened and shall remain open until the event has passed.

(2) At least once a year, the access hatch shall be opened to verify operation of the hatch, inspected for corrosion of parts (hinges, locking mechanisms, etc.) and maintained in operable status by replacement of corroded parts, properly lubricated, pointed,etc.

The scope of review of the diesel generator intake and exhaust system included layout drawings, piping and instrumentation diagrams, and descriptive information in FSAR Section 9.5.8 for the system and auxiliary support systems essential to its operation.

The bases for the acceptance in the staff review were (1) conformance of the design criteria and design of the diesel engine air intake and exhaust system to GDC 17 with respect to redundancy and physical independence, to the guidance of the cited RGs, to the guidance and additional acceptance criteria of SRP Section 9.5.8, to the recommendations of NUREG/CR-0660, and to industry codes and standards and (2) the ability of the system to provide sufficient combustion air and release of exhaust gases to enable the emergency diesel generator to perform on demand.

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*If the applicant can substantiate that the access hatch and the adjacent diesel combustion exhaust ductwork are adequately protected from tornado missiles, then "a tornado alert, or" shall be included in the Technical Specifications as shown. If the applicant cannot substantiate the tornado-missile protection for the access hatch, then adequate protection shall be provided for the diesel generator exhaust system.

On the basis of its review, the staff concludes that, except as noted above, the emergency diesel engine intake and exhaust system meets the requirements of GDC 5 and 17 and NUREG/CR-0660, the guidance of the cited RGs and SRP Section 9.5.8, and industry codes and standards. It does not meet the requirements of GDC 2 and 4 as noted above and in Section 9.5.4.1; thus, it cannot perform its design safety function and is, therefore, unacceptable. On receipt of the requested additional information, the staff will report its findings in a supplement to this SER.

Table 9.1 Conformance to NUREG/CR-0660

Recommendation	Conformance	SER section
1. Moisture in air starting system	Partial	9.5.6
2. Dust and dirt in diesel generator room	Yes	9.5.4.1
3. Turbocharger gear drive problem	N/A	
4. Personnel training	Yes	9.5.4.1
5. Automatic prelube	Yes	9.5.7
6. Testing, test loading, and preventive maintenance	Partial	9.5.4.1
7. Improve identification of root cause of failures	Yes	9.5.4.1
8. Diesel generator ventilation and combustion air systems	Yes	9.5.8
9. Fuel storage and handling	Yes	9.5.4.2
10. High temperature insulation	*	9.5.4.1
11. Engine cooling water	Yes	9.5.5
12. Concrete dust control	Partial	9.5.4.1
13. Vibration of instruments	Partial	9.5.4.1

*Explicit conformance is considered unnecessary by the staff in view of the equivalent provided by the design, margin, and qualification testing requirements that are normally applied to emergency standby diesel generators.

Note: N/A = not applicable.

10 STEAM AND POWER CONVERSION SYSTEM

10.1 Summary Description

The steam and power conversion system is designed to remove heat energy from the primary reactor coolant loop via four steam generators and to generate electric power in the turbine generator. After the steam passes through the high- and low-pressure turbines, the main condensers deaerate the condensate and transfer the rejected heat to the open-cycle circulating water system, which uses the Long Island Sound to dissipate the rejected heat. The condensate is reheated and returned as feedwater to the steam generator. The entire system is designed for the maximum expected energy from the nuclear steam supply system. As noted in the following parts of this section, the staff safety evaluation was performed in accordance with the applicable requirements of the SRP.

A turbine steam dump (bypass) system is provided to discharge directly to the condenser up to 40% of the main steam flow around the turbine during transient conditions. This bypass capacity, together with a 10% reactor automatic step-load reduction capability, is sufficient to withstand a 50% generator load loss without tripping the turbine, causing control rod movement, or tripping the reactor.

10.2 Turbine Generator

The turbine generator converts steam power into electrical power and has a turbine control and overspeed protection system. The design function of the turbine control and overspeed protection system is to control turbine action under all normal or abnormal conditions, to ensure that a full load turbine trip will not cause the turbine to overspeed beyond acceptable limits, and to minimize the probability of generation of turbine missiles in accordance with the requirements of GDC 4, "Environmental and Missile Design Bases." The turbine control and overspeed protection system is, therefore, essential for the overall safe operation of the plant.

The turbine generator is manufactured by the General Electric Company and is a tandem-compound type (single shaft) with one double-flow high-pressure turbine and three double-flow low-pressure turbines. It has a rotational speed of 1,800 rpm and is designed for a gross generator output of 1,261 MWe at a nominal plant exhaust pressure of 1.5 in. mercury (absolute).

The turbine generator is equipped with a digital electrohydraulic control (EHC) system. The EHC system consists of an electronic governor using solid-state control techniques in combination with a high-pressure hydraulic actuating system. The system includes electrical control circuits for steam pressure control, speed control, load control, and steam control valve positioning.

Overspeed protection is acccomplished by three independent systems, that is, normal speed governor, mechanical overspeed, and electric backup overspeed control systems. The normal speed governor modulates the turbine control valves to maintain desired speed load characteristics and will start to close the intercept valves and control valves at 101% of rated speed. The mechanical overspeed sensor trips the turbine stop, control, and combined intermediate valves by deenergizing the hydraulic fluid systems when 110% of rated speed is reached. After an overspeed condition is detected, the turbine steam valves' closure times are as follows: main steam stop valve - 0.2 sec, main steam control valve - 0.2 sec, reheat stop valve - 0.2 sec, reheat intercept valve - 0.2 sec, and extraction steam valves - less than 2 sec. After an overspeed condition is detected, these valves are designed to fail closed on loss of hydraulic system pressures. The electrical backup overspeed sensor will trip these same valves when 111% of rated speed is reached by independently deenergizing the hydraulic fluid system. Both of these actions independently trip the energizing trip fluid system. The overspeed trip systems can be tested while the unit is on line. Therefore, the requirements of GDC 4 are met.

To protect the turbine generator, the following signals will shut down the turbine: (1) manual emergency trip, (2) low bearing lubricating oil pressure, (3) low condenser vacuum, (4) high steam generator water level, (5) loss of stator cooling, (6) main shaft-driven lube oil pump low discharge pressure (above 75% speed), (7) low hydraulic fluid pressure, (8) high exhaust hood temperature, (9) generator electric trip, (10) reactor trip, (11) excessive thrust bearing wear, (12) turbine overspeed at 110% of rated speed, (13) turbine overspeed at 111% of rated speed, (14) high water level moisture separator, (15) safety injection signal, (16) loss of both primary and backup speed feedback signals (when the EHC speed control is in automatic mode), (17) loss of EHC 125-Vdc power supply (below 75% speed), and (18) low emergency trip system fluid pressure.

An inservice inspection program for the main steam stop and control valves and reheat valves is provided and includes (1) dismantling and inspection of one of each type of turbine steam valves, at approximately 3 1/3-year intervals, during refueling or maintenance shutdowns coinciding with the inservice inspection schedule and (2) exercising and observing the main steam stop and control, reheat stop, and intercept valves at least once a week. This will be included in the plant Technical Specifications. The applicant is also providing a quarterly inservice inspection program for the extraction steam valves. The inspection will ensure that the extraction check valve closing mechanism travels in the closing direction in a free and positive manner.

The applicant will include preoperational and startup tests of the turbine generator in accordance with RG 1.68, "Initial Test Programs for Water Cooled Power Plants." The adequacy of the test program is evaluated in Section 14 of this SER.

The turbine generator system meets the recommendations of BTPs ASB 3-1, "Protection Against Postulated Piping Failures in Fluid Systems Outside Containment" and MEB 3-1, "Postulated Break and Leakage Locations in Fluid Systems Outside Containment." Evaluation of protection against dynamic effects associated with the postulated pipe system failure is covered in Section 3.6 of this SER.

The scope of review of the turbine generator included descriptive information in FSAR Section 10.2, flow charts, and diagrams. The basis for acceptance in the staff's review was conformance of the design criteria and bases and design of the turbine generator system to GDC 4 with respect to the prevention of the generation of turbine missiles, the additional guidance of SRP Section 10.2, and industry codes and standards.

On the basis of its review, the staff concludes that the turbine generator overspeed protection system meets the requirements of GDC 4 and the guidance of SRP Section 10.2, can perform its designed safety functions, and is, therefore, acceptable.

10.2.1*

10.2.2*

10.2.3 Turbine Disk Integrity

See Section 3.5.1.3.

10.3 Main Steam Supply System

The function of the main steam supply system is to convey steam from the steam generators to the high-pressure turbine and other auxiliary equipment for power generation. Section 10.3.1 evaluates the safety-related portion of the main steam system including the main steam isolation valves (MSIVs). Section 10.3.2 evaluates the nonsafety-related portion of the main steam system downstream of the MSIVs up to and including the turbine stop valves.

The main steam supply system was reviewed in accordance with SRP Section 10.3 (NUREG-0800). Conformance with the acceptance criteria except as noted below formed the basis for the staff's evaluation of the main steam supply system with respect to the applicable regulations of 10 CFR 50.

The acceptance criteria for the main steam supply system include RG 1.115, "Protection Against Low Trajectory Turbine Missiles." Compliance with the guidelines of RG 1.115 is evaluated separately in Section 3.5.1.3 of this SER.

10.3.1 Main Steam Supply System (up to and Including the Main Steam Isolation Valves)

The function of the main steam supply system is to convey steam from the steam generators to the high-pressure turbine and other auxiliary equipment for power generation. The steam produced in the four steam generators is conveyed in four separate main steamlines from the steam generators through the main steam isolation valves to a main steam header and then through four lines to the high-pressure turbine. Each of the four main steamlines contains one main steam isolation valve (MSIV). The portions of the main steamlines from the steam generators through the containment, including the MSIVs, the main steam safety valves, and the power-operated relief valves are located in the seismic

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Category I, flood- and tornado-protected main steam valve building (see Sections 3.4.1 and 3.5.2 of this SER), thus complying with the guidelines of RG 1.29, Position C.2, as it relates to damage to the main steam line by other systems as a result of a safe shutdown earthquake (SSE). The lines are designed to Quality Group B and seismic Category I standards up to and including the MSIVs. The lines from the MSIVs to the turbine building wall are designed to Quality Group C, seismic Category I standards, as are the lines to the auxiliary feedwater pump steam turbine. All branch lines from the seismic Category I, Quality Group B portions of the main steamline up to and including the first normally shutoff valve in the branch line also are designed to seismic Category I, Quality group B standards, thus complying with the guidelines of RG 1.29, Position C.1.f, as it relates to the design of portions of main steam and branch lines, and Position C.3 as it relates to the extension of seismic Category I requirements. Thus, this portion of the main steamline satisfies the requirements of GDC 2.

Main steam isolation is provided by a spring-loaded, pneumatically operated valve in each steamline located just outside of containment. By pressurizing or venting appropriate piston compartments, the isolation valve will open or close. Steam is used for the operation of this valve and is automatically taken either from the inlet or the outlet side of the valve body depending on which side is pressurized. When steamline pressure is inadequate, the MSIV can be held open using nitrogen supplied at 185 psig. To ensure safety function actuation, all solenoid valves are provided in redundant pairs, powered from separate Class 1E power sources. The MSIVs automatically close on low-pressure signals in any steamline, on a high containment pressure signal, or on a high steam pressure rate in any steamline. The MSIVs can also be closed manually by operator action from the control room. The MSIVs are designed to close in 5 sec or less and to stop steam flow from either direction. A steamline break upstream or downstream of the MSIVs coupled with an MSIV failure to close will not result in a blowdown of more than one steam generator. In the event of a steamline break upstream of an MSIV and a failure of an MSIV to close on an unaffected steam generator, blowdown of the unaffected steam generator through the break is prevented by the closure of the MSIV for the affected steam generator. Blowdown through the turbine and condenser is prevented by closure of the nonseismic Category I turbine stop valves and main steam dump valves, which serve as an acceptable backup for this accident in accordance with the guidelines of Issue No. 1 of NUREG-0138, "Staff Discussion of Fifteen Technical Issues Listed in Attachment to November 3, 1976 Memorandum From Director, NRR, to NRR Staff."

One main steam presssure relief valve and one bypass pressure relief valve are provided on each main steamline downstream of the main steam safety valves but upstream of the MSIV. Three valves meet seismic Category I and Quality Group B requirements. The main steam pressure relief valves provide modulating pressure control of each steam generator when the normal turbine bypass is not available. The combined relief capacity of the main steam pressure valves on four steamlines is 15% of maximum steam flow. The operation of these valves is automatically controlled by steamline pressure during plant operations. The relief valves automatically open and exhaust to the atmosphere whenever the steamline pressure exceeds the opening setpoint. The bypass relief valves are provided as a backup to the main steam relief valves. These valves provide the capability to dump steam to the atmosphere remotely from the control room following an

SSE coincident with loss of offsite power. This complies with the requirements that safe shutdown be achieved with dependence on safety-grade components only with either onsite or offsite power, as specified in Positions A.2, A.3, and A.4 of BTP RSB 5-1, "Design Requirements of the Residual Heat Removal System."

Twenty seismic Category I, Quality Group B safety valves (five on each main steamline) are provided. The safety valves have a combined relief capacity of more than 100% of the design steam flow. The five safety valves on each line are located outside of containment, upstream of the main steam relief valve, the bypass relief valve, and the MSIV, in accessible areas of the seismic Category I main steam valve building. The MSIVs, safety valves, and power-operated relief valves will undergo preoperational functional testing at normal design temperature and pressure. MSIV closure times and safety and relief valve setpoints will be verified. Therefore, the staff concludes that the design of the main steam supply system meets the requirements of GDC 34, "Residual Heat Removal," by complying with the guidelines of BTP RSB 5-1 and Issue No. 1 of NUREG-0138.

The equipment that must function to ensure main steam isolation when necessary is protected against the effects of high-energy-pipe breaks (see Sections 3.6.1 and 3.11 of this SER). This equipment is in tornado-missile-protected structures and is such that it is not affected by internally generated missiles (see Section 3.5.1.1 of this SER). Thus, the requirements of GDC 4 and the guidelines of RG 1.117, Positions C.1 and C.2, are satisfied. There is no sharing between the three Millstone units of any portion of the main steam supply system; thus, the requirements of GDC 5 are not applicable.

On the basis of this review, the staff concludes that the main steam supply system from the steam generators through the main steam isolation valves meets the requirements of GDC 2, 4, and 34 with respect to protection against natural phenomena, missiles, environmental effects and residual heat removal because the system complies with the guidelines of RGs 1.29 (Positions C.1 and C.2) and 1.117 (Positions C.1 and C.2) as they relate to the system's seismic classification, protection against tornado missiles, and high- and moderate-energy-pipe breaks, and the guidelines of BTP RSB 5-1 and Issue No. 1 of NUREG-0138 as they relate to residual heat removal and limitation of blowdown. The system, therefore, is acceptable. The main steam supply system meets the acceptance criteria of SRP Section 10.3.

10.3.2 Main Steam Supply System (Downstream of the Main Steam Isolation Valves)

This portion of the main steam system is not required to effect or support safe shutdown of the reactor.

The main steam system is designed to deliver steam from the steam generators to the high-pressure turbine. The main steam and turbine steam systems provide steam to the feedwater pump turbines, auxiliary feedwater pump turbine, auxiliary steam system, turbine gland seal systems, turbine steam dump (bypass) system, and steam supply to the moisture separator reheaters. The main steam system from the MSIV to the turbine building wall is designed as seismic Category I, ASME Code, Section III, Class 3 (Quality Group C). The piping from the turbine building wall to the turbine stop valves and all branch lines are designed to the requirements of ANSI Std. B31.1 and are acceptable.

The scope of review of the main steam supply system (between the main steam isolation valves and up to and including the turbine stop valves) included descriptive information in FSAR Section 10.3, flow charts, and diagrams. The basis for acceptance in the staff review was conformance of the design criteria and bases and design of main steam supply system to the acceptance criteria of SRP Section 10.3.

On the basis of its review, the staff concludes that the main steam supply system between the main steam isolation valves and up to and including the turbine stop valves is in conformance with the above-cited criteria and design bases, can perform its designed functions, and is, therefore, acceptable.

10.3.3*

10.3.4*

10.3.5 Secondary Water Chemistry

In late 1975, the staff incorporated provisions into the Standard Technical Specifications that required limiting conditions for operation and surveillance requirements for secondary water chemistry parameters. The Technical Specifications for all pressurized-water-reactor plants that have been issued an operating license from 1974 until 1979 contain either these provisions or a requirement to establish these provisions after baseline chemistry conditions have been determined. The intent of the provisions was to provide added assurance that the operators of newly licensed plants would properly monitor and control secondary water chemistry to limit corrosion of steam generator components such as tubes and tube support plates.

In a number of instances, the Technical Specifications have significantly restricted the operational flexibility of some plants with little or no benefit with regard to limiting degradation of steam generator tube and the tube support plates. On the basis of this experience and the knowledge gained in recent years, the staff has concluded that Technical Specification limits are not the most effective way of ensuring that steam generator degradation will be minimized.

Because of the complexity of the corrosion phenomena involved and the state-of-the-art as it exists today, the staff believes that, instead of specifying limiting conditions in the Technical Specification, a more effective approach would be to specify a Technical Specification that required the implementation of a secondary water chemistry monitoring and control program containing appropriate procedures and administrative controls. This has been the approach for control of secondary water programs since 1979.

The required program and procedures are to be developed by applicants, with input from their reactor vendors or other consultants, to account for site- and

*The July 1981 edition of the Standard Review Plan (NUREG-0800) does not include sections addressing FSAR sections that consist of background or design data used in the review of other sections. The section numbers are included to provide continuity for subsequent SER section numbers that will correlate with the associated SRP section numbers.

plant-specific factors that affect water chemistry conditions in the steam generators. In the staff's view, plant operation following such procedures would provide assurance that licensees would devote proper attention to controlling secondary water chemistry, while also providing the needed flexibility to allow them to deal effectively with an offnormal condition that might arise.

Consequently, the staff requested that the applicant propose a secondary water chemistry program that would be referenced in a condition to the operating license and would replace any proposed Technical Specifications on secondary water chemistry.

This section was reviewed through Amendment 7 including letter dated April 19, 1984.

The staff has reviewed secondary water chemistry in accordance with SRP Sections 5.4.2.1 and 10.3.5 and BTP MTEB 5-3 (NUREG-0800). The proposed program addresses the six program criteria of the staff position discussed below and is based on the steam generator water chemistry program recommended by the Steam Generators Owner Group (SGOG).

The program monitors the critical parameters to inhibit steam generator corrosion and tube degradation. The limits and sampling schedules for these parameters have been established for condensate pump common discharge, steam generator blowdown, and feedwater. The modes include normal power operation, startup from hot shutdown/hot standby, and cold layup. Sampling frequencies, control points for the critical parameters, and process sampling points have been identified. Plant procedures used for measuring the values of the critical parameters have been similarly identified.

The staff finds that the applicant's secondary side chemistry monitoring and control program

(1) is capable of reducing the probability of abnormal leakage in the reactor coolant pressure boundary by inhibiting steam generator corrosion and tube degradation and thus meets the requirements of GDC 14

(2) adequately addresses all of the program criteria delineated in the NRC staff position on control and monitoring of secondary water

(3) is based on the Steam Generator Owners Group recommended steam generator water chemistry program

(4) monitors the secondary coolant purity in accordance with BTP MTEB 5-3, Revision 2, and thus meets Acceptance Criterion 3 of SRP Section 5.4.2.1, Revision 2

(5) monitors the water quality of the secondary side water in the steam generators to detect potential condenser cooling water inleakage to the condensate and thus meets Position II.3.f(1) of BTP MTEB 5-3, Revision 2

(6) describes the methods for control of secondary side water chemistry data and record management procedures and corrective actions for off-control point chemistry and thus meets Positions II.3.f.(2)-(6) of BTP MTEB 5-3, Revision 2

Routine changes in the program should be reviewed (see Section 6.5.1.6 of the Standard Technical Specifications) and should be reported under biannual FSAR updates as required by 10 CFR 50.71. Nonconservative changes, that is, relaxation in sample frequency or in impurity limits, should be submitted to the staff for review before the changes are implemented. However, all volatile treatment program changes that incorporate boric acid or calcium hydroxide additions to the steam generator water to further reduce corrosion problems such as tube denting or pitting do not require NRC review provided an evaluation performed in accordance with 10 CFR 50.59 demonstrates that the change does not involve an unreviewed safety question or require a change in the plant Technical Specifications.

The annual operating report should include an evaluation of the secondary side water chemistry program with an evaluation of the trends and a summary of the total time during the reporting period the various chemistry parameters were out of the recommended control range.

On the basis of its evaluation, the staff concludes that the proposed secondary water chemistry monitoring and control program meets (1) the requirements of GDC 14 insofar as secondary water chemistry control ensures primary boundary material integrity; (2) Acceptance Criterion 3 of SRP Section 5.4.2.1, Revision 2; (3) Position II.3 of BTP MTEB 5-3, Revision 2; and (4) the program criteria in the staff's position. The program, therefore, is acceptable.

10.3.6 Main Steam and Feedwater Materials

The staff concludes that the main steam and feedwater system materials are acceptable and meet the relevant requirements of 10 CFR 50.55a, GDC 1, and Appendix B to 10 CFR 50. This conclusion is based on the following.

The applicant selected materials for Class 2 and 3 components of the steam and feedwater systems that satisfy Appendix I of Section III of the ASME Boiler and Pressure Vessel Code, and meet the requirements of Parts A, B, and C of Section II of the Code. The applicant also has met the recommendations of RG 1.85, which describes acceptable Code cases that may be used in conjunction with this industrial standard.

In this time frame, the Code allowed waiving of impact testing of main steam and feedwater materials. However, on the basis of the results of impact testing by other applicants of the same specification steels, and correlations of the metallurgical characterization of these steels with the fracture toughness data presented in NUREG-0577, the staff concludes that the fracture toughness properties of the ferritic materials in the main steam and feedwater systems have adequate safety margins against the possibility of nonductile behavior and rapidly propagating fracture.

The applicant has satisfied to the extent practical, the recommendations of RG 1.71, "Welder Qualification for Areas of Limited Accessibility," by meeting the regulatory positions in the guide or by meeting alternative approaches, which the staff has reviewed and found to be acceptable (see Section 4.5.1). The onsite cleaning and cleanliness controls during fabrication satisfy the position given in RG 1.37, "Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants," and the requirements of ANSI Std. N 45.2-1-1973, "Cleaning of Fluid Systems and Associated Components During Construction Phase of Nuclear Power Plants."

10.4 <u>Other Features</u>

10.4.1 Main Condenser

The main condenser is designed to function as a heat sink for the turbine exhaust steam, turbine bypass steam, and other turbine cycle flows, and to receive and collect condensate flows for return to the steam generators. The main condenser transfers heat to the circulating water system, which uses Long Island Sound to dissipate the rejected heat.

The main condenser is not required to effect or support safe shutdown of the reactor to perform in the operation of reactor safety features. The main condenser has three shells and is designed to produce a turbine back pressure of 1.5 in. mercury absolute when operating at rated turbine output. The main condenser design includes provisions for condensate deaeration and hotwell surge storage of condensate for an approximately 5-min supply at design conditions. Offgas from the main condenser is processed in the condenser evacuation system, which is described and evaluated in Section 10.4.2 of this SER.

The main condenser is designed to accept full load exhaust steam from the main turbine and reactor feedwater pump turbines, up to 40% of the main steam flow from the turbine steam bypass system, and other cycle steam flows. The main condenser is also designed to deaerate the condensate to the required water quality. Titanium tubes have been used to minimize corrosion and erosion of condenser tubes. In addition, an impressed current cathodic protection system is provided to protect the titanium tube ends, copper-nickel water box cladding, and the aluminum bronze tube sheets. Condenser tube leakage could result in degradation of the feedwater quality with potential for corrosion of secondary system components. The applicant monitors condensate sodium content by means of an automatic hotwell sampling system to give an indication of tube leakage. The applicant, in response to a request for additional information, provided details on the detection, control, and correction of condenser cooling water leakage into the condensate. The adequacy of the secondary sampling system for leak detection is evaluated in Section 9.3.2 of this SER.

The applicant will include preoperational and startup tests of the main condenser in accordance with the recommendations of RG 1.68, "Initial Test Programs for Water Cooled Reactor Power Plants." The adequacy of the test program is evaluated in Section 14.1 of this SER.

The scope of review of the main condenser included layout drawings and descriptive information of the condenser in FSAR Section 10.4.1.

The basis for acceptance in the staff review was conformance of the design criteria and bases and design of the condenser with the acceptance criteria in Section II of SRP Section 10.4.1 and industry standards.

On the basis of its review, the staff concludes that the main condenser is in conformance with the above-cited criteria and design bases, can perform its design function, and is, therefore, acceptable.

10.4.2 Main Condenser Evacuation System

FSAR Section 10.4.2 contains information pertaining to the main condenser evacuation (air removal) system, the system design bases, and the applicable acceptance criteria.

The staff has reviewed the applicant's design, design criteria, and design bases for the main condenser evacuation system (MCES) for Millstone Unit 3 in accordance with Section II of SRP Section 10.4.2 (NUREG-0800). The SRP acceptance criteria include GDC 60 and 64 and Heat Exchanger Institute Standard, "Standards for Steam Surface Condensers." Guidelines for implementation of the requirements of the acceptance criteria are provided in the regulatory guides referenced in Section II of the SRP. Conformance to the acceptance criteria of the SRP provides the bases for concluding that the MCES meets the requirements of 10 CFR 50.

The MCES is designed to establish and maintain main condenser vacuum by removing noncondensible gases from the main condenser. Two 100% capacity steam jet air ejector units and two horizontal, motor-driven condenser air removal pumps are provided. The steam jet air ejectors are triple-element, first-stage and single-element, second-stage units with water-cooled inter- and after-condensers. Motive steam is supplied to the air ejectors from the auxiliary steam supply header. Air and noncondensible gases removed from the main condenser shells by the steam jet air ejector units are continuously monitored for radioactivity and discharged through the plant vent stack.

The condenser air removal mechanical vacuum pumps are rotary water-ring-type pumps and are used to draw initial condenser vacuum during plant startups. The air and noncondensible gases removed by air removal mechanical vacuum pumps are directly discharged to the atmosphere through a vent stack on the condensate polishing enclosure roof of Warehouse No. 5.

The applicant has taken an exception to SRP Section 11.5 by not monitoring airborne noble gas radioactivity in the exhaust from the main condenser air removal mechanical vacuum pumps. Instead, the applicant proposes indirect monitoring of noble gas in this pathway by a calculational method based on noble gas activity measured at the main condenser air ejector monitor. The applicant states that

(1) Because the main condenser air ejector monitor will be in service before any shutdown, air ejector monitor readings can be used to calculate the airborne noble gas radioactivity concentration during subsequent startup operation with the mechanical vacuum pumps.

(2) The detailed methodology for accounting for noble gas released from the mechanical vacuum pumps will be specified in the Offsite Dose Calculation Manual (ODCM).

The staff finds the proposed alternative method acceptable for the following reasons:

(1) The expected quantity of noble gas released from this pathway is less than 0.1 Ci per year (amounts are so low that the GALE code does not provide this value) compared with the expected plant total noble gas release of 560 Ci per year.

(2) The main condenser air ejector monitor readings before and after the startup operation can provide representative indication of the noble gas concentrations in the mechanical vacuum pump exhaust.

(3) The mechanical vacuum pumps will be used only during the plant startup operations for a period of less than 48 hours each.

The applicant has committed to provide (1) an acceptable ODCM, which will describe a detailed methodology for calculating noble gas releases from the mechanical vacuum pump operations at least 6 months before fuel loading for the staff's review and approval, and (2) a continuous sampling provision for iodines and particulates in the mechanical vacuum pump exhaust during the mechanical vacuum pump operation. This is a confirmatory item.

The applicant's provisions for quality assurance for the design, construction, and operational phases of the MCES were reviewed to determine conformance with RGs 1.33 and 1.123, as provided in the SRP. No exceptions to the criteria were noted. Equipment quality group classifications were reviewed to determine conformance with RG 1.26, as provided in the SRP. No exceptions were noted. The MCES capacity was reviewed to determine conformance with Heat Exchanger Institute Standard, "Standards for Steam Surface Condensers," as provided in SRP Section 10.4. No exceptions were noted.

The MCES includes equipment and instruments to establish and maintain condenser vacuum and to prevent an uncontrolled release of gaseous radioactive material to the environment. The scope of the staff's review included the system's capability to transfer radioactive gases to the offgas or ventilation exhaust systems and the design provisions incorporated to monitor and control releases of radioactive materials in effluents.

The staff has reviewed the applicant's system descriptions, piping and instrumentation diagrams, and design criteria for the MCES components in accordance with the SRP. It concludes that the MCES design is acceptable provided the applicant submits an acceptable methodology for calculating the noble gas releases from the mechanical vacuum pump operation in the ODCM at least 6 months before fuel loading.

10.4.3 Turbine Gland Sealing System

FSAR Section 10.4.3 contains information pertaining to the turbine gland sealing system, the design bases, and applicable acceptance criteria.

The staff has reviewed the applicant's design, design criteria, and design bases for the turbine gland sealing system for Millstone Unit 3 in accordance with Section II of SRP Section 10.4.3 (NUREG-0800). The acceptance criteria include GDC 60 and 64. Guidelines for implementation of the requirements of the acceptance criteria are provided in the regulatory guides identified in Section II of the SRP. Conformance to the acceptance criteria provides the bases for concluding that the turbine gland sealing system meets the requirements of 10 CFR 50.

The turbine gland sealing system provides sealing steam to the main turbine generator shaft to prevent the leakage of air into the turbine casings and the potential escape of radioactive steam into the turbine building. The turbine gland sealing system uses three steam sources: main, auxiliary, and/or extraction steam. The system is normally operated with extraction steam. During low-load operation (startup and shutdown), steam is taken from the main steamlines ahead of the turbine stop valves. When main steam is unavailable, the gland steam seal system is operated with auxiliary steam (nonradioactive).

The steam supply is passed through the turbine gland seals and condensed in the steam packing exhauster condenser. The condensate is returned to the main condenser hotwell, and noncondensible gases are discharged by one of two motor-driven blowers to the environment. The turbine gland sealing system is designed in accordance with Quality Group D standards, as defined in RG 1.26. A more detailed discussion of the turbine gland sealing system is presented in FSAR Section 10.4.3.

The applicant has taken an exception to SRP Section 11.5 by not monitoring airborne gaseous radioactivity in the exhaust from the turbine gland sealing condenser vent. Instead, the applicant proposes indirect monitoring of airborne radioactivity (noble gas, iodines, and particulates) in this pathway by a calculational method based on (1) noble gas activity measured at the main condenser air ejector and (2) particulate and iodine activities in the steam generator water. The staff finds the proposed alternative method acceptable for the following reasons:

(1) The condenser air ejector monitor provides continuous monitoring of noble gas concentrations in main steam which is the only source for the turbine gland sealing system.

(2) The steam generator blowdown monitor and associated sampling and analysis program provide a continuous monitoring of particulates and iodines in the secondary system.

(3) The expected quantity of noble gas, iodines, and particulates released from this pathway would be inconsequential (less than 0.1 Ci per year) from both a radiation dose standpoint and from a total plant release (560 Ci per year) standpoint for either normal releases or accident conditions.

The applicant has committed to provide an acceptable ODCM, which will describe the detailed methodologies for calculating noble gas, iodine, and particulate releases from the turbine gland sealing condenser vent at least 6 months before fuel loading for the staff's review and approval. This is a confirmatory item.

The staff concludes that the turbine gland sealing system design is acceptable provided the applicant submits an acceptable methodology for calculating the noble gas, iodine, and particulate releases in the exhaust from the turbine gland sealing condenser vent in the ODCM at least 6 months before fuel loading.

10.4.4 Turbine Bypass System

The turbine bypass system is designed to bypass up to 40% of main steam flow to the main condenser. This capacity together with a 10% reactor automatic step load capability is sufficient to withstand a 50% generator load loss without tripping the turbine or causing control rod movement. The turbine bypass system is used to control cooland temperature as follows: (1) during the reactor heat-up to rated pressure, (2) while the turbine generator is being brought up to speed and synchronized, (3) during power operation when the reactor steam generator exceeds the transient turbine steam requirements, and (4) during reactor cooldown. The system is not required to perform during accident conditions.

The bypass system is composed of the following: (1) nine air-operated valves, (2) associated instruments and controls, and (3) piping. Each valve is rated for a capacity of approximately 4.4% of the main steam flow at full load pressure and temperature. The nine bypass valves are connected to the main steam header downstream of the main steam isolation valves by a turbine bypass header and discharge the steam directly to the main condenser (three valves to each condenser shell). The turbine bypass system is not a safety-related system and is not required for plant shutdown following an accident. The turbine bypass valves are designed to fail closed on loss of electric power or air system pressure to the valve control system. The turbine bypass valves are designed to close on loss of main condenser vacuum.

The applicant will include preoperational and startup tests of the turbine bypass system in accordance with recommendations of RG 1.68. The adequacy of the test program is evaluated in Section 14.1 of this SER.

The turbine bypass system meets the recommendations of BTPs ASB 3-1, "Protection Against Postulated Piping Failures in Fluid System Piping Outside Containment," and MEB 3-1, "Postulated Break and Leakage Locations in Fluid System Piping Outside Containment." Evaluation of protection against dynamic effects associated with the postulated pipe system failures is covered in Section 3.6 of this SER.

The scope of review of the turbine bypass system included drawings, piping and instrumentation diagrams, and descriptive information of the system in FSAR Section 10.4.4.

The basis for acceptance in the staff review was conformance of the design criteria and bases and design of the turbine bypass system with the acceptance criteria and guidance of SRP Section 10.4.4 and industry standards.

On the basis of its review, the staff concludes that the turbine bypass system is in conformance with the above-cited criteria and design bases, can perform its design function, and is, therefore, acceptable.

10.4.5 Circulating Water System

The circulating water system was reviewed in accordance with SRP Section 10.4.5 (NUREG-0800).

The nonsafety-related, Quality Group D, nonseismic Category I circulating water system (CWS) provides cooling water to the main condensers. There are six one-sixth-capacity (152,000 gpm) circulating water pumps, which are located in the circulating and service water pumphouse. These pumps draw water from the Niantic Bay and pump it through buried lines to the main condenser located in the turbine building. The water from the main condenser flows through a discharge tunnel into a quarry and from the quarry through a discharge channel into the Long Island Sound. All portions of the circulating and service water pumphouse and discharge tunnel that support or, by failure, could damage the safety-related service water system are designed to quality assurance and seismic Category I designations.

The applicant has examined the effects of possible flooding of safety-related equipment as a result of postulated failure in the CWS and stated that a failure in any of the CWS lines would have no effect on safety-related equipment. There are no safety-related systems within the turbine building. There is only one pipe tunnel at el 14 ft 6 in. connecting the turbine building to the safety-related auxiliary building; this tunnel is totally sealed with a fire barrier and will prevent any water from entering the auxiliary building. There are no other passageways, pipe chases, or cableways that could be the route through which floodwater could reach safety-related equipment by failure of CWS lines.

High water level in the condenser circulating water discharge pit will sound an alarm in the control room enabling the operator to stop the circulating water flow within 15 min.

The applicant indicated that the sump alarm system is not safety related, but numerous other alarms would be generated in addition to a turbine trip signal. Because reliance on nonsafety-related equipment is not acceptable to the staff, the applicant committed to replace the existing siding liner panel between column lines A39 and A43 with pressure release siding. A circulating water expansion joint rupture in the turbine building will result in internal flooding until the water level reaches el 28 ft. Upon reaching this elevation, the siding liner panel located at el 24 ft 6 in. will blow out. The panel is located on the west side of the turbine building away from several Category I structures that are located east of the turbine building. Therefore, continued operation of the circulating water pumps will result in damage to safety-related systems or components.

On the basis of this review, the staff concludes that the CWS meets the requirements of GDC 4 with respect to protection of safety-related systems against failures in nonsafety-related systems and, therefore, is acceptable. The CWS meets the acceptance criteria of SRP Section 10.4.5.

10.4.6 Condensate Cleanup System

The purpose of the condensate cleanup system (CCS) is to remove dissolved and suspended solids from the condensate to maintain a high quality of the feedwater being supplied to the steam generators under all normal plant conditions (startup, shutdown, hot standby, and power operation). This is accomplished by directing the full flow of condensate to a set of mixed-bed demineralizer units. Because the demineralizers need periodic resin regeneration, spare units are provided in the system to replace units taken out of service. The system provides final polishing of the secondary cycle condensate water.

The staff has reviewed the CCS equipment design, materials, and operation in accordance with SRP Section 10.4.6 (NUREG-0800). The CCS is designed to assist in the control of the secondary side water chemistry and is part of the total control system.

The CCS includes all components and equipment necessary for the removal of dissolved and suspended impurities that may be present in the condensate. The system meets the requirements for condensate cleanup capacity and contains adequate instrumentation to monitor the effectiveness of the system.

The staff has reviewed the sampling equipment, sampling locations, and instrumentation to monitor and control the CCS process parameters. On the basis of this review, it finds that the instrumentation and sampling equipment provided are adequate to monitor and control process parameters.

On the basis of its review of the applicant's proposed criteria and design bases for the CCS and the requirements for operation of the system, the staff concludes that the design of the CCS and supporting systems meets its guidelines and is, therefore, acceptable.

The secondary water chemistry monitoring and control program is evaluated in Section 10.3.5.

10.4.7 Condensate and Feedwater System

The condensate and feedwater system was reviewed in accordance with SRP Section 10.4.7 (NUREG-0800).

The condensate and feedwater system provides feedwater from the condenser to the steam generators and includes the piping and components from the condenser hotwell, through the condensate pumps, condensate demineralizers, low-pressure feedwater heaters, feedwater pumps, high-pressure feedwater heaters, flow control valves, and containment isolation valves to the four steam generators. There are three 50% capacity condensate pumps and three 50% capacity feedwater pumps. The three condensate pumps and one of the feedwater pumps are motor driven. The remaining two feedwater pumps are turbine driven. One condensate pump and one motor-driven feedwater pump remain on standby and are normal means of starting up and shutting down the plant.

The system serves no safety function, with the exception of containment isolation integrity, and is, therefore, classified as nonsafety related, Quality Group D, nonseismic Category I. Adequate isolation is provided at connections between seismic and nonseismic Category I systems; therefore, failure of nonsafety related portions of the condensate and feedwater system will not affect safe plant shutdown.

The portions of the system classified as safety related are (1) the main feedwater piping from the containment isolation and check valves to the steam generators, (2) the piping in the main steam feedwater valve building, and (3) the interconnecting piping between the auxiliary feedwater system and the feedwater lines. These portions of the system are designed to seismic Category I, Quality Group B requirements to ensure feedwater isolation in accident situations and are located in seismic Category I, flood- and tornado-protected structures (see Section 3.4.1 and 3.5.2 of this SER). Thus, the requirements of GDC 2 and the guidelines of RG 1.29, Positions C.1 and C.2, are satisfied.

The structure also provides protection against tornado missiles. The essential equipment is separated from the effects of internally generated missiles and is not affected by failures in high-energy piping (see Sections 3.5.1.1 and 3.6.1 of this SER). Thus, the requirements of GDC 4 are satisfied. No portion of the condensate and feedwater system is shared with other Millstone units so that the requirements of GDC 5 are not applicable.

Automatic isolation of the main feedwater system is provided when required to mitigate the consequences of a steam or feedwater line break. The pneumatically operated main feedwater isolation valves (one per steam generator) close within 5 sec on receipt of an engineered safety features actuation signal. Redundant feedwater line isolation is provided by the fail-closed main feedwater regulating valves and bypass valves, which serve as an acceptable backup. The safety-related auxiliary feedwater system automatically provides flow to the steam generators through the main feedwater lines for decay heat removal upon failure of the condensate and feedwater system. See Section 10.4.9 of this SER for further discussion of the auxiliary feedwater system. Thus, the requirements of GDC 44 are satisfied. The safety-related portions of the system are located in accessible areas and will receive periodic inspection and testing in accordance with the Technical Specifications. Thus, the requirements of GDC 45 and 46 are satisfied.

The condensate and feedwater system is designed with features to preclude the potential for damaging flow instabilities (waterhammer). Millstone Unit 3 uses Westinghouse Model F steam generators that have top discharge feedrings. The feedwater system piping is arranged to prevent waterhammer from occurring at the steam generator.

On the basis of this review, the staff concludes that the safety-related portion of the condensate and feedwater system meets the requirements of GDC 2, 4, 44, 45, and 46 with respect to its protection against natural phenomena, missiles, and environmental effects, decay heat removal function, inservice inspection and testing and meets the guidelines of RG 1.29 with respect to its seismic classification and is, therefore, acceptable. The condensate and feedwater system meets the acceptance criteria of SRP Section 10.4.7.

10.4.8 Steam Generator Blowdown System

The steam generator blowdown system (SGBS) is used in conjunction with the condensate demineralizer, chemical addition, and sample systems to control the chemical composition of the steam generator shell-side water within specified limits during all operating modes. The blowdown fluids are directed to the blowdown tank and then to the condenser.

The staff has reviewed the SGBS in accordance with SRP Section 10.4.8 (NUREG-0800).

The SGBS controls the concentration of chemical impurities and radioactive materials in the secondary coolant. The scope of review of the SGBS included piping and instrumentation diagrams, seismic and quality group classifications, design process parameters, and instrumentation and process controls. The review has included the applicant's evaluation of the proposed system operation and the applicant's estimate of the controlling process parameters.

The SGBS is monitored continuously for radiation in the secondary side of the steam generator. It has the capability of diverting the blowdown liquid (after isolation and substantial cooldown of a steam generator) to the radioactive liquid waste system in the event of a high radiation signal resulting from a steam generator tube leak.

The portion of the steam generator blowdown system up to and including the containment isolation valves is seismic Category I and designated ASME Code, Section II, Class 2. All other piping and equipment in the steam generator blowdown system are not safety related and are designed and built to American National Standards Institute (ANSI) B31.1 requirements. Thus, the SGBS meets the quality standards requirements of GDC 1 and the seismic requirements of GDC 2.

Instrumentation and automatic controls are provided to monitor and control the operation of the blowdown system, with provision for sampling of the blowdown, in conformance with the guidelines of BTP MTEB 5-3.

The secondary water chemistry monitoring and control program is evaluated in Section 10.3.5.

On the basis of the foregoing evaluation, the staff concludes that the proposed steam generator blowdown system meets its guidelines and is acceptable.

10.4.9 Auxiliary Feedwater System

The auxiliary feedwater system (AFWS) was reviewed in accordance with SRP Section 10.4.9 (NUREG-0800).

The staff reviewed the AFWS against the specific acceptance criteria of SRP Section 10.4.9 as follows:

(1) GDC 2 as it relates to structures housing the system and the system itself being capable of withstanding the effects of earthquakes. Acceptability is based on meeting Position C.1 of RG 1.29 for safety-related portions and Position C.2 for nonsafety-related portions.

(2) GDC 4 with respect to structures housing the system and the system itself being capable of withstanding the effects of external missiles and internally generated missiles, pipe whip, and jet impingement forces associated with pipe breaks. The basis for acceptance for this criterion is set forth in SRP Sections 3.5 and 3.6.

(3) GDC 5 as it relates to the capability of shared systems and components important to safety to perform required safety functions.

(4) GDC 19 as it relates to the design capability of system instrumentation and controls for prompt hot shutdown of the reactor and potential capability for subsequent cold shutdown. Acceptance is based on meeting BTP RSB 5-1 with regard to cold shutdown from the control room using only safety-related equipment.

(5) GDC 34 and 44 to ensure the capability to transfer heat loads from the reactor system to a heat sink under both normal operating and accident conditions; redundancy of components so that under accident conditions the safety function can be performed assuming a single active component failure

(this may be coincident with the loss of offsite power for certain events); and the capability to isolate components, subsystems, or piping if required so that the system safety function will be maintained.

(6) GDC 45 as it relates to design provisions made to permit periodic inservice inspection of system components and equipment.

(7) GDC 46 as it relates to design provisions made to permit appropriate functional testing of the system and components to ensure structural integrity and leaktightness, operability and performance of active components, and capability of the integrated system to function as intended during normal, shutdown, and accident conditions.

The following evaluation discusses the implementation of these acceptance criteria and follows the order of SRP Section 10.4.9 (NUREG-0800). This evaluation also incorporates the staff review of the applicant's response to NUREG-0737, Item II.E.1.1. This includes

(1) an evaluation against the deterministic criteria of the SRP

(2) an evaluation against the generic recommendations of NUREG-0611

(3) an evaluation of system reliability based on the applicant's reliability study

(4) an evaluation of the design basis for the flow capability for the system

The AFWS is designed to supply high-pressure feedwater to the secondary side of the steam generators when the normal feedwater system is not available, thereby maintaining the heat sink capabilities of the steam generator. It is an engineered safeguards system that is relied on to aid in preventing core damage in the event of transients such as loss of normal feedwater or a secondary system pipe rupture. The system consists of two half-capacity motor-driven pumps and one full-capacity turbine-driven pump with associated valves, piping, controls, and instrumentation. The two motor-driven pumps are powered from two separate buses of emergency onsite electrical power and each discharge feedwater into two steam generators. The steam turbine-driven pump supplies water to all four steam generators. Each of these supply lines contains check valves, motor-operated isolation valves, and flow control valves. The steam for the turbine is supplied from three steam generators (A, B, and D) upstream of the main isolation valves. The AFW flow to the steam generators is limited by flow venturis located in each AFW line. These venturis are sized to restrict the flow to a depressurized steam generator. The turbine-driven pump and controls are powered completely independent of the motor-driven feedwater pumps and controls.

The Millstone Unit 3 AFWS is independent of the other Millstone units. The water source is provided by the demineralized water storage tank (DWST), which is seismic Category I, Quality Group C. The water flows through three seismic Category I, Quality Group C suction lines. Each line contains two locked-open manually operated isolation valves. Because there is a separate AFWS for Millstone Unit 3, GDC 5 is not applicable.

The AFWS is designed to seismic Category I, Quality Group C criteria from the DWST up to but not including the motor-operated isolation valves. The motor-operated isolation valves and the piping and valves from the motor-operated valves to the steam generator are designed to seismic Category I, Quality Group B standards. The Quality Group ratings (B and C) meet the guidelines of RG 1.29.

The AFWS is located within the engineered safety features building and containment building and is thus protected against the effects of natural phenomena and tornado missiles. The DWST is located outside the buildings but is protected against the effects of hurricanes, tornadoes, and missiles by a concrete enclosure. Thus, the AFWS meets the requirements of GDC 2.

There are separate cubicles for each AFW pump to prevent possible internally generated missiles from damaging more than one pump.

The separate cubicle enclosure for the turbine-driven pump protects the motor-driven pumps from potential missiles originating from the turbine-driven pump. As indicated in Section 3.5.1.1 of this SER, the applicant will install inside barriers on the basis of the analysis that the missile from the turbine-driven AFW pump cannot damage safety-related equipment. Pending this confirmation, the staff concludes that the design meets GDC 4 as it relates to protection against internally generated missiles. The AFWS can be operated for approximately 16 hours with the water (340,000 usable gallons) in the DWST. This includes 10 hours for the hot standby condition and an additional 6-hour cooldown period sufficient to reduce reactor coolant hot-leg temperature to 350°F. Makeup is provided to the DWST from the water treating system. An additional source of water (200,000 gal) is provided to each AFW pump suction by the condensate storage tank (nonsafety-related source). The service water system water is available as a long-term safety-grade source of AFW for the steam generators (See Section 9.2.1). Therefore, the AFWS complies with the guidelines of BTP RSB 5-1 and the requirements of GDC 19, with regard to cold shutdown from the control room using only safety-related equipment.

The AFWS has the capability to transfer decay heat loads from the secondary (steam) system under all conditions. The AFWS is designed to supply a minimum of 470-gpm total flow to at least two steam generators, even with the occurrence of a single failure, for the following transients:

(1) loss of normal feedwater
(2) loss of offsite power followed by reactor trip
(3) secondary system pipe rupture
(4) cooldown following steam generator tube rupture
(5) loss-of-coolant accident, small break

Thus, the AFWS complies with GDC 34 and 44.

The AFWS has been designed to permit periodic testing. In addition, the applicant has proposed periodic monthly tests in conformance with the Standard Technical Specifications for Westinghouse pressurized water reactors (NUREG-0452). This meets the requirements of GDC 46.

The AFWS has been designed to permit inservice inspection and periodic inspection of valves and pumps, thus meeting the requirements of GDC 45. The AFWS has two power sources, offsite or onsite (Class 1E) ac power for the motor-driven pumps and steam for the turbine-driven pump. There are no auxiliaries in the train for the turbine-driven pump that require ac power to maintain operation of the train. This meets the guidelines of BTP ASB 10-1.

The AFWS is so designed that the turbine-driven-pump portion of the system can be isolated from the portion containing the motor-driven pumps.

The AFWS is designed to supply water to the steam generators without throttling, thus avoiding throttling as a potential source of waterhammer. Waterhammer is also prevented by lines full of water before AFWS startup.

To meet BTP ASB 10.2 for top-feed design, the applicant committed to perform tests acceptable to the staff to verify that unacceptable feedwater hammer will not occur, using the plant operability procedures for normal and emergency restoration of steam-generated water following loss of normal feedwater flow.

The applicant has evaluated the AFWS against the short- and long-term recommendations of NUREG-0611, "Generic Evaluation of Feedwater Transients and Small Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants." The staff has reviewed the applicant's evaluation. The results of its review are discussed below:

Short-Term Recommendations

Technical Specification Time Limits: The applicant has stated that the outage time limit and the subsequent action time in the Technical Specifications will be as required by the Standard Technical Specifications. This commitment is acceptable.

Administrative Controls on Manual Valves: This recommendation is not applicable to Millstone Unit 3. Millstone Unit 3 does not have common suction piping between the primary water source (DWST) and the AFWS. There are no single valves or multiple valves in series that could interrupt all AFW flow if inadvertently left closed.

Throttling of AFW Flow: Operating procedures for Millstone Unit 3 do not require valve throttling of the auxiliary feedwater flow during initial phases of automatic operation. It should be noted that venturis are added on those lines.

Emergency Procedure for Initiating Backup Water Supply: The applicant has stated that a procedure will be written for transferring to the alternate source of AFW supply. This recommendation included two cases that are to be covered by this procedure.

Case 1 - Pressure switches that trip the pumps on low suction pressure protect the pumps against self damage in the event the DWST is not initially available.

Case 2 - The procedure will cover operator actions required if the DWST is being depleted.

This commitment is acceptable.

Initiation of AFW Flow Following a Loss of AC Power: Under loss of all ac power, the turbine-driven feedwater pump, its associated flow path, and all instrumentation will initiate and maintain the auxiliary feedwater flow using only Class 1E dc power.

AFW Flow Path Verification: The applicant has stated that the plant procedures will require verification of proper alignment of valves following testing or maintenance. The applicant referenced Technical Specifications for flow test following an extended cold shutdown. The staff will review these specifications for confirmation when they are submitted by the applicant.

Nonsafety Grade, Nonredundant Automatic Initiation Signals: The automatic start AFW signals and associated circuitry are safety grade. The details of the design are discussed in Section 7 of this report.

Automatic Initiation of AFW System: The AFWS is automatically initiated.

Additional Short-Term Recommendations

Primary AFW Water Source Low Level Alarm: The applicant stated that redundant DWST level indication is provided on the main control board. The DWST has a volume adequate to maintain the plant at hot standby for 10 hours followed by a 6-hour cooldown to 350°F hot-leg temperature before operation of the residual heat removal system. A low level alarm is provided to indicate that the DWST inventory has decreased to a level that is sufficient to supply feedwater to the auxiliary feedwater system for only 20 min. This is acceptable.

AFW Pump Endurance Test: The applicant stated that the motor-driven auxiliary feedwater pumps will be tested (48-hour endurance test) as part of the Phase II startup testing. The turbine-driven auxiliary feedwater pump will be endurance tested during the hot operational testing. This is acceptable.

Indication of AFW Flow to Steam Generators: Safety-grade flow transmitters, located upstream of the cavitating venturis, indicate flow to the steam generators. The details of the design are discussed in Section 7 of this report.

System Availability During Periodic Surveillance Testing: When either Class 1E auxiliary motor-driven pump is in the test mode, the turbine-driven pump is available for automatic operation.

Long-Term Recommendations

Automatic Initiation of AFW Systems: The automatic start AFW signals and associated circuitry are safety grade. The details of the design are discussed in Section 7 of this report.

Single Valves in the AFW Flow Path: Millstone Unit 3 does not have common suction piping. There are no single valves or valves in series that could interrupt all AFW flow if inadvertently left closed.

Elimination of Dependency of AC Power Following a Complete Loss of AC Power: Under loss of all ac power, the turbine-driven AFW pump, its associated flow path, and all instrumentation will initiate and maintain AFW flow using only Class 1E dc power.

Prevention of Multiple Pump Damage Due to Loss of Suction Resulting From Natural Phenomena: The DWST and interconnected piping are protected from earthquakes, flooding, and tornadoes.

Nonsafety Grade, Nonredundant Initiation Signals: The automatic start signals and associated circuitry are safety grade. The details of the design are discussed in Section 7 of this SER.

In its evaluation, the staff concludes that the AFWS meets the recommendations of NUREG-0611 pending satisfactory review of plant Technical Specifications for outage time limits flow test after extended cold shutdown and the 48-hour endurance test and plant emergency procedures for initiation of backup alternate water supply and verification of valve alignment after testing and maintenance.

NUREG-0737, Item II.E.1.1, requires that a reliability analysis of the AFWS be performed. The applicant has provided the results of the evaluation in the FSAR for only the following transients:

(1) loss of main feedwater - mean unavailability 6.8×10^{-5}

(2) loss of main feedwater due to loss of offsite power - mean unavailability 6.8×10^{-5}

(3) loss of main feedwater and loss of all ac power (station blackouts) - mean unavailability 4.52×10^{-2}

The applicant has submitted the AFW reliability analysis as part of the Millstone Unit 3 probabilistic safety study. The staff has reviewed this analysis and also performed an independent evaluation based on NUREG-0611 models and data. The results are given below:

(1) loss of main feedwater event	1.5×10^{-5}/demand
(2) loss of offsite power	1.0×10^{-4}/demand
(3) loss of all ac power	2.0×10^{-2}/demand

The differences between the applicant's and the staff's results are due to methodology and failure data. Despite the differences in methodology and failure data between the applicant's and the NUREG-0611 analysis, the staff believes that the unavailability of the Millstone Unit 3 AFWS satisfies the acceptance criterion of SRP Section 10.4.9 and the requirements of NUREG-0611.

On the basis of this review, the staff concludes that the AFWS complies with the requirements of GDC 2, 5, 19, 34, 44, 45, and 46 with regard to protection against natural phenomena, AFWS sharing between units, capability to aid in shutdown with system operated from the control room, decay heat removal, cooling water capability, and inspection and testing of the AFWS; the guidelines of

RG 1.29 and BTPs ASB 10-1 and ASB 10-2 concerning seismic classification, power diversity; and the recommendations of NUREG-0611 and NUREG-0737 concerning generic improvements to the AFWS design, reliability, and (pending confirmatory review) the procedures and Technical Specifications. However, further information is required to determine compliance with GDC 4 as it relates to internally generated missiles (see Section 3.5.1.1). The AFWS meets the acceptance criteria of SRP Section 10.4.9 except as noted above.

11 RADIOACTIVE WASTE MANAGEMENT

The radioactive waste management systems for Millstone Unit 3 are designed to provide for the controlled handling and treatment of liquid, gaseous, and solid wastes. The radioactive waste management systems are not shared among Units 1, 2, and 3. The liquid radioactive waste management system processes wastes from equipment and floor drains, sample wastes, decontamination and laboratory wastes, and chemical regeneration wastes. The gaseous radioactive waste management system provides (1) charcoal bed adsorbers to adsorb radioiodines and holdup to allow decay of short-lived noble gases and (2) treatment of ventilation exhausts through high-efficiency particulate air (HEPA) filters and carbon adsorbers, as necessary, to reduce releases of radioactive materials to as low as is reasonably achievable (ALARA) levels in accordance with 10 CFR 20 and 50.34a. The solid radioactive waste management system uses Dow binder for solidification. The radioactive waste management review area also includes the process and effluent radiological monitoring and sampling system provided for the detection and measurement of radioactive materials in plant process and effluent streams.

The staff has reviewed the applicant's design, design criteria, and design bases for the radioactive waste management systems for Millstone Unit 3. The acceptance criteria used as the basis for staff evaluation are in Section II of SRP Sections 11.1, 11.2, 11.3, 11.4, and 11.5 (NUREG-0800). These acceptance criteria include the applicable general design criteria (GDC) (Appendix A to 10 CFR 50), 10 CFR 20.106, Appendix I to 10 CFR 50, and American National Standards Institute (ANSI) Std. N13.1, "Guide to Sampling Airborne Radioactive Materials in Nuclear Facilities." Guidelines for implementation of the requirements of the acceptance criteria are provided in the ANSI standards, regulatory guides, and other documents identified in SRP Section II. Conformance to the acceptance criteria provides the bases for concluding that the radioactive waste management systems meet the requirements of 10 CFR 20 and 50.

11.1 Source Terms

The applicant provided the expected annual radioactive liquid releases from Millstone Unit 3 in FSAR Table 11.2-6. The staff has performed an independent calculation of the primary and secondary coolant concentrations and of the release rates of radioactive materials using the information supplied in the FSAR, the GALE computer program, and the methodology presented in NUREG-0017. Table 11.1 presents the principal parameters that were used in this independent calculation of source terms. These source terms were used to calculate individual doses in Sections 11.2 and 11.3 for Millstone Unit 3, in accordance with the mathematical models and guidance contained in RG 1.109, "Calculation of Annual Average Doses to Man From Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance With 10 CFR Part 50, Appendix I."

11.2 Liquid Waste Management System

11.2.1 System Description and Review

The liquid radioactive waste (radwaste) management system consists of process equipment and instrumentation necessary to collect, process, monitor, and recycle or dispose of radioactive liquid wastes from the operation of Millstone Unit 3. The liquid radwaste system is designed to collect and process wastes according to the source, activity, and composition of the fluids. All liquid waste will be processed on a batch basis to permit optimum control and disposal of radioactive waste. Before the waste is released, samples will be analyzed to determine the types and amounts of radioactivity present. On the basis of the results of the analyses, the waste will be recycled for eventual reuse in the plant, retained for further processing, or released under controlled conditions from the discharge line to the circulating water tunnel, which, in turn, discharges to the Niantic Bay. A radiation monitor in the discharge line will automatically terminate liquid waste discharges if radiation measurements exceed a predetermined level.

The liquid radioactive waste management system consists of the high-level waste, the low-level waste, the regenerant chemical waste, and the boron recovery subsystems. The systems are described in detail in Section 11.2 of the FSAR.

The high-level waste subsystem processes the low-conductivity, high-purity wastes. Wastes will be processed through a waste evaporator and a mixed-bed demineralizer in series, or through two mixed-bed demineralizers in series. The staff estimated that the equipment drain subsystem waste input flow will be approximately 1,340 gpd and assumed that 10% of the treated process stream will be released to the Niantic Bay through the circulating water system. The remainder will be recycled to the primary grade water storage tanks for eventual reuse within the plant. The design capacity of the equipment drain subsystem is 50,400 gpd. The difference between the expected flow and design flow provides adequate reserve for processing surge flows.

The low-level waste subsystem processes high-conductivity, low-purity wastes and includes turbine building floor drains. The wastes will be processed through a cartridge-type filter. The staff estimated that the low-level waste subsystem waste input flow, including turbine building floor drain flow, will be approximately 7,240 gpd and assumed that all of the treated process stream will be released to the environment. The design capacity of the low-level waste subsystem is 72,000 gpd. Turbine building floor drain flows are normally discharged directly to the environment. When radioactivity in the discharge line is detected, the floor drain flow is diverted to the low-level waste subsystem for processing. The difference between the expected flow and design flow provides adequate reserve for processing surge flows.

The regenerant chemical waste subsystem processes the regenerant chemical wastes. Wastes will be processed through a radwaste evaporator and a mixed-bed demineralizer. The staff estimated that the regenerant chemical waste input flow will be approximately 3,400 gpd and assumed that 10% of the treated process streams will be released to the environment. The design capacity of the evaporator is 50,400 gpd. The difference between the expected flow and design flow provides adequate reserve for processing surge flows.

The boron recovery subsystem processes reactor coolant bleed. The bleeds will be processed through a degasifier, a cation demineralizer, and the boron evaporator. The staff estimated that the boron recovery subsystem waste input will be approximately 1,440 gpd and assumed that 10% of the processed reactor bleed will be released to the environment. The design capacity of the subsystem is 36,000 gpd. The difference between the expected flow and design flow provides adequate reserve for processing surge flows.

In its evaluation of the liquid radioactive waste management system, the staff considered (1) the capability of the system to maintain releases below the limits in 10 CFR 20 during periods of fission product leakage (at design levels) from the fuel, (2) the capability of the system to meet the ALARA criterion in accordance with 10 CFR 50, Appendix I, Sections II.A and II.D, (3) the system design objectives for equipment necessary to control releases of radioactive effluents to the environment in accordance with 10 CFR 50.34a, (4) the system design to ensure adequate safety under normal and postulated accident conditions in accordance with GDC 61, and (5) the design features that are incorporated to control and monitor the releases of radioactive materials in accordance with GDC 60 and 64.

The estimated releases of radioactive materials in liquid effluents were calculated using the PWR-GALE Code described in NUREG-0017. The PWR-GALE Code is a computerized, mathematical model for calculating the routine releases of radioactive material in effluents from pressurized-water reactors (PWRs). The basic code has been in use since 1976 for all PWR licensing reviews. The calculations in the code are based on (1) data generated from operating reactors, (2) field and laboratory tests, (3) standardized coolant activities derived from American Nuclear Society (ANS) 18.1 Working Group recommendations, (4) release and transport mechanisms that result in the appearance of radioactive material in liquid streams, and (5) the Millstone Unit 3 radwaste system design features used to reduce the quantities of radioactive materials ultimately released to the environment. The principal parameters used in these calculations are given in Table 11.1 of this SER. The liquid source term is given in Table 11.2 of this SER.

11.2.2 Evaluation Findings

The liquid radwaste system includes the equipment necessary to control the releases of radioactive materials in liquid effluents in accordance with GDC 60 and 64. Capacities of principal components considered in the liquid waste processing system evaluation are listed in Table 11.3. The staff concludes that the design of the liquid waste management system is acceptable and meets the requirements of 10 CFR 20.106, 10 CFR 50.34a, Appendix I of 10 CFR 50, and GDC 60, 61, and 64, as referenced in the SRP. This conclusion is based on the following:

(1) The applicant has met the requirements of 10 CFR 20.106. The staff has considered the potential consequences resulting from reactor operation and has determined that the concentrations of radioactive materials in liquid effluents in unrestricted areas will be a small fraction of the limits in 10 CFR 20, Appendix B, Table II, Column 2.

(2) The applicant has met the requirements of Section II.A of Appendix I of 10 CFR 50 with respect to dose-limiting objectives by proposing a liquid

radwaste treatment system that is capable of maintaining releases of radioactive materials in liquid effluents so that the calculated individual doses in an unrestricted area from all pathways of exposure are less than 3 mrems to the total body and 10 mrems to any organ. In its evaluation, the staff considered releases of radioactive materials in liquid effluents for normal operation, including anticipated operational occurrences, based on expected radwaste inputs over the life of the plant for Millstone Unit 3 in accordance with SRP Section 11.1.

The applicant has also met the requirements of Section II.D of Appendix I of 10 CFR 50 with respect to meeting the ALARA criterion. The staff has considered the potential effectiveness of augmenting the proposed liquid radwaste treatment systems using items of reasonably demonstrated technology and has determined that further effluent treatment will not effect reductions in the cumulative population dose reasonably expected within a 50-mi radius of the reactor at a cost of less than $1,000 per man-rem or man-thyroid-rem.

(3) The staff has reviewed the applicant's quality assurance provisions for the liquid radwaste systems, the quality group classifications used for system components, and the seismic design applied to structures housing these systems. The design of the systems and structures housing these systems meets the intent of the criteria given in RG 1.143. The staff has reviewed the provisions incorporated in the applicant's design to control the release of radioactive materials in liquids resulting from inadvertent tank overflows and concludes that the measures proposed by the applicant are consistent with the criteria given in RG 1.143.

(4) The applicant has met the requirements of GDC 60, 61, and 64 with respect to controlling and monitoring the releases of radioactive material to the environment. The staff has considered the capabilities of the proposed liquid radwaste treatment system to meet the demands of the plant resulting from anticipated operational occurrences and has concluded that the system's capacity and design flexibility are adequate to meet the anticipated needs of the plant.

11.3 Gaseous Waste Management System

11.3.1 System Description and Review

The gaseous radioactive waste (radwaste) processing and plant ventilation systems are designed to collect, store, process, monitor, and discharge potentially radioactive gaseous wastes that are generated during normal operation of the plant. The systems consist of equipment and instrumentation necessary to reduce releases of radioactive gases and particulates to the environment. The principal sources of gaseous waste are the effluents from the gaseous waste processing (offgas) system and ventilation exhausts from the containment, auxiliary, and fuel buildings. The offgas system collects, processes, and stores fission product gases removed from the reactor coolant letdown by processing them through a degasifier, precooler, and glycol chiller, and then by adsorption and decay through charcoal adsorber tanks. After a delay, the gases pass through a HEPA filter and are discharged to the environment through the Millstone Unit 1 stack.

Ventilation exhaust air from the containment building is exhausted by three paths. The exhaust normally is not treated; however, the exhaust through one path can be treated. The principal exhaust path is the 35,000 ft^3/min purge path for the reactor building. The containment purge exhaust is not normally treated, but is provided with a connection to the auxiliary building engineered safety features (ESF) filtration system to filter the exhaust if radioactivity above a preset level is detected. The containment vacuum pump exhaust is not treated and is discharged after monitoring for radioactivity through the Millstone Unit 1 stack.

During plant startups after cold shutdown (refueling and extended plant maintenance), the containment vacuum steam ejector, operating with the plant auxiliary steam, is used to achieve an initial negative pressure before switching to the containment vacuum pumps. The containment vacuum steam ejector exhaust is neither treated nor in-line monitored for radioactivity. However, the containment air is continuously monitored for radioactivity before and during the operation of the vacuum steam ejector exhaust system.

The auxiliary and ESF building exhausts are normally discharged without treatment to the atmosphere through the Millstone Unit 1 stack after being monitored for radioactivity. These building exhausts are provided with a connection to the auxiliary building ESF filtration system to filter the exhausts if radioactivity above a preset level is detected. The turbine building exhaust is discharged without treatment.

Warehouse no. 5 at Millstone Unit 3 houses chemical radwaste process equipment for the condensate demineralizer liquid radwaste subsystem, including the chemical regenerant evaporator, evaporator feed tanks, chemical regenerant sumps, and regenerant demineralizer and filter, which are all vented to the rooms. The ventilation exhaust for warehouse no. 5 is discharged through the building roof vent without being monitored for radioactivity.

The applicant has committed to monitor airborne radioactivity (noble gases, iodines, and particulates) in this pathway with a continous air monitor (Eberline PING-3 or equivalent) on detection of radioactivity in the regenerant waste. The radioactivity concentration action levels in the regenerant waste to initiate continuous monitoring will be specified in the Millstone Unit 3 Technical Specifications.

In its evaluation of the gaseous radwaste management system, the staff considered the following SRP criteria: (1) the capability of the system to meet the processing demands of the station during anticipated operational occurrences, (2) the quality group and seismic design classification applied to the equipment and components and structures housing the system, (3) the design features that are incorporated to control and monitor the releases of radioactive materials in accordance with GDC 60 and 64, and (4) the potential for gaseous releases resulting from hydrogen explosion in the gaseous radwaste system, and (5) the capability of the system design to meet the ALARA criterion in accordance with 10 CFR 50, Appendix I, Sections II.B, II.C, and II.D.

The estimated releases of radioactive materials in gaseous effluents were calculated using the PWR-GALE Code described in NUREG-0017. The PWR-GALE Code is a computerized mathematical model for calculating the routine releases of radioactive material in effluents from PWRs. The basic code has been in use since

1976 for all PWR licensing reviews. The calculations in the code are based on (1) data generated from operating reactors, (2) field and laboratory tests, (3) standardized coolant activities derived from ANS 18.1 Working Group recommendations, (4) release and transport mechanisms that result in the appearance of radioactive material in gaseous streams, and (5) the Millstone Unit 3 radwaste system design features used to reduce the quantities of radioactive materials ultimately released to the environment. The principal parameters used in these calculations are given in Table 11.1 of this SER. The gaseous source term is given in Table 11.4.

The staff has reviewed the applicant's quality assurance provisions for the gaseous radwaste system, the quality group classifications used for system components, the seismic design criteria applied to the design of the system and structures housing the radwaste system. The design of the system and structures housing this system meets the intent of the criteria given in RG 1.143 and referenced in the SRP.

The staff has reviewed the provisions incorporated in the applicant's design to control releases resulting from hydrogen explosions in the gaseous radwaste system and concludes that the measures proposed by the applicant are adequate to prevent the occurrence of an explosion.

The staff has reviewed the provisions incorporated in the applicant's design to control and monitor radioactive materials in the normal ventilation exhaust systems during normal plant operation, including anticipated operational occurrences, and concludes that the system design is adequate to control and monitor airborne radioactivity.

11.3.2 Evaluation Findings

The staff concludes that the design of the gaseous waste management system is acceptable and meets the requirement of 10 CFR 20.106; 10 CFR 50.34a; GDC 3, 60, 61, and 64; and 10 CFR 50, Appendix I, as referenced in the SRP. This conclusion is based on the following findings:

(1) The applicant has met the requirements of GDC 60 and 64 with respect to controlling releases of radioactive material to the environment by ensuring that the design of the gaseous waste management system includes the equipment and instruments necessary to detect and control the release of radioactive materials in gaseous effluents. Capacities of principal components considered in the gaseous waste processing system evaluation are listed in Table 11.3.

(2) The applicant has met the requirements of Appendix I of 10 CFR 50 by meeting the ALARA criterion as follows:

   (a) Regarding Sections II.B and II.C of Appendix I, the staff has considered releases of radioactive material (noble gases, radioiodines, and particulates) in gaseous effluents for normal operation, including anticipated operational occurrences, based on expected radwaste inputs over the life of the plant. The staff has determined that the proposed gaseous waste management system is capable of limiting releases of radioactive materials in gaseous effluents so that the calculated individual doses from releases of radioiodine and radioactive material in particulate form in an unrestricted area from all pathways of exposure are less than 5 mrems to the total body, 15 mrems to the skin, and 15 mrems to any organ.

(b) Regarding Section II.D. of Appendix I, the staff has considered the potential effectiveness of augmenting the proposed gaseous waste management system using items of reasonably demonstrated technology and has determined that further effluent treatment will not effect reductions in the cumulative population dose within a 50-mi radius of the reactor at a cost of less than $1,000 per man-rem or man-thyroid-rem.

(3) The applicant has met the requirements of 10 CFR 20. The staff has considered the potential consequences resulting from reactor operation and determined that the concentrations of radioactive materials in gaseous effluents in unrestricted areas will be a small fraction of the limits specified in 10 CFR 20, Appendix B, Table II, Column 1.

(4) The staff has considered the capabilities of the proposed gaseous waste management system to meet the demands of the plant resulting from anticipated operational occurrences and has concluded that the system's capacity and design flexibility are adequate to meet these demands.

(5) The staff has reviewed the applicant's quality assurance provisions for the gaseous waste management system, the quality group classifications used for system components, and the seismic design applied to the design of the system and of structures housing the radwaste system. The design of the system and of structures housing the system meets the criteria given in RG 1.143.

(6) The staff has reviewed the provisions incorporated in the applicant's design to control releases resulting from hydrogen explosions in the gaseous waste management system and concludes that the measures proposed by the applicant are adequate to prevent the occurrence of an explosion in accordance with GDC 3.

11.4 Solid Waste Management System

11.4.1 System Description and Review

The solid waste management system (SWMS) consists of equipment and instrumentation necessary for the solidification or packaging of radioactive wastes resulting from operation of the reactor water letdown purification system, the condensate demineralizer system, the liquid and gaseous radwaste systems, and the miscellaneous debris resulting from normal operation and maintenance of the plant.

The SWMS is designed to process two general types of solid wastes: (1) "wet" solid wastes, which require solidification or dewatering, processing, and packaging before being shipped and (2) "dry" solid wastes, which require packaging and, in some cases, compaction before being shipped to a licensed burial facility.

Compressible dry solid wastes, consisting mainly of items such as ventilation air filters, contaminated clothing, paper, laboratory glassware, and tools, will be compacted in 55-gal drums by a waste compacter. The compacter is equipped with ventilation to control the emission of contaminated particles during the compaction process. Noncompressible wastes will be packaged manually in 55-gal drums or other suitable containers.

Wet solid wastes consist primarily of spent bead resins and evaporator bottoms. The resins will be dewatered before they are packaged in high-integrity containers for shipment and disposal. Any concentrated bottoms produced by the evaporators will be solidified with Dow binder by a portable solidification system before being shipped off site.

The portable solidification system consists of a fill head, a control panel, spent resin dewatering and hold tanks, a binder storage tank, and the piping, valves, pumps, and instruments required for transfer, dewatering, or solidification of the wet wastes. The applicant stated that the current solidification process calls for the use of Dow binder (vinyl ester resin) as referenced in Dow Chemical Company Topical Report, DNS-RSS-001P, which was approved by the staff in May 1980. Section 11.4 of the FSAR contains a detailed description of the SWMS.

The review of the SWMS, which was conducted in accordance with the SRP, included line diagrams of the system, piping and instrumentation diagrams (P&IDs), and descriptive information on the SWMS and those auxiliary supporting systems that are essential to its operation. The applicant's proposed design criteria and design bases for the SWMS and the applicant's analysis of those criteria and bases were reviewed and compared with those of the SRP. The staff also reviewed (1) the capability of the proposed system to process the types and volumes of wastes expected during normal operation and anticipated operational occurrences in accordance with GDC 60, (2) the provisions for the processing and packaging of wastes relative to the requirements of 10 CFR 20, 61, and 71 and applicable Department of Transportation (DOT) regulations, (3) the applicant's quality group classification and seismic design relative to RG 1.143, and (4) provisions for onsite storage before shipment. The basis for acceptance in the staff's review was conformance of the applicant's designs, design criteria. and design bases for the solid radwaste management system to the regulations, guides, staff technical positions, and industry standards referenced in the SRP.

11.4.2 Evaluation Findings

The annual production of solid wastes is estimated to be approximately 21,000 ft^3 of wet wastes, containing approximately 6,000 Ci of activity, and 3,500 ft^3 of compacted dry wastes. The applicant has provided minimum storage space for approximately 2 month's capacity of waste at the annual average generation rate. Because the staff's guidance specifies storage space for 1 month's capacity of waste, the staff finds the storage volume adequate for meeting the demands of the plant.

In Section 11.4 of the FSAR, the applicant stated that a process control program to ensure that the solidified and packaged wastes are suitable for shipment and disposal in accordance with Federal and state regulations and in conformance with the guidelines of SRP Section 11.4 (NUREG-0800) will be submitted at least 6 months before fuel loading. The applicant further stated, during review discussions with the staff, that Section 11.4 of the FSAR will be revised to indicate that a compliance program to meet 10 CFR 20.311 and 10 CFR 61 for waste classification stability requirements will also be submitted at least 6 months before fuel loading. The staff finds the applicant's proposals acceptable. These are confirmatory items.

The staff concludes that the design of the solid waste management system is acceptable and meets the requirements of 10 CFR 20.106; 10 CFR 50.34a; GDC 60, 63, and 64; and 10 CFR 71, as referenced in SRP Section 11.4. This conclusion is based on the applicant demonstrating that the SWMS includes the equipment and instrumentation used for the processing, packaging, and storing of radioactive wastes before shipment off site for burial.

The basis for acceptance in the staff's review has been conformance of the applicant's designs, design criteria, and design bases for the solid radwaste system to the regulations and guides referenced above and in SRP Section 11.4, as well as to staff technical positions and industry standards. On the basis of the foregoing evaluation and the condition that the applicant provide an acceptable process control program, which includes a compliance program to meet 10 CFR 61, the staff concludes that the proposed solid radwaste system is acceptable.

When a process control program and compliance program to meet 10 CFR 20.311 and 10 CFR 61 for waste classification and waste stability requirements are received from the applicant 6 months before fuel loading, the staff will perform the review and its evaluation will be provided in a supplement to the SER.

11.5 Process and Effluent Radiological Monitoring and Sampling Systems

11.5.1 System Description and Review

The process and effluent radiological monitoring systems are designed to provide information concerning radioactivity levels in systems throughout the plant, indicate radioactive leakage between systems, monitor equipment performance, and monitor and control radioactivity levels in plant discharges to the environment.

Table 11.5 provides the proposed locations of continuous monitors. Monitors on certain effluent release lines will automatically terminate discharges if radiation levels exceed a predetermined value. Systems that are not amenable to continuous monitoring or for which detailed isotopic analyses are required will be periodically sampled, and the samples will be analyzed in the plant laboratory. The potential airborne radioactive releases to the environment from Millstone Unit 3 are from the following release points: (1) plant vent, (2) Millstone Unit 1 stack, (3) ESF building, (4) warehouse no. 5 roof vent, (5) condenser mechanical vacuum pump discharge, (6) turbine gland seal steam noncondensible gas release, and (7) containment vacuum steam ejector release. The radioactivity monitoring provisions for Items (4), (5), (6), and (7) are discussed in Sections 11.3, 10.4.2, 10.4.3, and 11.3 of this SER, respectively.

The plant ventilation exhaust radiation monitoring system is designed to (1) provide continuous isokinetic and representative samples from the plant vent exhaust so that releases of radioactive particulates, iodines, and noble gases can be continuously indicated and recorded, (2) alarm if specified rates of release of radioactive material are exceeded, and (3) provide real-time indications of radioactive releases during postaccident modes.

The airborne radioactive releases through the Millstone Unit 1 stack are monitored for radioactivity by a supplementary leak collection and release system (SLCRS) monitor located upstream from the stack. The monitoring system is identical to the plant ventilation monitoring system. The monitor outputs are indicated, recorded, and alarmed in the main control room. The plant vent and SLCRS monitors are safety-related monitors and use the safety-related Class 1E buses. These monitoring systems use normal and high-range monitors and are capable of monitoring all postulated accident releases.

The ESF building exhaust is normally monitored by a nonsafety monitor before discharge into the atmosphere. Following an accident, this monitoring system is secured and the ESF building is ventilated by the SLCRS and monitored by the SLCRS monitor.

The potential radioactive liquid effluent release points to the circulating water discharge tunnel are (1) the processed liquid radwaste discharge, (2) the condensate demineralizer waste neutralization sump discharge, and (3) the turbine building sump discharge. All release points are continuously monitored for radioactivity before discharge. The liquid radwaste and neutralization sump discharges are automatically terminated if radioactivity concentration reaches a preset value. The turbine building sump discharge is diverted to the liquid radwaste system on detection of radioactivity in the sump discharge.

The main steam relief line monitors, located in the main steam valve building, monitor radioactivity concentrations in the main steam relief lines if a main steam relief valve lifts. The monitors' output is transmitted, indicated, and recorded by their dedicated microprocessors in the radiation monitoring system console located in the main control room.

11.5.2 Evaluation Findings

The staff concludes that the process and effluent radiological monitoring instrumentation and sampling systems for the liquid and solid radwastes are acceptable and meet the relevant requirements of 10 CFR 20.106 and GDC 60, 63, and 64. The process and effluent radiological monitoring and sampling systems for the liquid and solid radwastes include the instrumentation for monitoring and sampling radioactivity in contaminated liquid and solid waste process and effluent streams. The staff's review included (1) the provisions proposed to sample and monitor all liquid effluents in accordance with GDC 64; (2) the provisions proposed to provide automatic termination of liquid effluent releases and ensure control over discharges in accordance with GDC 60; (3) the provisions proposed for sampling and monitoring plant waste process streams for process control in accordance with GDC 63; (4) the provisions for conducting sampling and analytical programs in accordance with the guidelines in RGs 1.21 and 4.15; and (5) the provisions for sampling and monitoring process and effluent streams during postulated accidents in accordance with the guidelines in RG 1.97, Revision 2. The review included P&IDs and process flow diagrams for the liquid, gaseous, and solid radwaste systems and ventilation systems, and the location of monitoring points relative to effluent release points shown on the site plot diagrams.

On the basis of its review, the staff has determined that the applicant's designs, design criteria, and design bases for the process and effluent radiological monitoring instrumentation and sampling systems for the liquid and solid radwastes meet the guidelines of SRP Appendix 11.5-A and industry standards and concludes that the systems are acceptable.

Item II.F.1 Attachment 1, Noble Gas Effluent Monitor

The high range noble gas effluent monitoring system will be installed with an extended range designed to function during accident conditions, as well as normal operating conditions, at Millstone Unit 3. The monitors will be located in the plant vent exhaust, the ventilation inlet line to the Millstone Unit 1 stack, and the main steam relief exhaust lines.

The plant vent and the stack ventilation exhaust monitors are multistage, gaseous monitors, which consist of two detectors with overlapping ranges (10^{-7} to 10^{5} μCi/cc). The low range monitors use a beta scintillation detector (10^{-7} to 10^{-2} μCi/cc), and high range monitors use Geiger-Mueller (GM) solid-state detectors (10^{-3} to 10^{5} μCi/cc). The main steam relief monitor consists of GM tubes (10^{-1} to 10^{3} μCi/cc) viewing the main steamlines. Power supply to these monitors is from safety-related Class 1E buses. The outputs from these monitors are indicated, recorded, and alarmed in the main control room.

On the basis of its evaluation, the staff concludes that the high range noble gas monitoring systems to be installed at Millstone Unit 3 meet the requirements of Items (1), (2), (3), and (4) and TMI Action Plan Item II.F.1 and Table II.F.1-1, under discussion and clarification, and meet the intent of the guidelines in RG 1.97, Revision 2. However, the applicant has not provided the monitor calibration method, detector energy response characteristics, and calculational method to be used for converting instrument readings to release rate as a function of time after an accident. The staff requires that the applicant submit this information for the staff's review at least 6 months before fuel loading. This is a confirmatory item.

Item II.F.1.2 Attachment 2, Sampling and Analysis of Plant Effluents

The continuous sampling of gaseous effluents for postaccident releases of radioactive iodines and particulates and onsite laboratory capabilities to analyze these samples will be provided at Millstone Unit 3. These sampling capabilities are an integral part of the high range noble gas monitoring systems discussed under Item II.F.1-1.

The particulate filter and the iodine sample media will be housed in a lead shield, mounted for ease of removal and replacement of filter media, and capable of being transported to the onsite analysis facility during accident conditions with radiation exposures to the operator less than those of GDC 19. On the basis of its review, the staff concludes that the radioactive iodines and particulate sampling and analysis capabilities to be installed at Millstone Unit 3 meet the requirements of Items (1), (2), (3), and (4) of Task Action Plan Item II.F.1.2 and Table II.F.1-2, under discussion and clarification, and therefore, the staff finds the proposed system acceptable.

Table 11.1 Principal parameters and conditions used in calculating releases of radioactive material in liquid and gaseous effluents from Millstone Unit 3

Parameter	Value
Reactor power level (MWt)	3,636
Plant capacity factor	0.80
Failed fuel (%)	0.12*
Primary system	
Mass of coolant (lb)	4.7×10^5
Letdown rate (gpm)	75
Shim bleed rate (gpd)	1.74×10^3
Leakage to secondary system (lb/day)	100
Leakage to containment building (lb/day)	**
Leakage to auxiliary building (lb/day)	160
Frequency of degassing for cold shutdowns (times/yr)	2
Letdown cation demineralizer flow (gpm)	7.5
Secondary system	
Steam flow rate (lb/hr)	1.59×10^7
Mass of liquid/steam generator (lb)	1.03×10^5
Mass of steam/steam generator (lb)	8×10^3
Secondary coolant mass (lb)	1.6×10^6
Rate of steam leakage to turbine area (lb/hr)	1.7×10^3
Containment building volume (ft^3)	2.32×10^6
Frequency of containment purges (times/yr)	8
Containment low volume purge rate (ft^3/min)	0
Iodine partition factors (gas/liquid)	
Leakage to auxiliary building	0.0075
Leakage to turbine area	1.0
Main condenser/air ejector (volatile species)	0.15

LIQUID RADWASTE SYSTEM DECONTAMINATION FACTORS

Material	Boron recycle system	Radwaste system
Iodine	1×10^3	1×10^4
Cesium	1×10^4	1×10^5
Other	1×10^4	1×10^5

*See footnotes at end of table.

Table 11.1 (Continued)

INDIVIDUAL EQUIPMENT DECONTAMINATION FACTORS

(1) Evaporator

System	All nuclides except iodine	Iodine
Radwaste evaporator	10^4	10^3
Boron evaporator	10^3	10^2

(2) Demineralizers

System	Anions	Cesium, rubidium	Other nuclides
Letdown cation demineralizer	1	10	10
Letdown mixed-bed demineralizer	10	2	10
Boron recycle evaporator condensate demineralizer	10	10	10
Radwaste mixed-bed demineralizer	10	10	10

LIQUID WASTE INPUTS

Stream	Flow rate (gpd)	Fraction of PCA	Fraction discharged	Collection time (days)	Decay time (days)
Shim bleed rate	1,740	1.0	0.1	69	0.3
Clean wastes	1,340	0.077	0.1	15	12.8
Dirty wastes	40	0.05	1.0	80	0
Chemical wastes	3,400	-	0.1	3.2	0.2

GASEOUS WASTE INPUTS***

Holdup time for xenon (days)	140
Holdup time for krypton (days)	7.85
Fill time of decay tanks (days)	0

*This value is constant and corresponds to 0.12% of the operating power product source term as given in NUREG-0017 (April 1976).

**1%/day of the primary coolant noble gas inventory and 0.001%/day of the primary coolant iodine inventory.

***There is no continuous stripping of full letdown flow.

Table 11.2 Calculated releases of radioactive materials in liquid effluents from Millstone Unit 3

Nuclide	Value (Ci/yr/reactor)	Nuclide	Value (Ci/yr/reactor)
Activation-corrosion products		Fission products (continued)	
Cr-51	3.3(-3)*	Te-127m	8.7(-4)
Mn-54	1.1(-30)	Te-127	8.9(-4)
Fe-55	6.2(-3)	Te-129m	2.8(-3)
Fe-59	2.3(-3)	Te-129	1.8(-3)
Co-58	4.4(-2)	I-130	9(-5)
Co-60	7.8(-3)	Te-131m	2.4(-4)
Zr-95	1.6(-4)	Te-131	4(-5)
Nb-95	1.9(-4)	I-131	1.6(-1)
Np-239	2.1(-4)	Te-132	6.4(-3)
		I-132	7.3(-3)
Fission products		I-133	2.6(-2)
		I-134	1.5(-4)
Br-83	4(-5)	Cs-134	9.6(-2)
Rb-86	1.1(-4)	I-135	4.5(-3)
Rb-88	2.1(-4)	Cs-136	1.2(-2)
Sr-89	8.6(-4)	Cs-137	7.2(-2)
Sr-90	4(-5)	Ba-137m	6.7(-2)
Y-90	4(-5)	Ba-140	2(-4)
Sr-91	2(-5)	La-140	2.2(-4)
Y-91m	1(-5)	Ce-141	1.3(-4)
Y-91	1.8(-4)	Pr-143	5(-5)
Mo-99	1.7(-2)	Ce-144	1.2(-4)
Tc-99m	1.6(-2)	Pr-144	1.2(-4)
Ru-103	1(-4)		
Ru-103m	1(-4)	All others**	1(-5)
Ru-106	4(-5)		
Rh-106	4(-5)	Total (except H-3)	5.6(-1)
Te-125m	7(-5)	H-3	250

*Exponential notation: 2.6(-4) = 2.6×10^{-4}.

**Nuclides whose release rates are less than 10^{-5} Ci/yr/reactor are not listed individually but are included in the category "All others."

Table 11.3 Design parameters of principal components considered in the evaluation of liquid and gaseous radioactive waste treatment systems for Millstone Unit 3

Component	Number	Capacity, each
LIQUID SYSTEMS*		
High-Level Waste Subsystem		
Waste drain tank	2	25,000 gal
Filter	2	35 and 200 gpm
Demineralizer	2	35 gpm
Evaporator	1	35 gpm
Waste distillate tank	1	500 gal
Waste bottom holding tank	1	3,200 gal
Waste test tank	2	21,000 gal
Effluent filter	1	50 gpm
Low-Level Waste Subsystem		
Waste drain tank	2	4,000 gal
Effluent filter	1	50 gpm
Chemical Waste Subsystem		
Evaporator feed tank	2	14,000 gal
Evaporator	1	50 gpm
Distillate tank	1	500 gal
Demineralizer	1	50 gpm
Filter	1	50 gpm
GASEOUS SYSTEMS*		
Degasifier	1	150 gpm
Charcoal bed adsorber	2	27,000 lb
Gas compressor	2	3 scfm
Prefilter	1	3 scfm
SOLID SYSTEMS*		
Spent resin tank	1	3,200 gal
Spent resin dewatering tank	1	500 gal
Dow binder storage tank	1	6,000 gal
Shipping containers	As required	50-300 ft^3

*Quality group and seismic design in accordance with RG 1.143.

Table 11.4 Calculated releases of radioactive materials in gaseous effluents from Millstone Unit 3 (Ci/yr/reactor)

	To ventilation vent (133 ft above grade)			To Unit 1 stack (395 ft above grade)		
Nuclide	Reactor building (intermittent)	Auxiliary building (continuous)	Turbine building (continuous)	Waste gas system (continuous)	Air ejector system (continuous)	Total
Ar-41	25	a	a	a	a	25
Kr-83m	a	a	a	a	a	a
Kr-85m	a	2	a	a	1	3
Kr-85	1	a	a	260	a	260
Kr-87	a	1	a	a	a	1
Kr-88	a	4	a	a	2	6
Kr-89	a	a	a	a	a	a
Xe-131m	1	a	a	a	a	1
Xe-133m	2	a	a	a	a	2
Xe-133	220	36	a	a	22	280
Xe-135m	a	a	a	a	a	a
Xe-135	2	5	a	a	3	10
Xe-137	a	a	a	a	a	a
Xe-138	a	1	a	a	a	1
Total Noble Gases						560*
Mn-54	0.00082	0.018	b	0.0045	b	0.023
Fe-59	0.00028	0.006	b	0.0015	b	0.0078
Co-58	0.0028	0.06	b	0.015	b	0.078
Co-60	0.0013	0.027	b	0.007	b	0.035
Sr-89	0.000063	0.0013	b	0.00033	b	0.0017
Sr-90	0.000011	0.00024	b	0.00006	b	0.0031
Cs-134	0.00082	0.018	b	0.0045	b	0.023
Cs-137	0.0014	0.03	b	0.0075	b	0.039
Total Particulates						0.21
I-131	0.0034	0.046	0.00033	a	0.029	0.079
I-133	0.0016	0.067	0.00046	a	0.042	0.11
H-3	a	1200	a	a	a	1,200
C-14	1	a	a	7	a	8

*Sum is truncated.
a = less than 1.0 Ci/yr for noble gases and C-14, less than 10^{-4} Ci/yr for iodine.
b = less than 1% of total for this nuclide.

Table 11.5 Process and effluent monitors

Monitor	Number of channels	Medium	Sensitivity (μCi/cc)*	Range (decades)	Location
Ventilation vent		Air			Auxiliary bldg. 66 ft 6 in.
Normal range	1		1.0E-7 (Xe-133)	6	
High range	1		5.0E-02 (Xe-133)	6	
Hydrogenated vent	1	Gas	1.0E-03 (Kr-85)	5	Auxiliary bldg. 43 ft 6 in.
Fuel drop	2	Air	**	6	Containment 51 ft 4 in.
Supplementary leak collection		Air		5	Auxiliary bldg. 66 ft 6 in.
Normal range	1		1.0E-07 (Xe-133)	6	
High range	1		5.0E-02 (Xe-133)	6	
Condenser air ejector	1	Vapor	1.0E-06 (Xe-133)	5	Turbine bldg. 38 ft 6 in.
Control building inlet ventilation	2	Air	1.0E-06 (Xe-133)	5	Control bldg. 91 ft 6 in.
Containment atmosphere		Air			Auxiliary bldg. 66 ft 6 in.
Particulate	1		1.0E-10 (I-131)	5	
Gas	1		1.0E-06 (Xe-133)	5	
Auxiliary building		Air			Auxiliary bldg. 66 ft 6 in. and 43 ft 6 in.
Particulate	6		1.0E-10 (I-131)	5	
Gas	6		1.0E-06 (Xe-133)	5	
Fuel building		Air			Auxiliary bldg. 66 ft 6 in.
Particulate	1		1.0E-10 (I-131)	5	
Gas	1		1.0E-06 (Xe-133)	5	
Waste disposal building		Air			Auxiliary bldg. 66 ft 6 in.
Particulate	1		1.0E-10 (I-131)	5	
Gas	1		1.0E-06 (Xe-133)	5	
Control building		Air			Control bldg. 64 ft 6 in.
Particulate	1		1.0E-10 (I-131)	5	
Gas	1		1.0E-06 (Xe-133)	5	
ESF building		Air			ESF bldg. 36 ft 6 in.
Particulate	1		1.0E-10 (I-131)	5	
Gas	1		1.0E-06 (Xe-133)	5	

Table 11.5 (Continued)

Monitor	Number of channels	Medium	Sensitivity (μCi/cc)*	Range (decades)	Location
Hydrogen recombiner (HR) cubicle vent	2	Air	7.1E-04 (Kr-85)	4	HR bldg. 37 ft 6 in.
Containment recirculation cooler service water outlet	2	Water	1.0E-04 (Cs-137)	5	Yard
Liquid waste	1	Water	1.0E-06 (Cs-137)	5	Auxiliary bldg. 4 ft 6 in.
Steam generator blowdown sample	1	Water	1.0E-06 (Cs-137)	5	Auxiliary bldg. 43 ft 6 in.
Auxiliary condensate	1	Water	1.0E-06 (Cs-137)	5	Auxiliary bldg. 43 ft 6 in.
Turbine building floor drains	1	Water	1.0E-06 (Cs-137)	5	Turbine bldg. 14 ft 6 in.
Reactor plant component cooling water subsystem	1	Water	1.0E-06 (Cs-137)	5	Auxiliary bldg. 43 ft 6 in.
Failed fuel monitor		Water			Auxiliary bldg. 4 ft 6 in.
Gross activity	1		1.0E-06 (Cs-137)	5	
Specific fission product activity	1		1.0E-04 (Cs-137)	5	
Regenerant evaporator monitor	1	Water	1.0E-06 (Cs-137)	5	Warehouse no. 5 4 ft 6 in.
Waste neutralization sump	1	Water	1.0E-06 (Cs-137)	5	Condensate polishing facility 14 ft 6 in.
Main steam relief line	4	Steam	1.0E-01 (Kr-88)	4	Main steam valve bldg. 70 ft 6 in.
Turbine-driven auxiliary feedwater pump discharge	1	Steam	1.0E-01 (Kr-88)	4	ESF bldg. 36 ft 6 in.

*Exponential notation: 1.0E-7 = 1.0×10^{-7}.

**The fuel drop monitors are configured as high range area monitors having a minimum sensitivity of 0.1 R/hr and a range of 6 decades.

Source: FSAR Tables 11.5-1 and 11.5-2

12 RADIATION PROTECTION

The staff has evaluated the proposed radiation protection program presented in Chapter 12 of the Millstone Unit 3 FSAR against the review guidelines and criteria in SRP Section 12 (NUREG-0800). The radiation protection measures at Millstone Unit 3 are intended to ensure that internal and external radiation doses to plant personnel and contractors resulting from plant conditions, including anticipated operational occurrences, will be within applicable limits of 10 CFR 20 and will be as low as is reasonably achievable (ALARA).

The basis of the staff's acceptance of the Millstone Unit 3 radiation protection program is that doses to personnel will be maintained within the limits of 10 CFR 20, "Standards for Protection Against Radiation." The applicant's radiation protection design and program features are consistent with the guidelines of RG 8.8, "Information Relevant To Ensuring That Occupational Exposures at Nuclear Power Stations Will Be As Low As Is Reasonably Achievable" (Rev. 3). The applicant's radiation protection features will help to ensure that occupational radiation exposures are maintained ALARA, both during plant operation and during decommissioning.

On the basis of its review of the FSAR, the staff has concluded that the radiation protection measures incorporated in the design and the proposed radiation protection program will provide reasonable assurance that occupational doses will be maintained ALARA and will be below the limits of 10 CFR 20.

12.1 Ensuring That Occupational Radiation Doses Are As Low As Is Reasonably Achievable

The staff has audited the policy, design, and operational considerations contained in the Millstone Unit 3 FSAR against the criteria in SRP Section 12.1. The staff review consisted of ensuring that the applicant had either committed to follow the criteria of the RGs and staff positions referenced in SRP Section 12.1 or provided acceptable alternatives. In addition, the staff selectively reviewed the applicant's FSAR against the acceptance criteria of the SRP using the review procedures in the SRP. This selective review found the plant acceptable in these areas.

12.1.1 Policy Considerations

The applicant has provided a management commitment to ensure that Millstone Unit 3 will be designed, constructed, and operated in a manner consistent with RGs 8.8; 8.10, "Operating Philosophy for Maintaining Occupational Radiation Exposures As Low As Is Reasonably Achievable"; and 1.8, "Personnel Selection and Training" (Rev. 1). The applicant has identified the specific corporate plan to implement that policy and specified, in detail, facility and equipment design considerations to ensure its accomplishment. This objective is delineated in the ALARA program and is attained through administrative dose control procedures, adequate work planning, and safe practices in all activities related to the plant's operation. The Station Superintendent has the overall responsibility for implementing the ALARA program. He delegates the health physics support functions to the Station Services Superintendent, who delegates this authority to the Radiological Service Supervisor, who is responsible for maintaining the health physics program and for directing the Health Physics Supervisor and each unit-specific ALARA coordinator. The applicant has committed to provide revised FSAR Figures 13.1-6 and -8 showing recent changes for Millstone Unit 3. In addition, line supervisors also are responsible for maintaining plant doses ALARA. The ALARA philosophy was applied during the initial design of the plant. Since then, the applicant has continued to review, update, and modify the plant design and construction phases. The NUSCo, Nuclear and Environmental Engineering Department, Radiological Assessment Branch, periodically reviews, updates, and modifies plant design features and maintenance features as appropriate, using dose data and experience gained from operating nuclear power plants. This is done to ensure that occupational doses will be kept ALARA in accordance with RG 8.8 and SRP criteria.

12.1.2 Design Considerations

The objective of the plant's radiation protection design is to maintain individual doses and total person-rem doses to plant workers (including construction workers) ALARA, and to maintain individual doses within the limits of 10 CFR 20. The general arrangement and shielding provisions at Millstone Unit 3 are in accordance with RG 8.8 and are designed to limit doses to operating personnel to levels that are ALARA. Radiation protection experience from Haddam Neck and Millstone Units 1 and 2 is reflected in the design goals of Millstone Unit 3.

The applicant has used the following design features for ensuring that the occupational radiation dose is ALARA.

(1) Whenever practical, radioactive components are located in separate shielded cubicles to minimize exposure during maintenance, calibration, and inspection activities.

(2) Cubicle access openings incorporate a labyrinth design to preclude direct radiation shine.

(3) Reactor coolant pump designs include an assembled cartridge seal that minimizes service time.

(4) Filters are designed with lifting bails to facilitate remote removal, disposal, and assembly.

(5) Remote control refueling machine reduces exposures during refueling.

(6) Head closure systems have hydraulically operated stud gripper devices that minimize exposures during stud tension operations.

(7) Penetrations through walls separating higher radiation zone areas from lower radiation zone areas are located above head level.

(8) The steam generator tube ends are designed to be flush with the tube sheet in the channel head to eliminate a potential crud trap.

The radiation protection design review is ongoing throughout all phases of the design with formal reviews conducted at regular intervals by the applicant's staff. Therefore, the design considerations of the applicant meet the criteria of RG 8.8 and NUREG-0800 and are acceptable.

12.1.3 Operational Considerations

Operational considerations derived from operating plants were factored into the design considerations previously described. These operational considerations are to ensure that operating and maintenance personnel will follow specific plans and procedures to ensure that ALARA goals are achieved in the operation of the plant. Since the applicant operates two other facilities on the Millstone site, the procedures to be used at Millstone Unit 3 will reflect more than 14 years of experience and will be developed to a high degree of efficiency in maintaining occupational exposure ALARA. Consequently, the complexity of the performance of maintenance, repair, surveillance, and refueling tasks will be factored into the radiation protection and control procedures to minimize radiation dose in accordance with RG 8.8. Operations under conditions of high-radiation exposure are to be preplanned and carried out by personnel trained in radiation protection and using proper equipment. During such activities, personnel will be monitored for exposure to radiation and contamination. Upon completion of major maintenance jobs, personnel radiation exposures will be evaluated and compared with predicted person-rem exposures. The results will be used to make changes in future job procedures and techniques. Radiation dose trends will be reviewed periodically to determine major changes in problem areas and to determine which worker groups are accumulating the highest dose. Plant personnel will use these reports to recommend design modifications or changes in plant procedures. The operational considerations conform to RG 8.8 (Rev. 3) and NUREG-0800 and are acceptable.

12.2 Radiation Sources

The staff has audited the contained and airborne radioactive source terms in Section 12.2 and Chapter 11 of the Millstone Unit 3 FSAR against the criteria in SRP Section 12.2 (NUREG-0800). These source terms are used as inputs for dose assessment and for the design of the shielding and ventilation systems. The staff review consisted of ensuring that the applicant had either committed to follow the criteria of the RGs and staff positions referenced in SRP Section 12.2 or provided acceptable alternatives. In addition, the staff selectively compared source terms for specific systems used by the applicant against those used for plants of similar design. This selective review found the plant's source terms equivalent to those used at other plants.

The applicant has used radiation source terms for normal operation as inputs to shield design calculations to determine personnel protective measures, perform dose assessments, and determine access controls. Source terms used to perform a radiation and shielding review following an accident are in accordance with release fractions in Technical Information Document (TID) 14844 (AEC, March 23, 1962). Sources for normal operation include neutron and gamma fluxes outside the reactor vessel, coolant activities, and fission and corrosion products. During power operation, ^{16}N determines the shielding requirements of the secondary shield wall and portions of the chemical and volume control system. Source terms used for normal operation and anticipated operational occurrences are based on American National Standards Institute (ANSI) Std. N237, "Radioactive Materials in Principal Fluid Streams of Light-Water Cooled Nuclear Power Plants," and on computer code GALE input (NUREG-0017). The staff, therefore, finds the contained sources used at the plant acceptable.

Because of leakage from equipment, sources of the maximum expected airborne concentrations during reactor operation and at shutdown have been tabulated for areas inside major plant buildings that are frequently occupied. These sources are based on typical data from operating plants. The assumptions and parameters used in determining these leakage calculations also have been provided and found to be appropriate. Health physics and plant operating experience will be implemented to ensure that plant personnel will not be exposed to concentrations of airborne radioactive material exceeding those specified in 10 CFR 20.103 and that these concentrations will be maintained at levels that are ALARA. In accordance with NUREG-0800, the source terms used to develop airborne concentration values are comparable to estimates by other applicants with plants of similar design and are acceptable.

12.3 Radiation Protection Design Features

The staff has audited the facility design features, shielding, ventilation, and radiation and airborne monitoring instrumentation contained in the Millstone Unit 3 FSAR against the criteria in SRP Section 12.3 (NUREG-0800). The staff review consisted of ensuring that the applicant had either committed to follow the criteria of the RGs and staff positions referenced in SRP Section 12.3 or provided acceptable alternatives. In addition, the staff selectively reviewed the applicant's FSAR against the specific areas of review and review procedures identified in NUREG-0800. This review found the plant acceptable in these areas.

12.3.1 Facility Design Features

The applicant has addressed facility and equipment design considerations, planning and procedure programs, and techniques and practices used in the overall design for maintaining doses ALARA. The FSAR was reviewed with respect to

(1) the description of the equipment design to be used for ensuring that occupational exposure will be ALARA

(2) information concerning implementation of RG 8.8, Section C.2

(3) the description of any special protective features that use shielding, geometric arrangement, or remote handling to reduce occupational exposure

To maintain occupational doses ALARA, the applicant has designed the facility, to the extent practicable, so that systems and components handling high-activity fluids are in controlled areas, separated from uncontrolled areas by shielded walls. Equipment and components that require manual operation, or may need servicing and instrumentation requiring visual inspection, are located in a zone with the lowest possible radiation. When impractical to do this, such items are designed so that they may be removed to a low-radiation zone for maintenance. Steam generator and pressurizer manways are sized to facilitate entry and exit of personnel wearing protective clothing. Valves, pumps, demineralizers, and filters are designed to allow operation, maintenance, and inspection with minimal exposure. To control the production of crud (e.g., ^{60}Co and ^{58}Co), use of hard-facing materials with cobalt content and nickel-based alloys is limited to where component reliability requires their use. The choice of material is based on consideration of superior wear characteristics and reduced maintenance requirements. High-pressure and high-temperature graphite packing is used in the primary system valves to minimize crud buildup and maintenance. Flush and drain connections enable decontamination of radioactive piping before maintenance of equipment, and sample stations are located in low-radiation areas to minimize personnel exposure during sampling. Wherever it is not feasible to install permanent shielding, and where shielding may be required, the design philosophy of the applicant, which emphasizes adequate space for ease of motion, would allow the use of portable shielding.

12.3.2 Shielding

The objective of the plant's radiation shielding is to provide protection against radiation for operating personnel (both inside and outside the plant) and for the general public during normal operation, anticipated operational occurrences, and accidents. The shielding was designed to meet the criteria of the radiation dose rate zone system that are based on frequency and duration of occupancy. The design of the radiation shielding considers the dose rate criterion for each zone based on maximum access time estimates in each compartment within the zone. The design was reviewed, updated, and modified during all phases of the plant's design and construction. The health physics staff will update entry requirements in accordance with 10 CFR 20.203 or Standard Technical Specification requirements. Shielding analyses were made using accepted codes, models, and assumptions. The basic shielding analysis was performed using computer codes accepted by the staff. Shielding in the Millstone Unit 3 plant was designed using Stone & Webster Engineering Corporation's Topical Report "Radiation Shielding Design and Analysis Approach for Light Water Reactor Power Plants" (RP-8A). This approach is recommended by the staff and incorporates the design features in RGs 1.69 and 8.8. Besides limiting exposure to plant personnel, contractors, visitors, and others, the plant shielding also functions to reduce neutron activation of equipment, piping supports, and other items and to limit radiation damage to equipment and materials to below the specified integrated life dose limits. All concrete shielding in the plant is based on the criteria of RG 1.69, which provides guidance on the fabrication and installation of concrete radiation shields. The staff concludes that the shielding design methodology presented is acceptable.

Shielding for the spent fuel transfer system was reviewed. To preclude unacceptable radiation dose rates during fuel transfer, a special radiation shield has been provided inside containment where the fuel transfer tube traverses the gap between the containment wall and the refueling cavity wall. This shielding design concept further reduces the dose rate at the surface of the shield and at the personnel access hatch.

Outside containment, the fuel transfer tube is inaccessible to personnel because of backfill that covers the transfer tube and a security fence located between the containment and the fuel building, which ensures limited access to this area.

Five radiation monitors with local audible and visible alarms, as well as remote alarms in the control room, are used to monitor fuel transfer operations. Three radiation monitors are located in the passageway and access area adjacent to the fuel transfer tube in the containment building. The other two monitors are in the fuel building above the fuel transfer canal. The staff finds the spent fuel transfer tube shielding design acceptable in accordance with the criteria in SRP Section 12.3.

The applicant has provided a description of the borated silicon shields used to provide protection from neutrons and gamma rays streaming in from the annulus between the reactor pressure vessel and the biological shield. This shield is designed to minimize radiation leakage into occupiable areas of the containment, thus reducing the neutron and gamma dose rate on the operating floor, during normal operations, to acceptable levels. The staff finds this design acceptable.

The applicant has not provided the shielding design in accordance with Item II.B.2 of NUREG-0737, which requires applicants to evaluate the access to vital areas necessary to operate essential systems required after a loss-of-coolant accident (LOCA) with significant core damage. However, the applicant has committed to provide the information before fuel loading. This is a confirmatory item.

12.3.3 Ventilation

The applicant's ventilation systems are designed to provide ventilation air suitable to ensure that plant personnel are not exposed to airborne concentrations exceeding those in 10 CFR 20.103 and that concentrations to which personnel may be exposed do not exceed the limits of 10 CFR 20.101. In the design of all ventilation systems, the applicant intends to meet this objective and maintain exposures ALARA by

(1) directing the airflow from areas of lesser potential contamination to areas of greater potential contamination

(2) providing airborne radiation monitoring

(3) allowing adequate space around units for servicing and replacement

(4) providing for ease in maintaining and in-place testing of filters to preclude additional radiation exposure

After initial operation, periodic testing of filters and adsorbers will be performed and frequency of changeout determined as a result of these tests.

The design criteria are in accordance with the guidelines of RG 8.8, and the atmospheric cleanup units conform to the design, testing, and maintenance criteria of RG 1.52. The staff concludes that the applicant's ventilation system is designed to maintain personnel exposures at a small fraction of 10 CFR 20 values, meets the criteria of NUREG-0800, and, therefore, is acceptable.

12.3.4 Area Radiation and Airborne Radioactivity Monitoring Instrumentation

12.3.4.1 Area Radiation Monitoring Instrumentation

The applicant's area radiation monitoring system is designed to

(1) inform operations personnel of radiation levels in areas where area radiation monitoring system (ARMS) units are located

(2) provide warning, when abnormal levels occur, by audible and visual alarms locally, in the control room, and in the health physics office

(3) warn of equipment malfunction and leaks in specific areas

(4) provide a continuous record of the radiation level at key locations throughout the plant

To meet these objectives, the applicant plans to use 56 area monitors in the plant, all of which contain a remotely operated integral check source. Each channel will alarm if abnormally high-radiation levels exceed preset dose rate levels or whenever circuit failures occur. The criteria for locations of area radiation monitors are based on (1) occupancy factors, (2) potential for exposure to high radiation, (3) potential for equipment failure, (4) storage of new and spent fuel, (5) normally or potentially radioactive release points, (6) monitoring for accidental criticality, and (7) post-LOCA, long-term, high-range monitoring inside containment. These high-range monitors will be installed in accordance with TMI Action Plan Item II.F.1.3 (NUREG-0737) and meet the specifications of Table II.F.1-3 of NUREG-0737.

12.3.4.2 Airborne Radioactivity Monitoring Instrumentation

The design objectives of the airborne radioactivity monitoring program are

(1) to inform operations personnel of airborne activity trends and give early warning of abnormal increases in activity levels

(2) to warn of potential overexposure to airborne radioactivity so that respiratory protection can be used as required

(3) to furnish records of airborne radioactivity trends

(4) during postulated accidents, to alarm and initiate isolation of the normal ventilation system and actuate the emergency ventilation system

All airborne monitors have local annunciation and a display indicating airborne concentrations. Each channel has an independent power supply and is served by a dedicated microprocessor and associated computer for processing data in a channel. Source location can be identified by collecting local air samples in the specific areas being monitored. These monitors sample air from the reactor containment, the engineered safety features building, the control room, and locations in the reactor plant heating and ventilation system upstream of the ventilation vent monitor. Each monitor is capable of detecting airborne activated corrosion and fission products at low levels. The particulate and gas detector channels of these monitors are provided with an "alert level" alarm, in addition to the "high" alarm. These monitors should be capable of detecting 10 maximum permissible concentration (MPC) hours of particulate and iodine radioactivity from any compartment that may contain this activity and that may be occupied by personnel. Each detector of the airborne radioactivity monitoring system is initially given a primary calibration with typical sources of interest. Secondary standards are counted in reproducible geometry during primary calibration. These secondary standards will be used in subsequent calibrations and whenever repairs or maintenance is performed on the monitoring systems to ensure proper functioning. The frequency of calibration and associated accuracy of the monitoring system will be in accordance with the requirements of ANSI Std. 323, "Radiation Protection Instrumentation Test and Calibration." All installed instruments have independent emergency battery power supplies that are activated whenever a power failure occurs. The applicant will comply with the requirements of TMI Action Plan Item 2.1.8.C of NUREG-0578 (Item III.D.3.3 of NUREG-0737) on improved in-plant iodine monitoring by providing equipment to determine accurately the airborne iodine concentrations where plant personnel may be present during an accident. The staff finds that the applicant's area radiation and airborne radioactivity monitoring systems satisfy the design objectives of RG 8.8 and the criteria of NUREG-0800 and are acceptable.

12.4 Dose Assessment

The staff has audited the applicant's dose assessment provided in Section 12.4 of the FSAR against the criteria in SRP Section 12.3 (NUREG-0800). This review consisted of ensuring that the applicant had either committed to follow the criteria of the RGs and staff positions referenced in SRP Section 12.3 or provided acceptable alternatives. In addition, the staff selectively compared the dose assessment made by the applicant for specific functions against those made for other plants of similar design. This selective review found the plant's dose assessment equivalent to those of other plants.

The applicant has based his estimate of annual person-rem doses on experience from currently operating reactors, engineering judgment, and the manner in which his own station has been designed and will be operated. RG 8.19 was not used to perform a dose assessment that considers doses that will be received by personnel on the basis of occupancy factors in zones to be occupied, the dose rates in these zones, estimates of occupancy times, and the staff necessary to perform the various tasks involved in plant operations. Average doses were estimated by comparing Millstone Unit 3 with Millstone Unit 2 operating experience. Both Units 2 and 3 are PWRs operated by the same utility in accordance with the same basic operating, maintenance, repair, and refueling procedures. The annual collective dose equivalent based on plant systems is expected to be about 549 person-rems. Currently operating PWRs average more than 700 person-rems per unit annually, with particular plants experiencing an average lifetime annual dose as high as 1,300 person-rems. These dose averages are based on widely varying yearly doses at PWRs. The applicant's estimate is based on personnel exposure experience at Millstone Unit 2 for the years 1979, 1980, 1981, and 1982, as presented in FSAR Table 12.4-2. This table provides the specific man-rem exposures at Millstone Unit 2 by plant system regardless of when these exposures were obtained (e.g., during normal operations, maintenance, repair, or refueling activities) and by whom (e.g., plant operations personnel, plant maintenance personnel, or contractor/vendor personnel).

The applicant has provided a tabulation of the maximum expected doses to personnel caused by airborne radioactivity from inhalation and submersion. The tabulation is based on Millstone Unit 2 exposure histories and is within MPC limits.

Radiation exposure to construction workers during construction of Millstone Unit 3 resulting from operation of Millstone Units 1 and 2 will be maintained ALARA through the use of monitoring, administrative procedures, and physical barriers (e.g., fences, locked gates, and locked buildings). The total annual dose rate to Millstone Unit 3 construction workers will be within 10 CFR 20.105 limits and ALARA.

12.5 Operational Radiation Protection Program

The staff has audited the organization, equipment, instrumentation, facilities, and procedures for radiation protection contained in the Millstone Unit 3 FSAR against the criteria of SRP Section 12.5 (NUREG-0800). The plant's health physics program objectives are to (1) provide reasonable assurance that the limits of 10 CFR 20 are not exceeded, (2) further reduce unavoidable exposures, and (3) ensure that every reasonable effort is made to maintain occupational radiation doses ALARA. The staff review consisted of ensuring that the applicant had either committed to follow the criteria of the RGs and staff positions referenced in SRP Section 12.5 or provided acceptable alternatives.

12.5.1 Organization

The Radiological Service Supervisor, who reports directly to the Station Services Superintendent, will implement and enforce the Millstone Unit 3 health physics program. He directs the Health Physics Supervisor and each unit-specific ALARA coordinator for all Millstone units. The applicant has committed to provide the staff with an organization chart for the Millstone Unit 3 health physics program. The above organization is acceptable to the staff.

12.5.2 Health Physics Facilities

To conduct routine operations the health physics staff will maintain facilities such as a radiochemistry area, consisting of a radioactive chemistry laboratory and a low-background-count laboratory for counting air and swipe samples for gamma isotopic analysis or low-level counting; an instrumentation calibration room for calibrating health physics survey instruments and self-reading dosimeters; a change room for obtaining clean protective clothing and for removing and handling contaminated protective clothing; a laundry room for laundering protective clothing as well as respirators; and a personnel decontamination room. The counting room will contain equipment for analysis of alpha, beta, and gamma-ray activity in airborne radioactivity samples and in smear samples, and radionuclide concentrations in liquid samples. A whole-body counting system will be located on site, as needed, for in vivo measurement of radioactivity levels in station personnel, support personnel, or visitors to determine radionuclide body burdens, if any. This program will be conducted in parallel with a bioassay program for urine and fecal analysis. A thermoluminescent dosimeter (TLD) reader and associated equipment are on site to enable prompt processing of TLD badges to verify dose immediately.

12.5.3 Equipment and Instrumentation

Continuing evaluation and review of the radiological status of the station will be carried out by health physics personnel so that levels of radiation will be known at all times in areas where personnel are working. Equipment to be used for radiation protection purposes includes portable alpha, beta, gamma, and neutron survey meters. The applicant has added instruments that range to 10^4 R/hr to his portable survey and area radiation monitoring instrument inventory. Airborne gaseous, particulate, and iodine samplers anu continuous air monitors are available. All portable radiation detection equipment and monitoring systems are state-of-the-art to ensure that in-plant personnel receive timely and accurate information. As stated previously, area and airborne radioactivity monitoring equipment incorporates alarm setpoints to alert workers whenever radiation levels exceed their setpoint levels. Calibration of these monitors, as well as the portable survey meters, will be performed in accordance with ANSI Std. 323 calibration standards. Radiation protection personnel using this equipment are trained and experienced. For contamination control, portal monitors and friskers will be used at exits from radiation control areas to monitor personnel leaving the station. Protective clothing and respiratory equipment are also used, as required, to keep exposure ALARA.

All plant personnel are required to wear a TLD as the primary method for determining beta-gamma dose. For neutron dosimetry, Millstone Unit 3 will comply with the applicable recommendations of RG 8.14. Self-reading pocket dosimeters also will be issued as a secondary method for beta-gamma dosimetry and will provide a day-to-day estimate of personnel dose from gamma radiation that can be used for radiation work permit (RWP) job planning. Dose records for each individual will be maintained in accordance with RG 8.7 and as required by 10 CFR 20.407. The bioassay program at Millstone Unit 3 will be used to assess the effectiveness of the respiratory protection program and will follow the guidance of RGs 8.9 and 8.26. The whole-body counter will be located at the station for in vivo counting of radioactivity levels in station personnel, visitors, contractors, and others. Counting will be conducted on a scheduled basis, and other bioassay methods (e.g., urine and fecal samples) will be provided when necessary.

12.5.4 Procedures

Health physics surveillance of work activities is provided to ensure positive access control and stay time in radiation areas. Radiation protection personnel will routinely survey selected areas of the plant to assess radiation levels, radioactive contamination, and airborne radioactivity concentrations. These surveys will be performed at a selected frequency depending on location, potential radiation levels, occupancy factor, and station operating status. Areas found to be contaminated will be barricaded and posted with appropriate warnings before decontamination. Entry for work in radiation fields, contaminated areas, or airborne radioactivity areas will be controlled by RWPs that administratively control access and stay time in these areas. Approval of RWPs by radiation protection personnel will permit entry and work in radiation and/or air or surface contamination areas, on the basis of procedural requirements. These permits state personnel who are authorized to perform work, location and description of work, procedures and precautions to be observed, protective clothing and respiratory equipment required, and dosimetry and other procedural requirements.

The health physics training program ensures that plant personnel and visitors are trained in radiation protection practices and procedures to maintain their radiation doses ALARA.

The training program provides for at least annual training for personnel at all levels of the radiation protection organization. Annual general employee training is required for all employees and contractors entering the radiological controlled area. This instruction includes radiation work training and respiratory protection training. Station health physics technicians must initially complete a 2-week health physics training and certification course and an annual requalification course thereafter. Station health physics supervisory personnel must complete an annual requalification course consisting of more in-depth material. In addition, periodic specialized training is required for personnel with specialized skills such as dosimetry technicians, respiratory protection specialists, and instrument calibration technicians. Written examinations are administered in all these courses, and records are kept for future reference. Oral examinations are not used in the training program because they are not easily documented.

The content of the health physics training program will meet the intent of RGs 8.27, 8.13, and 8.29 and NUREG-0737.

Health physics technicians are required to meet or exceed the qualifications specified in ANSI N18.1, and the Radiological Services Supervisor is required to meet or exceed the qualification for Radiation Protection Manager in RG 1.8, Revision 1.

On the basis of the information in the FSAR and its amendments and the applicant's responses to its question, the staff concludes that the applicant intends to implement a radiation protection program that will maintain in-plant radiation exposures within the applicable limits of 10 CFR 20 and will maintain exposures ALARA in accordance with RG 8.8.

13 CONDUCT OF OPERATIONS

13.1 Organizational Structure of Applicant

13.1.1 Management and Technical Support Organization

13.1.1.1 General

On April 10, 11, and 12, 1984, an NRC review team visited the Northeast Utilities corporate office in Berlin, Connecticut, and the Millstone Unit 3 plant in Waterford, Connecticut. The purpose of the visit was to review the proposed organization for the operation of Millstone Unit 3 from the level of senior corporate officer down to and including the proposed operating staff at Millstone Unit 3. Northeast Utilities is a parent company of several electric utility subsidiaries, the Northeast Nuclear Energy Company (NNECo) and the Northeast Utilities Services Company (NUSCo). NNECo is responsible for the design, construction, and operation of Millstone Unit 3. NUSCo has been designated by NNECo to furnish engineering, design, and construction services for Millstone Unit 3 and to furnish services for the licensing, fuel management, and operation of the unit. NUSCo also provides support for the operation of Millstone Units 1 and 2, Haddam Neck Nuclear Power Station (Connecticut Yankee), and Northeast Utilities' fossil and hydro-generating plants.

During a meeting at the NNECo corporate office in Berlin, Connecticut, the applicant provided the review team with a thorough briefing on the organization for the operation and technical support of Millstone Unit 3. This briefing was followed by discussions with numerous corporate personnel, which were sufficient for the review team to gain a feeling for the responsibilities and attitudes of the individuals and an understanding of how they fit in the organization and how they interface with other onsite and offsite organizational units. Discussions with plant personnel were held at the site to obtain the same type of information.

The following description and evaluation are based on information in the FSAR and supplemental information obtained during the visit, some of which will require confirmatory documentation by the applicant.

The staff performed the review in accordance with SRP Section 13.1.1 (NUREG-0800).

13.1.1.2 Organizational Structure

The current organizational structure for NNECo and NUSCo is shown in Figure 13.1. The senior corporate officer in charge of the applicant's nuclear program is the Senior Vice President, Nuclear Engineering and Operations. The incumbent officer has over 20 years of nuclear experience, including a period as Assistant Plant Superintendent and Project Manager for Millstone Units 1 and 2. He has been responsible for the operation and maintenance of Millstone Units 1 and 2 and Haddam Neck (Connecticut Yankee) and will be responsibile for the operation and maintenance of Millstone Unit 3. Additionally, he provides engineering support for NNECo's fossil and hydro units. A separate division is responsible for the operation of the fossil and hydro units.

Reporting directly to the Senior Vice President, Nuclear Engineering and Operations, are the Vice President, Nuclear Operations; the Vice President, Nuclear and Environmental Engineering; the Vice President, Generation Engineering and Construction; and the Nuclear Project Manager. A brief description of these groups is presented below.

(1) Nuclear Operations Department

The Nuclear Operations Department is responsible for all nuclear plant operations of the Millstone and Haddam Neck nuclear power stations. The incumbent Vice President, Nuclear Operations, has over 20 years of nuclear experience, which includes several years as Station Manager for the operation of Millstone Units 1 and 2. He provides technical and administrative direction to the nuclear station superintendents and their operating staffs. He also directs the headquarters unit coordinating engineers who coordinate operating and NUSCo technical and licensing support activities. His present staff consists of about 20 individuals. Reporting to the Vice President is the Station Superintendent of Millstone, the Station Superintendent of Connecticut Yankee, a supervisor of Nuclear Operations, a staff engineer, and a nuclear operations consultant. The Millstone and Connecticut Yankee plant staffs report to the respective station superintendents.

The organization under the Vice President, Nuclear Operations, is shown in Figure 13.2.

(2) Nuclear and Environmental Engineering Division

The Nuclear and Environmental Engineering Division is responsible for providing engineering support to all generating facilities of Northeast Utilities. The incumbent Vice President, Nuclear and Environmental Engineering, with about 20 years of nuclear experience provides technical support for the environmental programs and engineering services for nuclear analysis and manages the quality assurance program, the reliability engineering program, the radiological program, and the licensing, training, and nuclear safety assessment programs. Reporting to the Vice President is the Director of Environmental Programs, the Director of Nuclear Engineering and Operations Services, the Director of Nuclear Engineering, and the Director of Nuclear Training.

The present staff of the Nuclear and Environmental Engineering Division consists of 421 individuals and is expected to grow to about 450 individuals by the projected fuel-loading date for Millstone Unit 3. Of the current staff, 41 are assigned to work on fossil and hydro units, 312 are assigned to Connecticut Yankee and Millstone Units 1 and 2, and 68 are assigned to Millstone Unit 3.

The organization of the Nuclear and Environmental Engineering Division is shown in Figure 13.3.

(3) Generation Engineering and Construction Division

The Generation Engineering and Construction Division is responsible for the project management of new facilities and provides the engineering and technical support for backfit and betterment projects. The incumbent Vice President, Generation Engineering and Construction, has about 16 years of experience at Northeast Utilities Services Company on nuclear projects. He is responsible for the construction of Millstone Unit 3; the mechanical, electrical and civil engineering for plant betterment and backfit projects; the planning for refueling and unscheduled outages; the construction management at new site projects; and the support of maintenance activities. Reporting to the Vice President are the Nuclear Project Manager for Millstone Unit 3, the Director of Generation Engineering and Design Department, the Director of the Generation Projects Department, and the Director of the Generation Construction Department.

The present staff of the Generation Engineering and Construction Division consists of about 510 individuals. Of this number, about 45 are assigned full time on the fossil and hydro units. The organization of this division is shown in Figure 13.4.

(4) Nuclear Project Manager

The Nuclear Project Manager is responsible for the simulator project. Northeast Utilities has contracted for four reference simulators: one each for Connecticut Yankee and Millstone Units 1, 2, and 3. The group under the Nuclear Project Manager has been established to provide contract management and technical expertise for this project and to begin the initial work on the simulator training programs. It is anticipated that operator training at the simulators will be the responsibility of the Director of Nuclear Training in the Nuclear and Environmental Engineering Division.

13.1.1.3 Summary and Conclusion

During its visit to the Northeast Utilities' headquarters and the site, the staff met with many of the key individuals noted above and some of their first-line supervisors. It discussed the staffing of the groups, their background and qualifications, and the present and projected interactions between the various groups that will support the operation of Millstone Unit 3. This organization is currently supporting the operation of Connecticut Yankee and Millstone Units 1 and 2. Therefore, it has extensive experience in the operation of nuclear power plants. This is reflected in the number and experience levels of the personnel.

The staff concludes that the applicant meets the acceptance criteria of SRP Section 13.1.1 (NUREG-0800).

13.1.2 Plant Staff

13.1.2.1 Organization

The organization for the Millstone station is shown in Figure 13.5. The Station Superintendent is responsible for the safe and efficient operation of the Millstone station (Units 1, 2, and 3). He reports to the Vice President, Nuclear Operations. The current authorized staff for all three units consists of 793 individuals. Reporting directly to the Station Superintendent are the Unit Superintendent of Unit 1, Unit Superintendent of Unit 2, Unit Superintendent of Unit 3, Station Services Superintendent, Office Supervisor, and a staff assistant. The Station Services Superintendent, the Office Supervisor, and the staff assistant have responsibilities common to all three units.

As shown on Figure 13.6, the Unit Superintendent of Unit 3 has reporting to him the Operations Supervisor, the Engineering Supervisor (now the Startup Supervisor), the Maintenance Supervisor, and the Instrument and Controls (I&C) Supervisor. Toward the end of the startup test program, it is planned that the Startup Supervisor's organization will become the Unit Engineering organization and the Startup Supervisor will become the Engineering Supervisor. The projected staffing for Unit 3 is about 272 individuals, of whom 18 will provide services common to the site. Corollary organizations are provided for Units 1 and 2.

(1) Plant Operations

Plant Operations, under the Operations Supervisor, is responsible for the safe and efficient operation of the unit. The Operations Supervisor is responsible for the day-to-day operation of the unit. He ensures the safe and efficient operation of his unit in accordance with applicable licenses, operating instructions and procedures, emergency procedures, and safety rules and regulations. He coordinates unit operations with maintenance and other departments. The Operations Supervisor will hold a senior reactor operator (SRO) license on Millstone Unit 3.

The applicant plans to operate Millstone Unit 3 with six shift crews. Each shift crew will be under the direction of a senior licensed Shift Supervisor. Reporting to the Shift Supervisor will be a Shift Supervisor Staff Assistant (unlicensed) and a senior licensed Supervising Control Operator. The Shift Supervisor Staff Assistant will assist the Shift Supervisor as directed and, during accident situations, will provide communications assistance to the Shift Supervisor. Reporting to the Supervising Control Operator will be two licensed control operators and three unlicensed equipment operators. In addition, one health physicist technician and one chemical technician will be assigned to each shift.

The applicant does not plan to have shift technical advisors. He plans to provide engineering expertise on shift by upgrading the training and qualifications of the Shift Supervisor. The staff considers this an open item pending final Commission action on the "Draft Commission Policy Statement on Engineering Expertise on Shift," which was published for comment in the Federal Register on July 25, 1983.

The proposed shift staffing does not provide for individuals on each operating shift with hot operating experience on a comparable nuclear power plant. The Commission policy regarding operating experience requirements and the proposed means for meeting the policy are still under consideration by the Commission. However, it now appears that at least one senior operator on each shift must have at least 6 months of hot operating experience at a similar type plant or a shift advisor must be provided who has had at least 1 year of hot operating experience and who is adequately qualified to advise the shift crew. Until such time as this matter is resolved, the staff considers that the applicant should plan to provide for individuals with adequate operating experience on each shift.

(2) Plant Maintenance

The Maintenance Supervisor, who reports to the Superintendent of Unit 3, is responsible for the maintenance of all unit electrical and mechanical equipment. Reporting to the Maintenance Supervisor are two assistant maintenance supervisors and several engineers and technicians. The proposed staffing is about 60 individuals.

Additionally, an interplant maintenance force exists whereby mechanics, electricians, and I&C technicians can be assigned to a particular unit when major outages occur. A permanent staff of physical construction workers is also assigned at Millstone to support maintenance and backfit work at any of the nuclear units.

(3) Instrument and Controls

The Instrument and Controls (I&C) Supervisor, who reports to the Unit Superintendent of Unit 3, is responsible for the unit's instruments and controls. He establishes standards and frequency of calibration and ensures that instrumentation and related testing equipment are properly used, inspected, and maintained. Reporting to the I&C Supervisor are three assistant instrument and control supervisors. The planned staffing for the I&C group is about 32 individuals.

(4) Unit Engineering

The Engineering Supervisor, who reports to the Superintendent of Unit 3, is responsible for Unit 3 engineering services. These include the coordination and engineering review and approval of design changes to nuclear plant systems, the trending of data relating to operations and maintenance, the implementation of the nuclear in-service inspection program, and the evaluation of overall core performance. The unit reactor engineer reports to the Engineering Supervisor.

This organizational unit will not be established until about the time of completion of the startup test program. It will be formed from the current startup group which, under the supervision of the Startup Supervisor, reports to the Superintendent of Unit 3. It is anticipated that the group will consist of about 20 engineers who are now part of the startup group and who will obtain experience during the startup test program.

It is also anticipated that the current Startup Supervisor will become the Engineering Supervisor of Unit 3. (See discussion below.)

The Station Services Superintendent reports to the Station Superintendent and provides services for all three units. He is responsible for station security, quality assurance, training, plant chemistry, and radiation protection. Reporting to the Station Services Superintendent are the Security Supervisor, Quality Services Supervisor, and Radiological Services Supervisor. The projected staff of the Station Services group for Unit 3 is about 62 individuals of whom 12 are common to the site.

The Security Supervisor is responsible for station security and has the Security Shift Supervisors reporting to him.

The Quality Services Supervisor plans, schedules, coordinates, and supervises the plant activities related to quality assurance/quality control, nuclear records, and stores. He conducts drills and exercises to maintain proficiency in all emergency procedures. The Quality Assurance Supervisor, who reports to the Quality Services Supervisor, performs audits of plant activities to ensure compliance with applicable rules and procedures. The quality assurance program is described in detail in Section 17 of this SER.

The Radiological Services Supervisor plans, schedules, coordinates, and supervises the plant activities related to health physics, chemistry, as low as is reasonably achievable criteria, and medical, radwaste management, and emergency planning. The health physics program is described in detail in Section 12. Reporting to the Radiological Services Supervisor is the Chemistry Supervisor to whom the assistant chemistry supervisors for each unit report. The Chemistry Supervisor is responsible for chemical analysis and tests of all water and steam systems and the formulation of test results to ensure operation within license requirements and all applicable Federal and state codes.

The Startup Supervisor, who reports to the Superintendent of Unit 3, is responsible for the preoperational and startup test program. He coordinates the preparation, review, control, evaluation, and distribution of all test procedures; coordinates the testing functions of the Millstone Unit 3 departments; ensures completion of test prerequisites and notification of readiness to perform preoperational and subsequent tests; ensures adequacy of resources for the conduct of each test; coordinates the resolution of design, construction, or testing deficiencies; and coordinates the review and approval of test results.

The Startup Supervisor is currently assigned a staff of about 40 individuals, including the Unit 3 reactor engineer. These are augmented by about 13 Stone & Webster engineers who report to him.

At about the time Unit 3 becomes operational, the position of Startup Supervisor will be abolished and the position of Engineering Supervisor will be created.

In addition to the Startup Supervisor for the startup, a Joint Test Group (JTG) has been formed. The JTG acts in an advisory capacity to the Station Superintendent for the proper conduct of the Unit 3 startup test programs. The JTG consists of the Millstone Unit 3 Superintendent as Chairman, the Startup Supervisor, the Stone & Webster Engineering Corporation Lead Advisory Engineer, and the Westinghouse Startup Manager, or their designated representatives. The JTG reviews and approves all preoperational and subsequent test procedures, approves the release of preoperational and subsequent test procedures for execution, reviews system deficiencies before procedure release, approves changes to preoperational and subsequent test procedures, reviews and approves the results of preoperational and subsequent tests performed under the startup test program, and ensures that all preoperational and subsequent test deficiencies are resolved.

13.1.2.2 Qualifications

The applicant has listed individually or by classification the station positions for Unit 3 and has stated that the qualification requirements will meet those for the comparable positions in ANSI N18.1-1971, as referenced by RG 1.8, Revision 1 (1977). The Millstone Unit 3 Startup Supervisor will have 8 years of applicable power plant experience with a minimum of 2 years of applicable nuclear power plant experience. The Millstone Unit 3 Startup Engineer must have a bachelor's degree in engineering or the physical sciences or the equivalent and 2 years of applicable power plant experience, at least 1 year of which should be applicable nuclear power plant experience.

The staff finds the described qualifications acceptable because they meet those described in RG 1.8, Revision 1 (1977). In addition, the staff has reviewed the qualifications of individuals assigned to key management and supervisory positions and finds them acceptable.

13.1.2.3 Conclusion

The staff concludes that the applicant meets the acceptance criteria of SRP Section 13.1.2 (NUREG-0800).

13.1.3 Summary and Conclusion

On the basis of its review of information in Section 13.1 of the FSAR and information received during its meeting with the applicant on April 10, 11, and 12, 1984, the staff concludes the following:

(1) The corporate structure provides for clear lines of authority and communication between the plant staff and the corporate entities that will provide technical support for the operation of the plant. The staff finds this proposed structure and the number of persons projected for assignment within this structure and their qualifications acceptable.

(2) The plant staff structure provides for clear lines of authority and communication from the technical groups to the Station Superintendent and to the Superintendent of Unit 3.

(3) The staff has reviewed the qualification requirements established by the applicant for plant staff personnel and considers them acceptable since they meet the position requirements in Revision 1 of RG 1.8. In addition, its review of résumés currently available indicates that the individuals in key supervisory positions meet the staff's qualification requirements.

(4) The staff has reviewed the composition of the shift crew and concludes that it meets the requirements of 10 CFR 50.54. In addition, the applicant's proposal to provide staff for six-shift rotation ensures adequate personnel so that there should be no need for routine use of overtime. Currently 34 individuals are in training to obtain cold licenses for shift staffing purposes. The staff considers that this is an adequate number to meet its shift staffing requirements at fuel loading. As noted earlier, the issues pertaining to shift technical advisors and personnel with hot operating experience on each shift are open items.

(5) The staff has reviewed the organizational structure, qualification requirements, and number of individuals assigned or to be assigned to the test program and finds that the applicant has provided for clear lines of authority and an adequate number of testing personnel; accordingly, the staff finds the test program acceptable.

(6) The staff has reviewed the assignment of responsibilities for fire protection and the composition of the fire brigade and finds they are in accordance with Branch Technical Position CMEB 9.5-1 and are acceptable.

The findings contribute to the staff's judgment that the applicant complies with the requirements of 10 CFR 50.40(b) (is technically qualified to operate a nuclear power plant) and that the applicant will have the necessary managerial and technical resources to provide assistance to the plant staff in the event of an emergency, as specified in SRP Section 13.1 (NUREG-0800).

13.2 Training

The applicant's training programs for licensed reactor operators and nonlicensed plant staff were reviewed according to SRP Section 13.2 (NUREG-0800). The staff acceptance criteria included applicable portions of 10 CFR 19, 50, and 55, and RGs 1.8 and 1.149 as well as the TMI Action Plan (NUREG-0737) and H. R. Denton's letter of March 28, 1980, to all power reactor applicants and licensees.

13.2.1 Licensed Operator Training Program

A training program for Millstone Unit 3 licensed reactor operators has been implemented to develop and maintain an organization fully qualified to operate the plant and maintain plant safety. The initial, requalification, and replacement programs, which are designed to meet the requirements of 10 CFR 50 and 10 CFR 55 and the TMI Action Plan, are based on the individual employee's level of education, experience, and skills as well as on the level of assigned responsibility and intended position.

13.2.1.1 Initial Operator Training Program

The initial training program for personnel who will be licensed consists of the following discrete segments.

(1) Operating PWR Training

Two programs are used: Fundamentals of Nuclear Training is approximately an 11-week program, and Fundamentals of Nuclear Training and Nuclear Plant Training is approximately a 21-week program. The duration and scope reflect the academic requirements of the license candidate. The 11-week program is designed for individuals who have operated an operating commercial light water nuclear power plant as equipment operators and have a minimum of 2 years of nuclear power plant experience. The 21-week program is designed for individuals with no previous nuclear-related experience. This phase of the training program is designed to provide license candidates with the appropriate amount of academic training and with a thorough understanding of basic principles, characteristics, and unique features of the nuclear system. The major areas to be covered are mathematics, basic nuclear physics, reactor operations, core physics, radiation protection, plant chemistry, instrumentation and control, fluid flow, thermodynamics, heat transfer, and plant performance.

(2) Simulator Training

This phase of the training program is designed to teach the trainee plant operation and transient characteristics on the simulator. Two different PWR cold-license simulator training programs will be used. Individuals who are seeking a cold license and have not held a reactor operator license will attend a 5- to 7-week simulator training course on the Millstone Unit 3 simulator. Individuals who previously have held an NRC license on a commercial PWR, a Naval reactor equivalent qualification, or have simulator certification will be given a special course designed to familiarize them with a large Westinghouse four-loop PWR similar to Millstone Unit 3. This course, given at the Westinghouse Training Center, will consist of approximately 5 weeks of lectures and 2 weeks of simulator training. In addition, before taking the NRC licensing examination, all cold-license candidates will receive a minimum of a 1-week simulator refresher course that will provide them with an in-depth review of PWR systems, procedures, and operating characteristics.

The applicant has not submitted the content details of the above simulator training programs for NRC review; however, the applicant has committed to structure each of the simulator training programs to contain at least those transients and manipulations specified in Enclosure 4 of H. R. Denton's March 28, 1980, letter. The staff, therefore, finds these simulator training programs acceptable. It will verify that all the requirements listed in Enclosure 4 of H. R. Denton's March 28, 1980, letter have been met before an operating license is issued.

(3) Millstone Unit 3 Site School

This training course is designed to provide cold-license candidates with an in-depth study of the Millstone Unit 3 systems and equipment and also prepare cold-license candidates for NRC licensing examinations. This phase of the training program is of approximately 16 weeks' duration, of which approximately half of the time will be spent in formal classroom instruction. The remaining half of the time will be spent in structured study and/or in the plant identifying the equipment associated with the topics covered in the classroom sessions.

(4) Shift Work

Following site school, operations personnel will begin shift work with one shift designated as a training shift. As a minimum, each license candidate will receive 6 weeks of formal classroom training during the shift work phase of the training program. The course topics include secondary plant chemistry; radiochemistry, waste processing, and health physics; plant transients and accident analysis; procedures; mitigating core damage; and facility license and Technical Specifications. In addition, lectures covering other topics, such as system design and/or procedure changes, will be presented.

(5) On-the-Job Training

The on-the-job training program consists of actual involvement in the preparation of plant system operating procedures and check lists. Throughout this period of training, the operating personnel must demonstrate that they have acquired adequate skill and knowledge to perform the duties to which they will be assigned for operation of the unit.

(6) Evaluation of Training Program Effectiveness

The program effectiveness for each individual will be evaluated by periodic examinations. The examinations will be designed to measure how well each student meets the program objectives. In addition, each individual will be required to satisfactorily complete a written and oral examination before taking the NRC examination.

On the basis of its review, the staff finds that the applicant's initial training program conforms to the requirements of the applicable portions of 10 CFR 50 and 55 and follows the guidance in RG 1.8. In addition, the applicant's initial training program conforms to the requirements in H. R. Denton's March 28, 1980, letter. Therefore, the staff concludes that the initial training program for all reactor operators and senior reactor operators is acceptable.

13.2.1.2 Licensed Operator Requalification and Replacement Training Programs

Following the initial licensing of cold-license candidates, requalification and replacement training programs will be implemented to maintain and demonstrate the continued competence and level of proficiency of all licensed personnel.

13.2.1.2.1 Requalification Training Program

A requalification training program conducted by the applicant for all licensed reactor operators and senior reactor operators will be implemented following the initial licensing. This program will consist of the following:

(1) Lecture Series

The requalification program will include planned lectures on a regular and continuing basis. Annual written examination results will indicate the scope and depth needed in the following areas as listed in 10 CFR 55, Appendix A:

(a) theory and principles of operation
(c) plant instrumentation and control systems
(d) plant protection systems
(e) engineered safety systems
(f) normal, abnormal, and emergency operating procedures
(g) radiation control and safety
(h) Technical Specifications
(i) applicable portions of NRC rules and regulations

In addition to the above areas, the lecture series will include instruction in heat transfer, fluid flow, thermodynamics, and mitigation of core damage, as specified in H. R. Denton's March 28, 1980, letter. As appropriate, the lecture series may also cover other topics of pertinent information such as new or modified equipment, problem areas brought out by incidents, and plant evolutions that generally are not performed frequently.

(2) On-the-Job Training

The on-the-job training portion of the requalification program will consist of the following segments:

(a) Control Manipulations

The applicant has indicated that during each 2-year license period, each licensed reactor operator is required to perform all of the control manipulations listed below and each senior operator is required to perform, direct, or evaluate all of these control manipulations:

*o plant or reactor startup (to include a range so that reactivity feedback from nuclear heat addition is noticeable and heatup rate is established)

o plant shutdown

*o manual control of steam generators and/or feedwater during startup and shutdown

o boration and/or dilution during power operation

*o any significant (> 10%) power changes in manual rod control

o any reactor power change of 10% or greater where load change is performed with load limit control

*o loss of coolant

- including significant PWR steam generator leaks
- inside primary containment
- large and small, including leak-rate determination
- including saturated reactor coolant response

o loss of instrument air

o loss of electrical power (and/or degraded power sources)

*o loss of core coolant flow/natural circulation

c loss of condenser vacuum

o loss of service water if required for safety

o loss of residual heat removal (RHR) system

o loss of component cooling system or cooling to an individual component

o loss of all normal feedwater and feedwater system failure

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*Performed on annual basis.

- *° loss of all feedwater (normal and emergency)
- ° loss of protective system channel
- ° mispositioned control rod or rods (or rod drops)
- ° inability to drive control rods
- ° conditions requiring use of emergency boration
- ° fuel cladding failure or high activity in reactor coolant or offgas
- ° malfunction of automatic control system(s) that affect reactivity
- ° malfunction of reactor coolant pressure/volume control system
- ° reactor trip
- ° main steamline break (inside or outside containment)
- ° nuclear instrumentation failure(s)

The manipulations with an asterisk shall be performed on an annual basis; all others shall be performed on a 2-year cycle. An appropriate simulator, which reproduces the general operating characteristics of and has instrument and control arrangements similar to those at Millstone Unit 3, may be used to perform these control manipulations.

The staff finds that the applicant's commitment to the above control manipulations required for licensed operators does comply with the requirements specified in Enclosure 4 of H. R. Denton's letter of March 28, 1980, and is, therefore, acceptable.

(b) Design, Procedure, and Facility License Change Review

This program ensures that licensed reactor operators and senior reactor operators will review revisions to the Operating Technical Specifications and Environmental Technical Specifications, significant procedure changes, and completed facility design changes that would affect plant operations.

(c) Emergency Procedure Review

To ensure a continuing awareness of the action and responses necessary during abnormal and emergency situations, each licensed reactor operator and senior reactor operator will review annually the content of all emergency procedures and all abnormal procedures.

(d) Operator Proficiency Evaluation

Observation and evaluation of the performance of licensed reactor operators and senior reactor operators by supervisors or training staff members will include evaluation of performance during actual or simulated emergency conditions. Observation and evaluation of the performance of licensed personnel during simulated emergency conditions will be conducted by simulator training staff personnel. A poor performance on two or more evaluations will result in a performance review. All evaluations will be critiqued with the individual concerned and filed in the individual's training records.

(3) Simulator Training

The applicant has indicated that whenever it is necessary to fulfill the requirements of 10 CFR 55, Appendix A, each license holder will be required to attend an NRC-approved simulator training program. A formal evaluation will be conducted for each individual's performance during the simulator training. The evaluation will be critiqued with the individual to emphasize any weak areas.

(4) Annual Examination

An annual comprehensive examination, comparable in scope and degree of difficulty to an NRC examination, will be given to each licensed reactor operator and senior reactor operator. The examination, given in two segments, will contain the categories described under "Lecture Series." If an operator receives a grade of less than 80% in any segment or a score of less than 70% in a category, the operator will undergo a performance review. Each license holder's graded examination will be retained as part of the training record.

(5) Performance Review Program

The applicant has indicated that a performance review program will be implemented when the performance of a licensed operator or senior operator falls below the following criteria:

(a) an examination segment score less than 80% or category score less than 70%

(b) poor marks on operator proficiency evaluations

(c) prolonged absence from license responsibilities

A review board, formed by the Operation Supervisor and the Training Supervisor, will determine a course of action necessary to upgrade the individual's performance to an acceptable level. If there is doubt concerning the individual's ability to safely operate the plant, that individual will be removed from licensed responsibilities pending satisfactory completion of the program specified by the review board.

(6) Record Retention

Records of the requalification program, including written examinations, answers, evaluations, and additional training, will be maintained for a period of 24 months to document each licensed operator's and senior operator's participation in the requalification program.

On the basis of its review, the staff finds that the applicant's requalification training program conforms to the requirements of 10 CFR 50 and Appendix A of 10 CFR 55 and follows the guidance in RG 1.8. In addition, the program conforms to the requirements specified in the letter from H. R. Denton to all power reactor applicants and licensees dated March 28, 1980. Therefore, the staff concludes that the applicant's requalification training program is acceptable.

13.2.1.2.2 Replacement Training

Replacement training will be conducted to fill vacancies and prepare individuals for increased responsibility on the supervisory, technical, or operating staff. Replacement personnel will receive training comparable to that received by the initial staff. This will ensure that the required level of proficiency is maintained.

The applicant has indicated that each license candidate will attend an NRC-approved simulator training program. The content details of the program are being developed; however, the applicant has committed to meet, as a minimum, the requirements of H. R. Denton's March 28, 1980, letter regarding the use of simulators for operator training. The staff finds the applicant's commitment acceptable. It will verify that all the above-cited requirements have been met before an operating license is issued.

13.2.1.3 TMI-Related Requirements for New Operating License

I.A.2.1 Immediate Upgrading of Operator and Senior Reactor Operator Training and Qualifications

The applicant has established a program to ensure that all reactor operator (RO) and senior reactor operator (SRO) license candidates have the prescribed experience, qualification, and training.

Each licensed operator candidate will be certified competent to take the NRC license examination by the Vice President, Nuclear Engineering and Operations, before applying for the examination. As an operating license applicant, this applicant is not subject to the 1-year experience requirement for cold-license SRO candidates. However, after 1 year of station operation, individuals applying for an SRO license will be required to comply with the 1-year experience requirement for hot-license SRO applicants, unless they have had experience in an equivalent position at another nuclear plant or at a military propulsion reactor. The experience of license applicants in the latter category will be documented by the applicant on a case-by-case basis in sufficient detail so that the staff can make a finding regarding equivalency. SRO license applicants who possess a degree in engineering or applicable science are considered to meet the 1-year experience requirement as an RO provided they (1) satisfy the requirements set forth in Sections A.1.a and A.2 of Enclosure 1 to the letter from H. R. Denton dated March 28, 1980, to all power reactor applicants and licensees and (2) have participated in a training program equivalent to that of a cold-license SRO applicant.

Also, 3 months of onshift experience as an extra person on shift for control room operators and SRO candidates is not required for cold-license candidates and, thus, is not applicable to Millstone Unit 3. However, the applicant will comply with this requirement for hot-license candidates after 3 months of station operation.

The applicant's training program includes topics in heat transfer, fluid flow, thermodynamics, and reactor and plant transients. All license candidates will attend simulator training programs as part of the initial or replacement training program. In addition, the applicant has committed to provide a simulator training program to all licensed operators as part of the requalification program.

On the basis of its review, the staff concludes that the applicant has satisfied the requirements of this task of the TMI Action Plan.

I.A.2.3 Administration of Training Programs

The applicant has indicated that instructors who will teach licensed operator training or retraining courses covering systems, integrated responses, and transients will be certified or licensed at the SRO level and will participate in appropriate requalification programs. Guest lecturers considered to be used on a limited basis shall be monitored by a qualified instructor. These guest lecturers are exempt from the SRO criterion.

On the basis of its review, the staff concludes that the applicant has satisfied the requirements of this task of the TMI Action Plan.

II.B.4 Training for Mitigating Core Damage

The applicant has indicated that (1) shift technical advisors and personnel in the operating chain up to and including the plant superintendent will receive training for mitigating core damage and (2) supervisors and technicians in the instrumentation and control, health physics, and chemistry departments will receive training for mitigating core damage commensurate with their responsibilities.

On the basis of its review, the staff finds that the applicant has satisfied the requirements of this task of the TMI Action Plan.

13.2.2 Training for Nonlicensed Plant Staff

The applicant has described in the FSAR the details of the training given to nonlicensed plant personnel. The program for nonlicensed personnel will provide training for maintenance, instrumentation and control, radiation protection, radwaste, nuclear physics, management and supervisory, and technical personnel and for training instructors.

All permanently employed plant personnel will participate in a general employee training program consisting of, but not limited to, radiological health and safety, quality assurance, industrial safety, plant security, emergency plan, fire protection, and other appropriate plant plans and procedures.

The applicant has indicated that a long-term training program is provided to upgrade shift supervisors and senior reactor operators for the eventual phase-out of the shift technical advisor (STA) function. Pending a decision on the

NRC proposed Policy Statement that would allow licensees and applicants for operating licenses to combine the SRO and STA functions, the staff will require the applicant to provide for its review a program in accordance with the requirements specified in Appendix C of NUREG-0737 for training STAs.

The fire protection training program includes classroom instruction and training in fire-fighting equipment use, strategies, techniques, and periodic drills. The staff concludes that the applicant's fire protection training program conforms to the guidance given in SRP Section 13.2.2.II.C.A and is acceptable.

On the basis of its review the staff finds that the applicant's training program for nonlicensed plant staff meets the requirements of 10 CFR 19 and 50 and follows the guidance in RG 1.8. Therefore, the staff concludes that the applicant's training program for nonlicensed plant staff, with the exception of the STA training program, is acceptable.

13.3 Emergency Planning

On February 2, 1983, the applicant submitted, pursuant to 10 CFR 50.54(r), a draft emergency plan, dated October 1982. The plan, however, contained only those aspects of emergency planning that were common to the Millstone site, which includes two operating reactors as well as Millstone Unit 3 which is under construction. Specific emergency planning information regarding Millstone Unit 3 has been submitted by the applicant in a revised plan. The staff will review this revised plan and provide a finding as to its adequacy in a supplement to the SER. The Federal Emergency Management Agency is in the process of providing the NRC with its findings on the state of offsite preparedness; these findings will be reported in a supplement to the SER.

13.4 Operational Review

The applicant has established a program for the review and evaluation of operating activities that are important to safety. This program focuses primarily on the provisions the applicant will use to review and evaluate proposed changes, tests, and experiments; the review of unplanned events (such as licensee event reports); and provisions for the evaluation of plant operations.

13.4.1 Plant Staff Review

The applicant currently has two types of onsite review committees because the site contains units of such diverse types. These are a Site Operations Review Committee (SORC), which reviews and advises the Station Superintendent on site or common matters, and a Plant Operations Review Committee (PORC) for each unit, which reviews and advises the Unit Superintendent on unit-specific matters. This concept will be adopted for the Unit 3 plant staff review.

The SORC will be composed of the Station Superintendent, the Unit Superintendent of each of the three units, the Station Services Superintendent and a designated member of each of the unit PORCs. The SORC will meet at least once each month following fuel loading. The assigned responsibilities of the SORC will include review of all common site procedures required by the Technical Specifications, review of plant security and emergency plans, common site proposed tests and experiments that affect nuclear safety, and common site proposed changes or modifications to systems or equipment affecting nuclear safety.

The Unit 3 PORC will be composed of the Superintendent of Unit 3; the Operations Supervisor; the Maintenance Supervisor, the Instrument and Controls Supervisor; the Station Services, General Services, Quality Services, or Radiological Services Supervisor; and a staff engineer. The PORC will meet at least once a month following fuel loading. The assigned responsibilities of the PORC will include those described for such a committee in the Standard Technical Specifications except for those common items that will be reviewed by the SORC as described above. It will act in an advisory capacity to the Superintendent of Unit 3.

13.4.2 Independent Review

Independent review will be performed by two Nuclear Review Boards, which will report to the Senior Vice President, Nuclear Engineering and Operations. The Nuclear Review Board (NRB), which will review Millstone Unit 3 activities, will be implemented 90 days before the scheduled fuel loading of Millstone Unit 3. The Site Nuclear Review Board (SNRB), which reviews activities common to the site and all units at the Millstone station, performs its function at the present time.

The NRB will consist of a chairman and at least four members, who will be appointed in writing by the Senior Vice President, Nuclear Engineering and Operations. Each member will have an academic degree in an engineering or physical science field. In addition, each member will have a minimum of 5 years of technical experience, of which a minimum of 3 years will be in the member's respective field of expertise. Meetings will be held at least once per calendar quarter during the initial year of unit operation and at least once each 6 months thereafter. The NRB will function to provide an independent offsite review of activities as described in Section 6.5.2 of the Standard Technical Specifications except for proposed changes to Section 6.0 of the Technical Specifications (see below).

The SNRB consists of the chairman of each unit NRB and a designated member from each unit NRB. The SNRB will review proposed changes to Section 6.0 of the Technical Specifications that affect all units; any indication of an unanticipated deficiency in some aspect of design or operation of safety-related structures, systems, or components common to all units; and reports and meeting minutes of the SORC. Additionally, audits of site activities will be performed under the cognizance of the SNRB.

13.4.3 Independent Safety Engineering Group

The Independent Safety Engineering Group (ISEG) function will be performed as part of an integrated safety function under the Supervisor, Nuclear Safety Engineering, who reports to the Manager, Nuclear Operations Analysis. The Nuclear Safety Engineering (NSE) section performs an operational safety assessment function for the Millstone and Haddam Neck (Connecticut Yankee) plants. The NSE section currently consists of a supervisor and 13 individuals. This group will be expanded to about 16 individuals by the time of fuel loading of Millstone Unit 3, several of whom will be at the corporate office and eight of whom will be at the Millstone site. The qualification level of persons performing this function will meet or exceed those described in Section 4.7 of ANS 3.1 (1978).

The functions performed by the ISEG will include the independent review and evaluation of plant activities, including maintenance, modifications, and operational problems; operational analysis; and evaluation of plant operations and maintenance activities to provide independent verification that these activities are performed correctly and that human errors are reduced as far as practicable. In implementing these functions, the ISEG will evaluate licensee event reports and plant operating experience, assess plant safety programs, review plant procedures and plant data (for trends), and support the review effort of the NRB.

The ISEG meets the requirements of Task Action Plan Item I.B.1.2 of NUREG-0737 and is acceptable.

13.4.4 Summary and Conclusion

The program includes reviews by the plant staff organization, reviews of safety-related activities independent of the operating organization, and reviews and assessments of plant activities by an independent group located on site. The staff has reviewed the provisions of this program with respect to organizational arrangements, qualification requirements of those performing the review, and the subject matter to be reviewed. The staff finds that the applicant's program for the review of operational activities is in conformance with staff guidance in RG 1.33 and the applicable industry standard (ANSI N18.7); the qualification levels for plant staff personnel performing reviews meet the guidelines of RG 1.8 and the applicable industry standard (ANSI N18.1, Section 4.4); the provisions for an independent review meet the guidelines of RG 1.33 and the applicable industry standards (ANSI N18.7 (ANSI 3.2), Section 4.3, and ANSI/ANS 3.1, Section 4.7); and the applicant's Independent Safety Engineering Group meets the guidelines of Section I.B.1.2 of NUREG-0737 and the acceptance criteria of SRP Section 13.4 (NUREG-0800). Therefore, the applicant's program is acceptable.

13.5 Station Administrative Procedures

13.5.1 Administrative Procedures

The staff has reviewed the administrative procedures according to SRP Section 13.5.1 (NUREG-0800).

All activities affecting nuclear safety will be conducted according to written and approved procedures. Unit 3 operational procedures will be reviewed by the Plant Operations Review Committee and approved by the Unit 3 Superintendent before implementation. Administrative procedures are reviewed by the Site Operations Review Committee and approved by the Station Superintendent. Unit 3 administrative procedures will be those currently being used by Units 1 and 2. RG 1.33 (February 1978), "Quality Assurance Program Requirements," and ANSI 18.7/ANS 3.2-1976, "Administrative Controls for Nuclear Power Plants," will be used for guidance in the preparation of administrative and station procedures.

13.5.1.1 Shift Supervisor Responsibility

The administrative duties of the Shift Supervisor have been reviewed by the Senior Vice President, Nuclear Engineering and Operations. Administrative functions that detract from or are subordinate to the management responsibility for ensuring the safe operation of the plant are delegated to other operational personnel not on duty in the control room.

The Senior Vice President, Nuclear Engineering and Operations, has issued and will periodically reissue a management directive that emphasizes the primary management responsibility of the Shift Supervisor for safe operation of the plant under all conditions on his shift and that clearly establishes his command duties.

13.5.1.2 Limitation on Working Hours

The applicant has established a policy governing the working hours of licensed operators performing safety-related functions to ensure that personnel are in the proper physical condition to operate the plant safely. This policy applies to onshift licensed operators and provides for the following:

(1) An individual should not be permitted to work more than 16 hours straight (excluding shift turnover time).

(2) An individual should not be permitted to work more than 16 hours in any 24-hour period (excluding shift turnover time).

(3) A break of at least 8 hours should be allowed between work periods (including shift turnover time).

(4) An individual should not be permitted to work more than 24 hours in any 48-hour period (excluding shift turnover time).

(5) An individual should not be permitted to work more than 72 hours in any normal work week (excluding shift turnover time).

Deviations from Items 1-3 above may be approved by first-level supervision; the Operations Supervisor, Duty Officer, management representative, or above must approve deviations from Item 4; and the Station Superintendent must approve deviations from Item 5. This policy is currently in effect for Units 1 and 2.

The applicant has not described a policy in respect to other plant staff members that perform safety-related functions, such as auxiliary operators, health physicists, and key maintenance personnel. Until it resolves the applicant's policy governing the limitation on working hours regarding other than onshift licensed personnel, the staff considers this an open item.

13.5.1.3 Shift Relief and Turnover Procedure

Millstone Unit 3 will prepare procedures and checklists for shift personnel to implement shift relief and turnover. These procedures and checklists will be designed to (1) ensure that critical plant parameters are monitored and are within allowable limits, (2) ensure the availability and correct alignment of essential systems, (3) identify all systems or components that are in a degraded mode of operation, and (4) compare the length of time each system or component is in the degraded mode with Technical Specification requirements.

Checklists or logs are provided for completion by the auxiliary operators and technicians when they arrive and leave. The checklists or logs include any equipment under maintenance or test that, by itself, could degrade a system critical to the prevention and mitigation of operational transients and accidents or initiate an operational transient. Criteria for acceptable status are included in the checklist.

13.5.1.4 Control Room Access

The applicant has prepared and implemented a procedure to limit access to the control room. This procedure is currently in effect for Connecticut Yankee and Millstone Units 1 and 2. The Shift Supervisors are responsible for controlling access to the control room. Access will be limited to those authorized personnel who have official business in the control room. In the absence of the Shift Supervisor, the supervising control operator will exercise this responsibility.

13.5.1.5 Feedback of Operating Experience

The Nuclear Safety Engineering (NSE) Unit, reporting to the Manager, Nuclear Operations Analysis, has the lead responsibility for performance of the operating experience assessment and feedback function. This unit will use the Institute of Nuclear Power Operations SEE-IN Program for reviewing and assessing information on external operating experience. This will be in the form of significant event reports and significant operating event reports. In addition, this unit reviews all in-plant licensee event reports. The Supervisor, NSE, is responsible for screening information on operating experience, conducting analyses on screened information, obtaining approval of recommendations, and maintaining records of information received on operating experience through the closeout of recommended actions. This program is in effect for Units 1 and 2.

The Vice President, Nuclear Operations, is responsible for reviewing NRC bulletins, circulars, and information notices; initiating corrective actions as necessary; preparing responses to the NRC when required; and closing out items. The Station Superintendent and Nuclear Operations Engineer provide support for the review and response as requested.

13.5.1.6 Verification of Correct Performance of Operating Activities

The applicant has has established a procedure to implement a system for the verification of operating activities important to safety at the Millstone station. This procedure is currently in effect for Millstone Units 1 and 2 and will apply to Unit 3.

The applicant has established a procedure that assigns responsibility to the Shift Supervisor for clearance on plant equipment for maintenance and construction. In the absence of the Shift Supervisor, a supervising control operator holding an SRO license may assume the Shift Supervisor's responsibilities.

Following all maintenance or surveillance activities, the Unit Operations Department performs an independent position verification of repositioned valves, circuit breakers, and control switches of systems that are important to safety and safety related. Following a cold shutdown outage, the Unit Operations Department performs an independent position verification of all valves, circuit breakers, and control switches of systems that are important to safety and safety related, as designated by the Operations Supervisor. This shall include valves, circuit breakers, and control switches of nonremote indication.

An independent verification of valves, circuit breakers, and control switches will be performed by qualified, licensed or nonlicensed operators.

13.5.1.7 Initial Test Program

Test procedures will be prepared by or under the cognizance of the Northeast Nuclear Energy Company (NNECo) staff using guidelines provided by Stone & Webster Engineering Company (SWEC) and/or Westinghouse (W). After the test procedure is written, it is reviewed by selected members of the NNECo, Northeast Utilities Services Company, Nuclear Engineering and Operations, SWEC, and/or W staff. Westinghouse is responsible for the review of preoperational and subsequent startup tests involving W-supplied systems. These reviews are coordinated by the NNECo Startup Supervisor, and review comments are resolved by the originator of the test procedure. Preoperational and subsequent test procedures are reviewed by the Millstone Unit 3 PORC.

13.5.1.8 Summary and Conclusions

The applicant has described the program and procedures that provide administrative controls over activities important to safety. These include (1) the preparation, review, and approval of plant operating and maintenance procedures; (2) the responsibilities and duties of shift personnel; (3) shift relief and turnover procedures; (4) access to the control room; (5) limitations on working hours; (6) the feedback of information on operating experience to plant personnel; (7) the procedure for verifying the correct performance of operating activities; and (8) the administrative provisions for the control of the initial plant test program. The staff has reviewed these provisions and finds that they meet the guidance in Section 5.2 of ANSI/ANS 3.2, RG 1.33, and the applicable parts of Task Action Plan Items I.A.1.2, I.A.1.3, I.C.2, I.C.3, I.C.4, I.C.5, and I.C.6, except as noted above with respect to the limitation on working hours and verification of the correct performance of operating activities. Therefore, except for the two open items, the staff concludes that the administrative procedures are acceptable and contribute to meeting the requirements of 10 CFR 50.40(b) and 10 CFR 50.54(1).

13.5.2 Operating and Maintenance Procedures

13.5.2.1 General

The staff has reviewed the applicant's plan for development and implementation of operating and maintenance procedures to determine the adequacy of the applicant's program for ensuring that routine operating, offnormal, and emergency activities are conducted in a safe manner. The following description and evaluation are based on information contained in the applicant's FSAR through Amendment 7, and in an April 15, 1983, letter from W. G. Counsil of Northeast Nuclear Energy Company, to D. G. Eisenhut, NRC, and in an April 19, 1984, letter from W. G. Counsil to B. J. Youngblood, NRC.

In determining the acceptability of the applicant's program, the criteria of SRP Section 13.5.2 (NUREG-0800) were used. The review consisted of an evaluation of the following:

(1) the applicant's procedure classification system for procedures that are performed by licensed operators in the control room and the classification for other operating and maintenance procedures

(2) the applicant's plan for completion of operating and maintenance procedures during the initial plant testing phase to allow for correction before fuel loading

(3) the applicant's program for compliance with the guidance contained in RG 1.33, Revision 2 (February 1978) regarding the minimum procedural requirements for safety-related operations

(4) conformance with the guidance contained in ANSI N18.7-1976/ANS 3.2 (American National Standards Institute/American Nuclear Society) and Section 5.3 of ANSI/ANS 3.2-1981 (Draft 7)

(5) the applicant's program for compliance with NUREG-0737, Supplement 1, "Requirements for Emergency Response Capability," for the development of emergency operating procedures (EOPs)

13.5.2.2 Operating and Maintenance Procedures Program

The applicant has committed in the FSAR to a program that uses RG 1.33, Revision 2 (February 1978), and ANSI N18.7-1976/ANS 3.2 as guidance for the preparation of administrative and station procedures.

As described in the FSAR, the applicant uses the following categories of procedures for those operations performed by operators in the control room:

(1) general operating
(2) system operating
(3) abnormal operating
(4) emergency operating

Other procedures include the following areas:

(1) health physics
(2) emergency preparedness
(3) instrument maintenance instructions
(4) chemistry
(5) radioactive waste
(6) maintenance and modication
(7) fire protection

The staff's review determined that, with the exceptions described below, the applicant's program for use of operating and maintenance procedures meets the relevant requirements of 10 CFR 50.34 and is consistent with the guidance provided in RG 1.33, Revision 2, and ANSI N18.7-1976/ANS 3.2.

The applicant has also committed to implement Supplement 1 to NUREG-0737 and to submit his procedures generation package (PGP) on October 1, 1984, which is 3 months before the start of operator training on the Millstone Unit 3 simulator. The PGP will be based on staff-approved Westinghouse revised Emergency Response

Guidelines. The staff finds the commitment to Supplement 1 to NUREG-0737 and the October 1, 1984, PGP submittal date acceptable.

The applicant did not commit to implement Section 5.3 of ANSI/ANS 3.2-1981 (Draft 7) in accordance with the SRP. Section 5.3 addresses the need for symptom/functional-based EOPs. However, because the applicant has explicitly committed to develop functional-based EOPs based on NRC-approved procedure guidelines, the staff finds the existing commitment acceptable. The staff will require additional information on one item before it can conclude that the applicant's program fully meets the guidance provided in SRP Section 13.5.2. The item requiring resolution is TMI-2 Task Action Plan (TAP) Item I.C.1 as described in Section 13.5.2.3.

13.5.2.3 Reanalysis of Transients and Accidents; Development of Emergency Operating Procedures

In letters dated September 13 and 27, October 10 and 30, and November 9, 1979, the staff required licensees of operating plants, applicants for operating licenses, and licensees of plants under construction to

(1) perform analyses of transients and accidents
(2) prepare EOP guidelines
(3) upgrade EOPs
(4) conduct operator retraining (see also NUREG-0737, Item I.A.2.1)

EOPs are required to be consistent with the actions necessary to cope with the transients and accidents analyzed. Clarification of the scope of the task and appropriate schedule revisions were included in NUREG-0737, Item I.C.1. On December 17, 1982, NRC issued Generic Letter 82-33 (Supplement 1 to NUREG-0737), which clarified Item I.C.1 of NUREG-0737 and required development and submittal of procedures generation packages (PGPs) to NRC.

The NRC staff reviewed the proposed Westinghouse Owners' Group Emergency Response Guidelines (ERGs) as described in Westinghouse Owners' Group letters of November 30, 1981, July 21, 1982, and January 4, 1983, and in the material accompanying those letters. The staff concluded in Generic Letter 83-22, dated June 3, 1983, that the guidelines are acceptable for implementation.

In a letter dated April 15, 1983, from W. G. Counsil to D. G. Eisenhut, the applicant committed to implement the provisions of Supplement 1 to NUREG-0737 and to prepare the PGP and plant-specific EOPs on the basis of the NRC-approved Westinghouse ERGs. The PGP described in Supplement 1 to NUREG-0737 must be submitted for staff review. The PGP should be submitted as an FSAR amendment because it provides the basis for developing the plant's EOPs. The PGP is scheduled for submittal to NRC on October 1, 1984, 3 months before the start of control room operator training on the Millstone Unit 3 simulator.

The staff's review of the PGP will confirm that the PGP provides adequate guidance for developing EOPs. The results of the review will be addressed in a supplement to the SER. Until completion of the staff review of the PGP, Task Action Plan Item I.C.1 will remain a confirmatory item.

In accordance with NUREG-0737, Item I.C.7, nuclear steam supply system (NSSS) vendor review of low-power testing, power ascension, and emergency operating procedures was necessary to further verify adequacy of the procedures. Because the applicant has committed to implement procedures based on the NRC-approved Westinghouse ERGs, the staff does not consider an additional NSSS vendor review of the EOPs necessary. In addition, because an NSSS vendor representative is a member of the Joint Test Group (Section 14.2.2.2.6 of the FSAR) that reviews operating and testing procedures, the staff does not consider an additional NSSS review of low-power and power-ascension testing procedures necessary. The staff considers TMI Task Action Plan Item I.C.7 resolved.

The applicant committed in the FSAR to support TMI Task Action Plan Item I.C.8, "Pilot Monitoring of Selected Emergency Procedures for Near-Term Operating License Applicants." This pilot monitoring program was used on an interim basis to evaluate the applicant's emergency operating procedures before staff approval of generic technical guidelines and staff development of the long-term program for upgrading of emergency operating procedures. This is no longer necessary as a result of NRC approval of the Westinghouse ERGs and the applicant's commitment to develop the EOPs based on the ERGs. The staff considers Task Action Plan Item I.C.8 resolved.

13.6 Physical Security Plan

The applicant has filed with the NRC a draft physical security plan that encompasses Units 1, 2, and 3. The staff has conducted a preliminary review and concluded that, with certain changes, the physical security plan will comply with regulatory requirements. The guard training and qualification and safeguards contingency plans were previously reviewed and approved during the safeguards review of Units 1 and 2.

The staff considers the physical security plan a confirmatory item and expects that all issues will be resolved before the final plan is submitted.

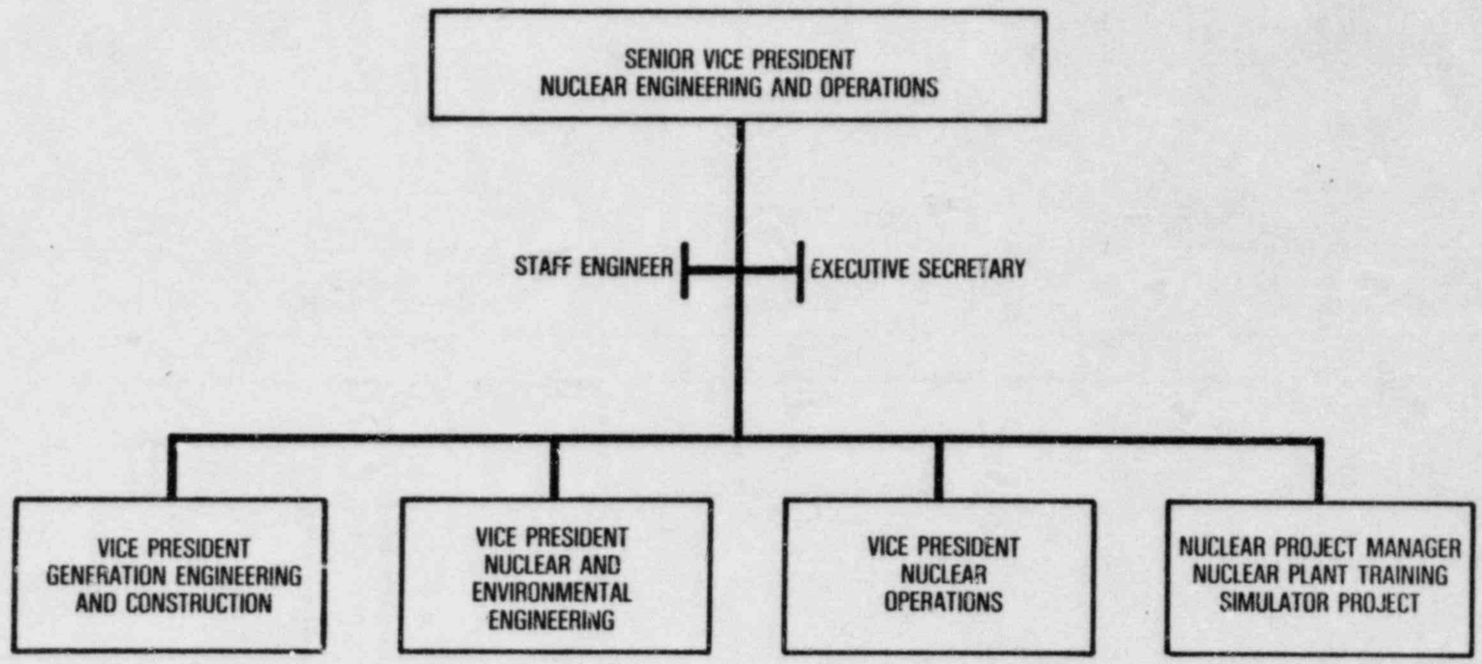

Figure 13.1 Nuclear Engineering and Operations Group

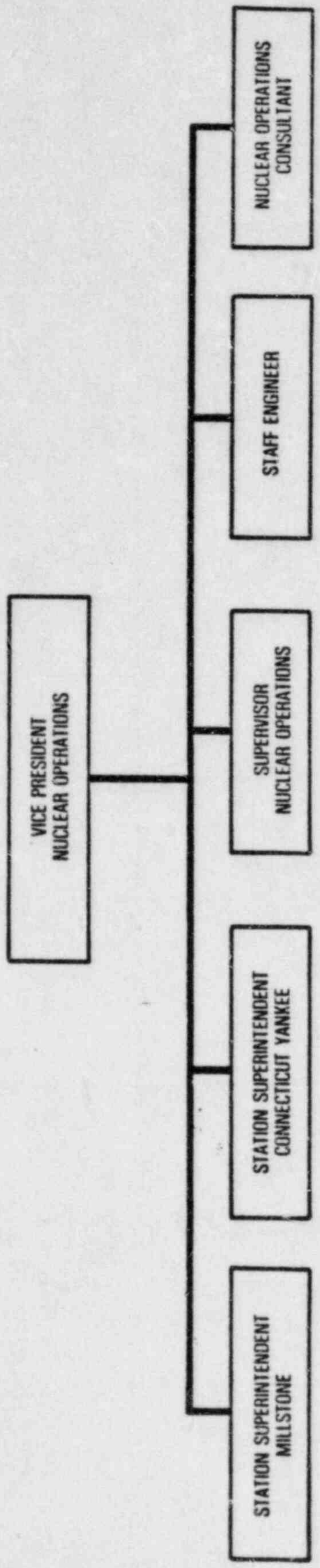

Figure 13.2 Nuclear Operations Department

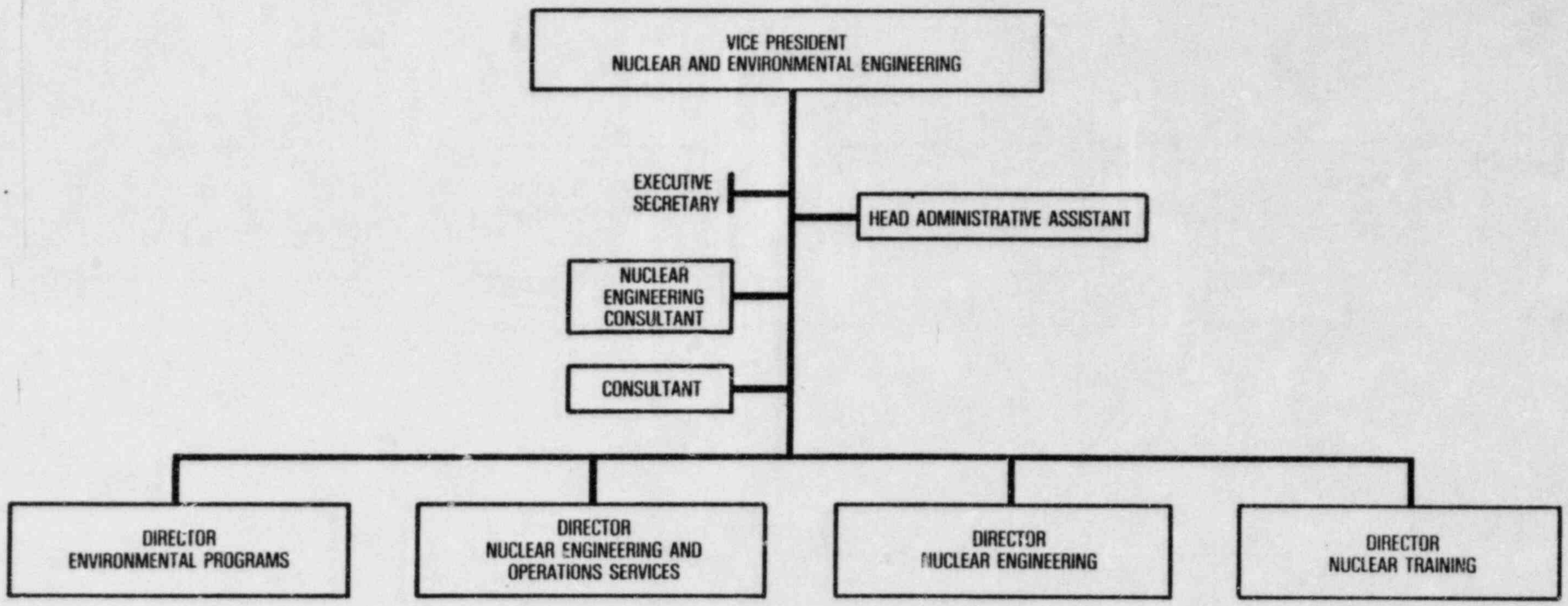

Figure 13.3 Nuclear and Environmental Engineering Division

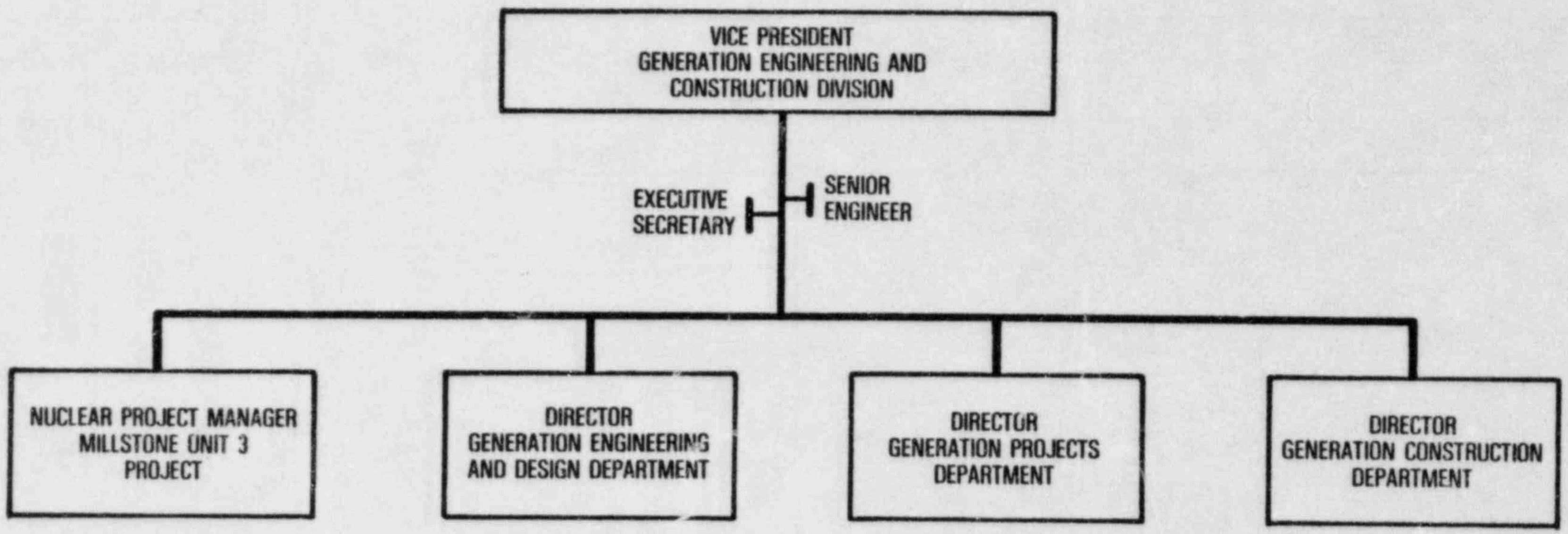

Figure 13.4 Generation Engineering and Construction Division

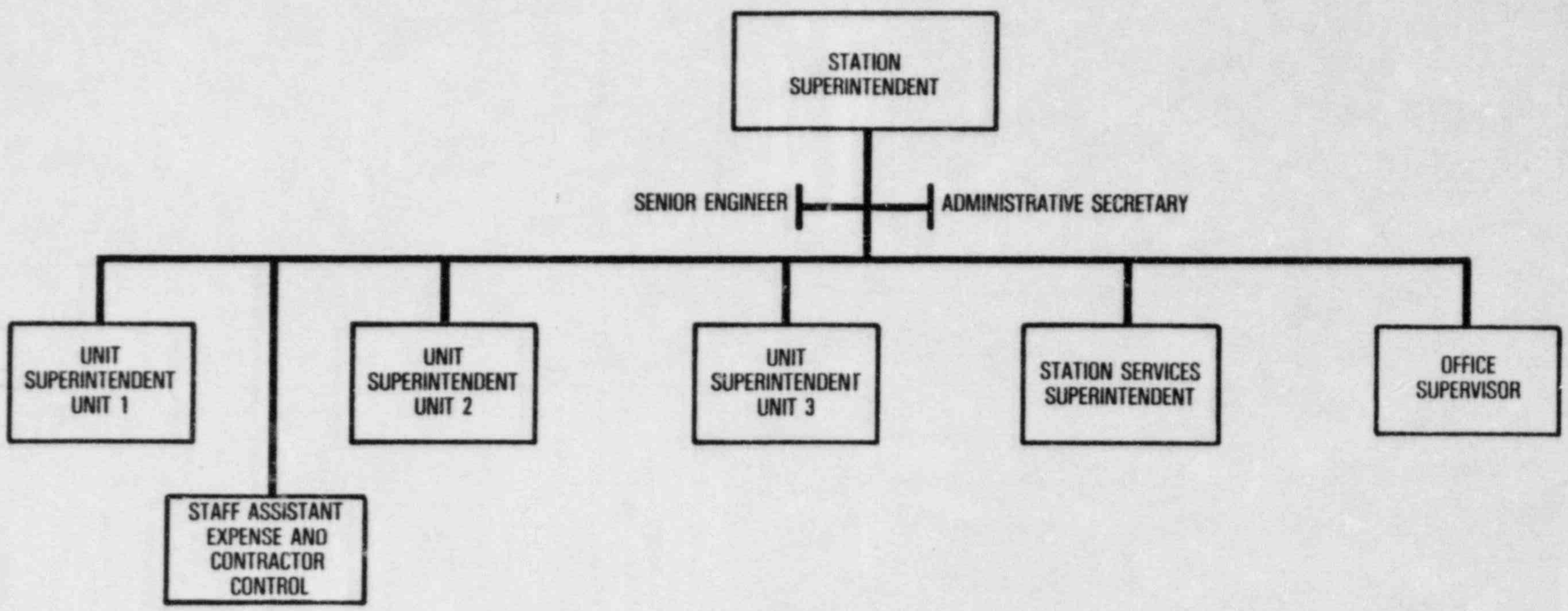

Figure 13.5 Millstone station organization

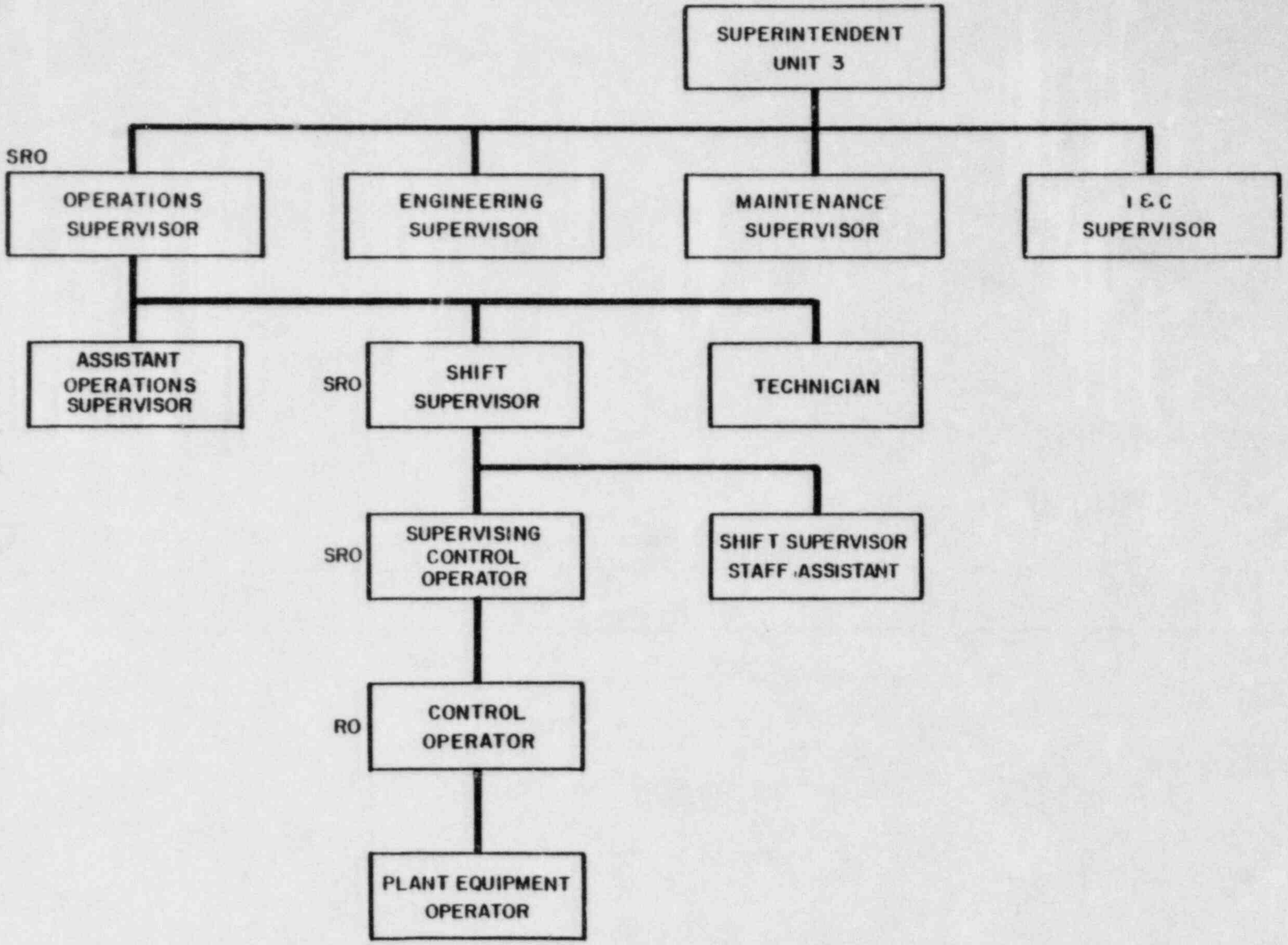

Figure 13.6 Millstone Unit 3 organization

14 INITIAL TEST PROGRAM

The initial test program for Millstone Unit 3 has been reviewed in accordance with SRP Section 14.2 (NUREG-0800).

The initial test program encompasses the scope of events that begins with the completion of system construction and ends with the completion of power-ascension testing. The objectives of the test program are to provide assurance that

(1) The plant has been properly designed and constructed and is ready to operate in a manner that will not endanger the health and safety of the public.

(2) The plant procedures have been demonstrated to be adequate.

(3) The operating organization is knowledgeable about the plant and procedures and is prepared to operate the plant in a safe manner.

The initial test program for Millstone Unit 3 will be accomplished in nine distinct and sequential major phases.

Phase I - Initial Inspection and Component Testing

These tests include phase rotation, insulation resistance, and valve operator tests, coupling alinement checks, cleanliness tests, and similar tests. These tests will be conducted to ensure that equipment and components are ready for operation.

Phase II - Preoperational and Acceptance Testing

These tests demonstrate, to the extent practical, the capability of structures, systems, and components to meet performance requirements to satisfy design criteria. Additionally, these tests verify the operational reliability of systems, components, and protective devices to ensure that equipment can be operated safely during integrated plant testing and that performance will be in accordance with design criteria.

(1) Preoperational Tests

In general, preoperational testing is performed on quality assurance (QA) Category I systems and structures and on those non-Category I systems that normally handle radioactive materials or provide direct support to a Category I system. (QA Category I is defined in the, "Northeast Utilities Quality Assurance Program Topical Report," NU-QA-1.)

Test procedures include, as appropriate, manual system or component operation, operation of subsystems and components within systems, automatic operation of systems and components, operation in all alternate or secondary modes of control, and operation and verification tests to demonstrate expected operation following loss of power sources and failures of components for which the systems are designed to remain operational. Preoperational test procedures also include, as appropriate, verification of proper functioning of instrumentation and controls, permissive and prohibit interlocks, and equipment protective devices whose malfunction or premature actuation may unnecessarily shut down or defeat the operation of systems or equipment.

(2) Acceptance Tests

Acceptance tests are normally performed on non-Category I systems that are not preoperationally tested.

Acceptance tests are performed to demonstrate that applicable systems meet design requirements. The testing method and format for acceptance test procedures is similar to that employed for preoperational testing.

Phase III - Hot Functional Testing Before Fuel Loading

Before fuel loading the reactor coolant pumps and the pressurizer heaters will be operated to bring the reactor coolant system (RCS) to normal operating temperature and pressure. Tests will be conducted to ensure that normal and emergency core cooling systems perform in accordance with design criteria. These tests are performed to ensure that when the fuel is loaded into the core, it can be cooled under all plant conditions.

Phase IV - Initial Fuel Loading

The plant operating staff, under the direction of the plant reactor engineer, will use written and approved procedures to perform the initial fuel loading. These procedures will include precautions to preclude inadvertent criticality.

Phase V - Hot Functional Testing After Fuel Loading

After fuel loading and before initial criticality, the reactor coolant pumps and pressurizer heater will again be used to heat up and pressurize the RCS. Additional hot functional tests will be conducted to validate design criteria used in accident analyses to ensure protection of the general public in an accident situation. These tests also ensure the reliability of the control rod drive system.

Phase VI - Initial Criticality

The purpose of the initial approach to criticality procedure is to provide a safe and controlled method of achieving initial reactor criticality.

Phase VII - Low-Power Physics Testing

The purpose of the low-power physics testing program is to obtain as-built reactor characteristics and to verify Westinghouse predictions and physics design parameters.

Phase VIII - Power-Ascension Testing

The power-ascension test procedures describe the detailed steps required for the initial power startup from completion of the low-power physics test phase to the full-rated power level, including those tests necessary to demonstrate safe plant operation within design specifications.

Phase IX - Warranty Run

When Phase VIII is completed, the plant is operated for a period of at least 100 hours at full power to determine the performance of all systems and equipment under sustained full-power conditions. When Phase IX has been satisfactorily completed, the startup test program is complete and the plant can begin the in-service phase.

The staff's review of FSAR Chapter 14 concentrated on the administration of the test program and the completeness of the preoperational and startup tests (Phases II through VIII). Additionally, the staff reviewed FSAR Chapter 14, the SER-CP, other FSAR chapters, Licensee Event Report summaries, and Standard Technical Specifications. Post-TMI-related testing requirements were examined according to NUREG-0660, NUREG-0694, and NUREG-0737. And finally, startup test reports for other Westinghouse reactor plants were examined to identify problem areas that should be emphasized in the initial test program.

The staff's review included verification of the following features of the applicant's description of the initial test program:

(1) The applicant plans to develop test procedures using input from the nuclear steam supply system (NSSS) vendor, the architect-engineer, the applicant's engineering staff, and other equipment suppliers and contractors. Operating experiences at similar plants will be factored into the development of the test procedures.

(2) The applicant plans to conduct tests using approved test procedures. Administrative controls cover (a) the completion of test prerequisites, (b) the completion of necessary data sheets and other documentation, and (c) the review and approval of modifications to test procedures. The applicant has stated that administrative procedures also cover implementation of modifications or repair requirements identified as being required by the tests, and any necessary retesting.

(3) The applicant plans to review the results of each test for technical adequacy and completeness using review groups that include the NSSS vendor and the architect-engineer, as appropriate. Preoperational test results will be reviewed before fuel loading, and the startup test results from each test condition or power level will be reviewed before proceeding to the next test condition or power level.

(4) The applicant plans to use normal plant operating and emergency operating procedures in performing the initial test program, thereby verifying the correctness of the procedures to the extent practicable.

(5) The schedule for conducting the initial test program allows adequate time to conduct all preoperational and startup tests. Preoperational test procedures will be available for NRC Regional Administrator review at least 60 days before scheduled implementation. Startup test procedures will be available for review not less than 60 days before the scheduled fuel-loading date.

(6) A description of each test is presented in FSAR Chapter 14. The staff verified that there are test descriptions for those structures, systems, components, and design features that

   (a) will be used for shutdown and cooldown of the reactor under normal, transient, and accident conditions and for maintaining the reactor in a safe condition for an extended shutdown period

   (b) will be used for establishing conformance with safety limits or limiting conditions for operation that will be included in the facility Technical Specifications

   (c) are classified as engineered safety features or will be relied on to support or ensure the operations of engineered safety features within design limits

   (d) are assumed to function or for which credit is taken in the accident analysis of the facility, as described in the FSAR

   (e) will be used to process, store, control, or limit the release of radioactive materials.

(7) The test objectives, prerequisites, test methods, and acceptance criteria for each test description are in sufficient detail to establish that the functional adequacy of the structures, systems, components, and design features will be demonstrated.

(8) Exceptions to RG 1.68, Revision 2, are identified and adequate technical justification is provided.

In a letter from B. J. Youngblood (NRC) to W. G. Counsil (Northeast Utilities) dated June 29, 1983, the staff forwarded requests for additional information regarding the FSAR Chapter 14 review. The applicant forwarded responses to these questions in letters from W. G. Counsil to B. J. Youngblood dated August 29, 1983, and September 27, 1983.

The applicant has made a number of changes to the initial test program as a result of the staff's questions. Examples of these changes follow:

(1) expanded the sources of acceptance criteria, in the summary FSAR Table 14.2-3, for traceability and improved site inspector access

(2) increased the testing for the instrument air preoperation test to provide for testing of sudden loss of pressure on each individual valve

(3) expanded testing to include the failed fuel monitoring system

(4) expanded test abstract descriptions to include the sources of acceptance criteria

(5) included a verification that a manual trip will remove power from the reactor trip breaker undervoltage coil and energize the shunt trip coil

(6) clarified and amplified the natural circulation testing related to TMI Action Plan Item I.G.1

(7) expanded the temperature testing of containment penetrations

(8) modified the station blackout test to ensure operation on battery power only

Resolution of the issues remaining from the review was submitted by letters from W. G. Counsil to B. J. Youngblood dated April 19, 1984, and May 15, 1984. The staff has evaluated these submittals and accepts the applicant's responses. These responses will be confirmed by FSAR amendment at a later date.

On the basis of its review, the staff has concluded that the initial plant test program is acceptable and meets the requirements of 10 CFR 50.34(b)(6)(iii); 10 CFR 50, Appendix B, Section XI; and NUREG-0737, Item I.G.1. The staff has furthur concluded that the initial test program described in the application will meet the acceptance criteria of SRP Section 14.2 (NUREG-0800) and that the successful completion of the test program will demonstrate the functional adequacy of plant structures, systems, and components.

Because the holder of an operating license has the legal option to make changes to the initial test program pursuant to 10 CFR 50.59, the staff will condition the operating license to require the applicant to complete the initial startup test program as described in the FSAR without making any major modifications. Major modifications will be defined in the license as

(1) elimination of any test described in FSAR Chapter 14 and not identified therein as being nonessential

(2) modification of test objectives, methods, or acceptance criteria for any test described in FSAR Chapter 14 and not identified therein as being nonessential

(3) performance of any test at a power level different from that stated in the FSAR

(4) failure to complete any tests included in the described program (planned or scheduled for power levels up to authorized power level)

15 ACCIDENT ANALYSES

The accident analyses for Millstone Unit 3 have been reviewed in accordance with SRP Section 15 (NUREG-0800). Conformance with the acceptance criteria, except as noted for each of the sections, formed the basis for concluding that the design of the facility for each of the areas reviewed was found to be acceptable for Millstone Unit 3.

In accordance with SRP Section 15.1.1.I, the applicant evaluated the capability of the Millstone facility to withstand anticipated operational occurrences and a broad spectrum of postulated accidents without undue hazard to the health and safety of the public. The results of these analyses are used to show conformance with GDC 10 and 15.

For each event analyzed, the worst operating condition and single failure were assumed and credit was taken for minimum engineered safeguards response. Parameters specific to individual events were conservatively selected. Two types of events were analyzed:

(1) those incidents that might be expected to occur during the lifetime of the reactor (anticipated transients)

(2) those incidents not expected to occur that have the potential to result in significant radioactive material release (accidents)

The nuclear feedback coefficients were conservatively chosen to produce the most adverse core response. The reactivity insertion curve, used to represent the control insertion, accounts for a stuck rod in accordance with GDC 26.

Review of the thermal-hydraulic code THINC-IV is described in Section 4.4 of this SER. The staff review of the FACTRAN code has progressed to the point that there is reasonable assurance that results of the analyses dependent on the code will not be appreciably altered by any revisions that may be required by the staff.

For transients and accidents, the applicant used a method that conservatively bounds the consequences of the event by accounting for fabrication and operating uncertainties directly in the calculations. Departure from nucleate boiling ratios (DNBRs) were calculated using the W-3 correlation; a minimum DNBR of 1.3 was used as the threshold for fuel failure.

The applicant accounts for variations in initial conditions by making the following assumptions as appropriate for the event being considered:

(1) core power, 3,425 MWt, ±2%
(2) average reactor vessel temperature (T_{avg}), 587.1° ± 6.5°F
(3) pressure (at pressurizer), 2,250 ± 30 psi

The values the applicant has chosen as the initial conditions for the transients and accidents described in the following sections are the nominal values with given great instrumentation uncertainties. However, as a result of reviews of Technical Specifications of plants with near-term operating licenses, the staff is concerned that these initial conditions may not be ensured by the Technical Specifications. The applicant must ensure that the Technical Specifications will restrict operation of the plant within the values assumed in the safety analyses.

The staff concludes that the assumptions for initial conditions are acceptable because they are conservatively applied to produce the most adverse effects. For transients and accidents used to verify the engineered safety features design, the applicant has used the safeguards power design value of 3,579 MWt.

The applicant has analyzed several events expected to occur one or more times during the life of the plant. A number of transients can be expected to occur with moderate frequency as a result of equipment malfunctions or operator error in the course of refueling and power operation during the plant lifetime. Specific events were reviewed to ensure conformance with the acceptance criteria in the SRP.

The acceptance criteria for transients of moderate frequency in the SRP include the following considerations:

(1) Pressure in the reactor coolant and main steam systems should be maintained below 110% of design values (ASME Code, Section III).

(2) Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR will remain above the 95/95 DNBR limit for PWRs. (The 95/95 criterion discussed in SRP Section 4.4 provides a 95% probability at a 95% confidence level that no fuel rod in the core experiences a departure from nucleate boiling.)

(3) An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

(4) For transients of moderate frequency in combination with a single failure, no loss of function of any fission product barrier, other than fuel element cladding, shall occur. Core geometry is maintained in such a way that there is no loss of core cooling capability and control rod insertability is maintained.

Conformance with SRP acceptance criteria constitutes compliance with GDC 10, 15, and 26 of Appendix A to 10 CFR 50.

The transients analyzed are protected by the following reactor trips:

(1) power range high neutron flux
(2) high pressure
(3) low pressure
(4) overpower ΔT
(5) overtemperature ΔT
(6) low reactor coolant flow
(7) reactor coolant pump shaft low speed
(8) low steam generator water level
(9) high steam generator water level

Time delays to trip, calculated for each trip signal, are included in the analyses. See Section 7.2 of this SER for a discussion of the staff review of reactivity control system functional design.

All of the transients that are expected to occur with moderate frequency can be grouped according to the following plant process disturbances: undercooling transients, increased cooling transients, changes in coolant inventory, and changes in core reactivity.

15.1 Increase in Heat Removal by the Secondary System

The applicant's analyses of events that produce increased heat removal by the secondary system are addressed in the following paragraphs.

15.1.1 Decrease in Feedwater Temperature

See Section 15.1.4.

15.1.2 Increase in Feedwater Flow

See Section 15.1.4.

15.1.3 Excessive Increase in Steam Flow

See Section 15.1.4.

15.1.4 Inadvertent Opening of a Steam Generator Relief Valve or Safety Valve

The most limiting transient with respect to fuel performance is the inadvertent opening of the steam generator relief or safety valve. The increased steam demand causes a reactor power increase that results in a reactor trip. The continued steam flow through the open valve will cause additional cooldown and additional positive reactivity insertion to the primary coolant system. The safety injection system (SIS) will inject highly concentrated boric acid from the refueling water storage tank into the primary coolant system on either two out of four pressurizer low pressure signals, two out of three high containment pressure signals, or two out of three low steamline pressure signals in any one loop. This ensures the reactor will remain shut down with any subsequent cooldown. The normal steam generator feedwater supply will be isolated automatically upon SIS initiation and then an orderly cooldown will be effected. The transient is terminated with the use of only safety-related equipment. Departure from nucleate boiling does not occur during this transient.

The transient that is most limiting with respect to the peak pressure is the increase in feedwater flow. The applicant has calculated a peak pressure of 2,302 psig during this transient, which is well below the system design pressure of 2,485 psig.

15.1.5 Steamline Rupture

The applicant has submitted analyses of postulated steamline breaks that show no fuel failures are attributed to the accident. These results are similar to those obtained for previously reviewed Westinghouse four-loop plants.

A postulated double-ended rupture at hot standby power with offsite power available was analyzed as the worst case. The applicant referenced Westinghouse Topical Report WCAP-9227 as justification for this selection. WCAP-9227 is currently under review by the staff. This review has progressed to the point that there is reasonable assurance that the analysis results presented in this topical report will not be appreciably altered by any revisions that may be required by the staff.

The double-ended rupture would cause the reactor to increase in power because of the decrease in reactor coolant temperature. The reactor would be tripped by either reactor overpower ΔT or by the actuation of the safety injection system. The safety injection system will be actuated by any of the following: two out of four low pressurizer pressure signals, two out of three high containment pressure signals, or two out of three low steamline pressure signals in any one loop.

Although a return of criticality will occur, there is no fuel damage predicted since the minimum DNBR will remain greater than 1.30.

In a meeting held at NRC Headquarters on May 9, 1984, the staff requested the applicant to provide the most limiting single failure assumed for this analysis. The applicant has provided the information to indicate that the failure of a safeguards train to deliver borated water to the reactor coolant system is the most limiting single failure. The staff finds the information acceptable.

The staff concludes that the consequences of postulated steamline breaks meet the relevant requirements in GDC 27, 18, 31, and 35 regarding control rod insertability and core coolability and TMI Action Plan items. This conclusion is based on the following:

(1) The applicant has met the requirements of GDC 27 and 28 by demonstrating that the resultant fuel damage was limited so that control rod insertability would be maintained and no loss of core cooling capability resulted. The minimum DNBR experienced by any fuel rod was greater than 1.30, resulting in no rod experiencing cladding perforation.

(2) The applicant has met the requirements of GDC 31 with respect to demonstrating the integrity of the primary system boundary to withstand the postulated accident.

(3) The applicant has met the requirements of GDC 35 with respect to demonstrating the adequacy of the emergency core cooling systems to provide abundant core cooling and reactivity control (via boron injection).

(4) The parameters used as input to this model were reviewed and found to be suitably conservative.

(5) The applicant has met the requirements of Task Action Plan Items II.E.1.1 and II.E.1.2 with respect to demonstrating the adequacy of the auxiliary feedwater system design to remove decay heat following steam system piping failure.

(6) The applicant has met the requirements of Task Action Plan Item II.K.3.25 with respect to demonstrating the integrity and operation of the reactor coolant pumps to withstand the postulated accident. (See Section 15.9.12.)

(7) The applicant has met the requirements of Task Action Plan Item II.K.3.5 with respect to the operation and tripping of the reactor coolant pumps. The assumptions used are conservative and consistent with the generic resolution to Item II.K.3.5. (See Section 15.9.12.)

15.2 Decrease in Heat Removal by the Secondary System

The applicant's analyses of events that result in a decrease in heat removal by the secondary system are addressed in the following paragraphs.

15.2.1 Loss of External Load

See Section 15.2.7.

15.2.2 Turbine Trip

See Section 15.2.7.

15.2.3 Loss of Condenser Vacuum

See Section 15.2.7.

15.2.4 Inadvertent Closure of Main Steam Isolation Valve

See Section 15.2.7.

15.2.5 Steam Pressure Regulator Failure

See Section 15.2.7.

15.2.6 Loss of Nonemergency Power to the Station Auxiliaries

See Section 15.2.7.

15.2.7 Loss of Normal Feedwater Flow

Plant transients that result in an unplanned decrease in heat removal by the secondary system that might be expected to occur with moderate frequency are identified above. All these postulated transients have been reviewed. It was found that the most limiting pressurization event within the reactor coolant and main steam systems for this group of events was the loss of normal feedwater caused by a loss of offsite power. The reactor is tripped on the high pressurizer pressure signal, and the peak pressure during the transient is 2,565 psia, well below the 110% design pressure criterion, which is 2,750 psia.

The applicant stated in FSAR Section 15.2.7 that the most limiting event with respect to fuel performance and maximum pressure within the reactor coolant and main steam system is the loss of normal feedwater caused by a loss of offsite ac power. In this transient, the loss of offsite power is closely followed by a turbine trip and reactor trip. Assuming the worst single failure in the auxiliary feedwater system (AWFS) occurs (i.e., failure of a redundant AFWS train), the AFWS is automatically started but only one auxiliary feedwater pump is assumed to be feeding two steam generators. It is also assumed that only safety-related equipment is used to mitigate the event. All residual heat must be removed through the steam generator atmospheric steam dump valves, which are safety-related components. The first few seconds after the loss of normal feedwater transient will closely resemble a simulation of the complete loss of forced reactor coolant flow event (discussed in FSAR Section 15.3.2). The DNBR is always greater than 1.30. The peak pressure during the transient is 2,565 psia, well below the SRP criterion that maximum pressure be limited to 110% of design pressure.

15.2.8 Feedwater System Pipe Break

The applicant has provided a feedwater line break analysis for Millstone using assumptions that would minimize secondary system heat removal capability, maximize heat addition to the primary system coolant, and maximize the calculated primary system pressure.

The system code used to perform these analyses is LOFTRAN. The analysis assumed a double-ended rupture of the largest feedwater line and the most restrictive single failure of the auxiliary feedwater system (i.e., failure of an auxiliary feedwater train), and that emergency feedwater flow is supplied to only two intact steam generators. There is sufficient feedwater flow to adequately remove the residual heat after reactor shutdown. The use of only safety-related equipment is sufficient to mitigate this accident. No fuel damage was calculated to occur, and the peak calculated pressurizer pressure was about 2,500 psia. The staff finds these results to be within the required limits.

15.3 Decrease in Reactor Coolant Flow Rate

15.3.1/15.3.2 Loss of Forced Reactor Coolant Flow, Including Trip of Pump and Flow Controller Malfunctions

The applicant has analyzed the total loss of forced reactor coolant flow event, which bounds partial loss of forced reactor coolant flow. This event is reviewed using the review procedures and acceptance criteria in SRP Sections 15.3.1 and 15.3.2.

The loss of offsite power and resulting loss of all forced coolant flow through the reactor core causes an increase in the average coolant temperature and a decrease in the margin to departure from nucleate boiling. The reactor is tripped from an undervoltage trip monitoring the reactor coolant pump power supply, and a minimum DNBR of 1.32 is reached approximately 3 sec into the transient. The maximum calculated reactor coolant system pressure is 2,330 psia during the transient. In Revision 1 to the FSAR, the applicant stated that the most limiting single failure for this event was the loss of one of the redundant protection trains. The staff concludes that the results of the analysis meet the guidelines of SRP Sections 15.3.1 and 15.3.2 and are acceptable.

15.3.3/15.3.4 Reactor Coolant Pump Rotor Seizure and Shaft Break

The applicant's analyses of the locked reactor coolant pump rotor and a sheared reactor coolant pump shaft in FSAR Section 15.3 assume the availability of offsite power throughout the event. In accordance with SRP Sections 15.3.3 and 15.3.4 and GDC 17, the staff requires that this event be analyzed assuming turbine trip and consequential loss of offsite power to the plant auxiliaries and resulting coastdown of all undamaged pumps. Appropriate delay times may be assumed for loss of offsite power following turbine trip if suitably justified.

The event should also be analyzed assuming the worst single failure. Maximum Technical Specification primary system activity and steam generator tube leakage at the rate specified in the Technical Specifications should be assumed. The results of the analyses should demonstrate that offsite doses following the accident are less than the 10 CFR 100 guideline values.

In response to the staff's request, the applicant in a revision to the FSAR, provided the results of a reactor coolant pump rotor seizure and a shaft break accident assuming the worst single failure (i.e., loss of one protection train) and no offsite power. In the analysis, the peak cladding temperature reached is 1762°F, which is well below the cladding melting point, and the maximum reactor coolant pressure is calculated to be 2,548 psia. However, to evaluate the consequences of radiological release, the applicant stated that 6% of the fuel was assumed to experience cladding damage. As a result, the calculated offsite doses following the accident are a small fraction of the guidelines of 10 CFR 100.

In a meeting held at NRC Headquarters on May 9, 1984, the staff requested the applicant to provide the most limiting single failure with regard to radiological consequences (such as failure of steam generator power-operated relief valve (PORV) to close) assumed for this analysis. The applicant provided the results of an analysis (using LOFTRAN and FACTRAN codes) and stated that no fuel failures were predicted. Thus, the radiological consequences of the locked rotor event, assuming a failed steam generator PORV, are similar to those of a loss of nonemergency ac power and are well within the limits of 10 CFR 100.

15.4 <u>Reactivity and Power Distribution Anomalies</u>

In the following sections the staff addresses the applicant's evaluation of events that result in reactivity and power distribution anamolies.

15.4.1 Uncontrolled Rod Cluster Control Assembly (Rod) Bank Withdrawal From Zero-Power Conditions

The staff has reviewed this event according to SRP Section 15.4.1 (NUREG-0800).

The consequences of an uncontrolled rod cluster control assembly bank withdrawal at zero power have been analyzed. Such a transient can be caused by a failure of the reactor control or rod control systems. The analysis assumed a conservatively small (in absolute magnitude) negative Doppler coefficient and a conservative moderator coefficient. Further, hot zero-power initial conditions with the reactor just critical are chosen because they are known to maximize the calculated consequences. The reactivity insertion rate is assumed to be equivalent to the simultaneous withdrawal of the two highest worth banks at maximum speed (45 in./min).

Reactor trip is assumed to occur on the low setting of the power range neutron flux channel at 35% of full power (a 10% uncertainty has been added to the set point value). The maximum heat flux is much less than the full-power value, and average fuel temperature increases to a value lower than the nominal full-power value. The minimum departure from nucleate boiling ratio (DNBR) at all times remains above the limiting value of 1.30.

The possibilities for single failures of the reactor control system, which could result in uncontrolled withdrawal of control rods under low-power startup conditions, have been reviewed. The scope of the review has included investigations of initial conditions and control rod reactivity worths, the course of the resulting transients or steady-state conditions, and the instrument response to the transient or power maldistribution. The methods used to determine the peak fuel rod response and the input into the analysis, such as power distributions and reactivity feedback effects as a result of moderator and fuel temperature changes, have been examined.

The staff concludes that the requirements of GDC 10, 20, and 25 have been met.

The applicant has met the requirement of GDC 10 that the specified acceptable fuel design limits are not exceeded, of GDC 20 that the reactivity control systems are automatically initiated so that specified acceptable fuel design limits are not exceeded, and of GDC 25 that single malfunctions in the reactivity control system will not cause the specified acceptable fuel design limits to be exceeded. The applicant has met these requirements by comparing the resulting extreme operating conditions and response for the fuel (i.e., fuel duty) with the acceptance criteria for fuel damage (e.g., critical heat flux, fuel temperatures, and cladding strain limits should not be exceeded) to ensure that fuel rod failure will be precluded for this event. The bases for acceptance in the staff review are that (1) the applicant's analyses of the maximum transients for single error control rod withdrawal from a subcritical or low-power condition have been confirmed, (2) the analytical methods and input data are reasonably conservative, and (3) specified acceptable fuel design limits will not be exceeded.

15.4.2 Uncontrolled Rod Cluster Control Assembly (Rod) Bank Withdrawal at Power

The staff has reviewed this event according to SRP Section 15.4.2 (NUREG-0800).

The consequences of uncontrolled withdrawal of a rod bank in the power operating range have been analyzed. The effect of such an event is an increase in coolant temperature (resulting from the core-turbine power mismatch) that must be terminated before fuel design limits are exceeded.

The analysis is performed as a function of reactivity insertion rates, reactivity feedback coefficients, and core power level. Protection is provided by the high neutron flux trip, the overtemperature ΔT and overpower ΔT trips, and the pressurizer pressure and pressurizer water level trips. In no case does the DNBR fall below the limiting value of 1.30. Adequate fuel cooling is therefore maintained. The maximum heat flux reached, including uncertainties, does not exceed 118% of full power, thus precluding fuel centerline melting.

The possibilities for single failures of the reactor control system that could result in uncontrolled withdrawal of control rods beyond normal limits under power operation conditions have been reviewed. The scope of the review included investigations of possible initial conditions and the range of reactivity insertions, the course of the resulting transients, and the instrumentation response to the transient. The methods used to determine the peak fuel rod response and the input into the analysis, such as power distributions, rod reactivities, and reactivity feedback effects of moderator and fuel temperature changes, have been examined.

The staff concludes that the requirements of GDC 10, 20, and 25 have been met.

The applicant has met the requirements of GDC 10 that the specified acceptable fuel design limits are not exceeded, of GDC 20 that the reactivity control systems are automatically initiated so that specified acceptable fuel design limits are not exceeded, and of GDC 25 that single malfunctions in the reactivity control system will not cause the specified acceptable fuel design limits to be exceeded. The applicant has met these requirements by comparing the resulting extreme operating conditions and response for the fuel (i.e., fuel duty) with the acceptance criteria for fuel damage (e.g., critical heat fuel temperatures and cladding strain limits should not be exceeded) to ensure that fuel rod failure will be precluded for this event. The bases for acceptance in the staff review are that (1) the applicant's analysis of maximum transients for single error control rod malfunctions have been confirmed, (2) the analytical methods and input data are reasonably conservative, and (3) specified acceptable fuel design limits will not be exceeded.

15.4.3 Rod Cluster Control Assembly Malfunctions

The staff has reviewed this event according to SRP Section 15.4.3 (NUREG-0800).

Rod cluster control assembly misalignment incidents, including a dropped full-length assembly, a dropped full-length bank, a misaligned full-length assembly, and the withdrawal of a single assembly while operating at power, have been analyzed by the applicant. Misaligned rods are detectable by (1) asymmetric power distributions sensed by excore nuclear instrumentation or core exit thermocouples, (2) rod deviation alarm, and (3) rod position indicators. A deviation of a rod from its bank by about 15 in. or twice the revolution of the rod position indicator will not cause power distribution to exceed design limits. Additional surveillance will be required to ensure rod alignment if one or more rod position channels are out of service.

In the event of a dropped assembly or group of assemblies, the reactor will typically scram on a neutron flux negative rate trip, and analysis indicates that thermal limits will not be exceeded for the event. However, if the rod locations are such that the reactor does not scram, the automatic controller may return the reactor to full power and the control could result in a power overshoot. An analysis methodology for this event has been developed by Westinghouse and reported in WCAP-10297-P. This methodology has been reviewed and approved by the NRC staff. The review is discussed in a memorandum to

F. Miraglia from L. Rubenstein dated December 1983. Generally, detailed analyses for most reactors for most cycles show that if this event occurs, thermal limits will not be exceeded. However, the analyses are reactor and cycle specific, and the analysis for Millstone Unit 3 for cycle 1 has not been completed. The staff also has accepted an interim position for operating reactors that consists of a restriction on operations above 90% power so that either the reactor is in manual control or rods are required to be out more than 215 steps. This restriction will be applied to Millstone Unit 3 in the event that calculations for cycle 1 operation are not completed in time for initial operations. With this restriction, thermal limits will not be exceeded. Approval of the analysis specific to Millstone Unit 3 for cycle 1 will result in removing the restriction. A similar analysis also will be needed for each subsequent reload cycle.

For cases where a group of assemblies is inserted to its insertion limit with a single rod in the group stuck in the fully withdrawn position, analysis indicates that departure from nucleate boiling (DNB) will not occur. The staff has reviewed the calculated estimates of the expected reactivity and power distribution changes that accompany postulated misalignments of representative assemblies. The staff has concluded that the values used in this analysis conservatively bound the expected values, including calculational uncertainties.

The inadvertent withdrawal of a single assembly would be the result of multiple failures in the control system or multiple operator errors or deliberate operator actions combined with a single failure of the control system. As a result, the single assembly withdrawal is classified as an infrequent occurrence. The resulting transient is similar to that resulting from a bank withdrawal, but the increased peaking factor may cause DNB to occur in the region surrounding the withdrawal assembly. Less than 5% of the rods in the core experience DNB for such a transient.

The possibilities for single failures of the reactor control system that could result in a movement or malposition of control rods beyond normal limits have been reviewed. The scope of the review has included investigations of possible rod malposition configurations, the course of the resulting transients or steady-state conditions, and the instrumentation response to the transient or power maldistribution. The methods used to determine the peak fuel rod response and the input to that analysis, such as power distribution changes, rod reactivities, and reactivity feedback effects as a result of moderator and fuel temperature changes, have been examined.

The staff concludes that the requirements of GDC 10, 20, and 25 have been met.

The applicant has met the requirements of GDC 10 that the specified acceptable fuel design limits are not exceeded, of GDC 20 that the reactivity control systems are automatically initiated so that specified acceptable fuel design limits are not exceeded, and of GDC 25 that single malfunctions in the reactivity control system will not cause the specified acceptable fuel design limits to be exceeded. The applicant has met these requirements by comparing the resulting extreme operating conditions and response for the fuel (i.e., fuel duty) with the acceptance criteria for fuel damage (e.g., critical heat flux, fuel temperatures, and cladding strain limits should not be exceeded) to ensure that fuel rod failure will be precluded for this event. The bases for acceptance in the staff review are that (1) maximum configurations and transients for single error control rod malfunctions have been analyzed, (2) the analysis methods and input data are resonably conservative, and (3) specified acceptable fuel design limits will not be exceeded.

15.4.4/15.4.5 Startup of a Reactor Coolant Pump at an Incorrect Temperature

In FSAR Section 15.4.4, the applicant provided the results of an analysis for startup of an inactive reactor coolant pump event. This event is reviewed using the review procedures and acceptance criteria in SRP Section 15.4.4.

During the first part of the transient, the increase in core flow with cold water results in an increase in nuclear power and a decrease in core average temperature. Reactivity addition for the inactive loop startup event is due to the decrease in core inlet water temperature. This transient was evaluated by the applicant using a mathematical model that has been reviewed and found acceptable to the staff. The maximum calculated reactor coolant system pressure is 2,350 psia, and the minimum DNBR is above 1.93 during the transient.

In Revision 1 to the FSAR, the applicant stated that the most limiting single failure for this event was the loss of one of the redundant protection trains. The staff concludes that the results of the analysis meet the criteria in SRP Section 15.4.4 and are acceptable.

15.4.6 Inadvertent Boron Dilution

SRP Section 15.4.6 requires that at least 15 min should be available from the time the operator is made aware of an unplanned boron dilution event to the time a loss of shutdown margin occurs during power operation, startup, hot standby, hot shutdown, and cold shutdown. A warning time of 30 min is required during refueling. The staff has requested that control room alarms be available to alert the operator to boron dilution events in all modes of operation. If a second alarm is not provided, the applicant must show that the consequences of the most limiting unmitigated boron dilution event meet the staff criteria and are acceptable. The applicant should provide analyses in accordance with the guidelines of SRP Section 15.4.6 for each of the six operational modes. The analyses should confirm that time intervals meet the SRP criteria. Also, Technical Specifications should be established that are consistent with the analyses assumptions.

In response to the staff's request, the applicant submitted a report in January 1984 entitled "Millstone Unit 3 Boron Dilution Analysis Failure in the Chemical and Volume Control System." The applicant performed a failure mode analysis for the chemical and volume control system using the criteria in SRP Section 15.4.6. The calculated minimum time intervals available (time interval between an alarm announcing an unplanned modulator dilution and the time of loss of shutdown margin) for the operator to take appropriate corrective actions are

(1) 33.7 min for refueling operation
(2) 15.2 min for hot standby and hot shutdown modes of operation
(3) 15.8 min for cold shutdown operation

Instruments and alarms, such as source range high flux, power range high flux, overtemperature ΔT, are provided to alert the operator to a boron dilution event. The primary instrument used to detect this event is the redundant safety-grade source range. The alarms are also safety grade and redundant.

In addition, a boronometer and a flow differential alarm, which indicates improper proportioning of reactor makeup water and boric acid solution, warn the operator of a potential boron dilution event. The volume control tank high level alarm also provides an indication of a potential boron dilution event to the operator. The staff finds the response acceptable.

15.4.7 Inadvertent Loading of a Fuel Assembly Into Improper Position

The staff has reviewed this event according to SRP Section 15.4.7 (NUREG-0800).

Strict administrative controls in the form of previously approved established procedures and startup testing are followed during fuel loadings to prevent operation with a fuel assembly in an improper location or a misloaded burnable poison assembly. Nevertheless, an analysis of the consequences of a loading error has been performed.

Comparisons of power distributions calculated for the nominal fuel loading pattern and those calculated for five loadings with misplaced fuel assemblies or burnable poison assemblies are presented by the applicant. The selected loadings that are not normal represent the spectrum of potential inadvertent fuel misplacement. Calculations included, in particular, the power in assemblies that contain provisions for monitoring with in-core detectors.

As part of the required startup testing, the in-core detector system is used to detect misloaded fuel before operating at power. The analysis described above shows that all but one of the above misloading events would be detected by this test. In the excepted case, an interchange of region 1 and 2 assemblies near the center of the core, the increase in the power peaking is approximately equal to the uncertainty in the measurement of this quantity (5%). This uncertainty is allowed for in analyses so that this misloading event does not result in unacceptable consequences.

The staff has evaluated the consequences of a spectrum of postulated fuel loading errors. The staff concludes that the analyses provided by the applicant have shown for each case considered that either the error is detectable by the available instrumentation (and hence remediable) or the error is undetectable, but the offsite consequences of any fuel rod failures are a small fraction of 10 CFR 100 guidelines. The applicant affirms that the available in-core instrumentation will be used before the start of a fuel cycle to search for fuel loading errors.

The staff concludes that the requirements of GDC 13 and 10 CFR 100 have been met.

The applicant has met the requirements of GDC 13 with respect to providing adequate provisions to minimize the potential of a misloaded fuel assembly going undetected and of 10 CFR 100 with respect to mitigating the consequences of reactor operations with a misloaded fuel assembly. The applicant has met these requirements by providing acceptable procedures and design features that will minimize the likelihood of loading fuel in a location other than its designated place.

15.4.8 Rupture of a Control Rod Drive Mechanism Housing (Rod Cluster Control Assembly Ejection)

The staff has reviewed this event according to SRP Section 15.4.8 (NUREG-0800).

The mechanical failure of a control rod mechanism pressure housing would result in the ejection of a rod cluster control assembly. For assemblies initially inserted, the consequences would be a rapid reactivity insertion together with an adverse core power distribution, possibly leading to localized fuel rod damage. Although mechanical provisions have been made to make this accident extremely unlikely, the applicant has analyzed the consequences of such an event.

Methods used in the analysis are reported in WCAP-7588, Revision 2, which has been reviewed and accepted by the staff. This report demonstrated that the model used in the accident analysis is conservative relative to a three-dimensional kinetics calculation.

The applicant's criteria for gross damage of fuel are a maximum cladding temperature of 2700°F and an energy deposition of 200 calories per gram in the hottest pellet. These criteria are more conservative* than those proposed in RG 1.77 and are, therefore, acceptable.

Four cases were analyzed: beginning-of-cycle at 102% and zero power and end-of-cycle at 102% and zero power. The highest cladding temperature, 2430°F, and the highest fuel enthalpy, 158 calories per gram, were shown to occur in the end-of-cycle zero-power and full-power cases, respectively. The analysis also shows that less than 10% of the fuel experiences DNB and less than 10% of the hot pellet melts. Analyses have been performed to show that the pressure surge produced by the rod ejection is mild and will not approach the reactor coolant system emergency limits. Further analyses have shown that a cascade effect, that is, the ejection of a further rod resulting from the ejection of the first one, is not credible.

The staff concludes that the analysis of the rod ejection accident is acceptable and meets the requirements of GDC 28.

The applicant met the requirements of GDC 28 with respect to preventing postulated reactivity accidents that could result in damage to the reactor coolant pressure boundary greater than limited local yielding or could cause sufficient damage that would signficantly impair the capability to cool the core. The applicant met the requirements by demonstrating compliance with the positions of RG 1.77. The staff has evaluated the applicant's analysis of the assumed control rod ejection accident and finds the assumptions, calculation techniques, and consequences acceptable. Because the calculations resulted in peak fuel enthalpies less than 280 calories per gram, prompt fuel rupture with consequent rapid heat transfer to the coolant from finely dispersed molten UO_2 was assumed not to occur. The pressure surge was, therefore, calculated on the basis of conventional heat transfer from the fuel and resulted in a pressure increase

*RG 1.77 has an acceptance criterion of 280 calories per gram energy deposition and no criterion for cladding temperature other than that implicit in requirements for fuel and pressure vessel damage.

below service limit C (as defined in Section III, "Nuclear Power Plant Components," of the ASME Code) for the maximum control rod worths assumed. The staff believes that the calculations contain sufficient conservatism, both in the initial assumptions and in the analytical models, to ensure that primary system integrity will be maintained.

15.4.8.1 Control Rod Ejection Accident

A nonmechanistic rupture of a control rod drive housing was postulated. Because of the resultant opening in the pressure vessel, primary coolant would be lost to the containment with concurrent rapid depressurization of the reactor pressure vessel. Reactor trip, initiated by one of several trip signals, would occur rapidly.

Ejection of a control rod results in rapid reactivity insertion. The applicant has calculated and conservatively assumed that 10% of the fuel elements will experience cladding failure, releasing all their gap radioactivity. In addition, the applicant has conservatively calculated that 0.25% of the fuel rods will experience fuel melting. The released radioactivity is mixed immediately with the primary coolant. The staff assumed that release to the environment may occur by either of two pathways. The first pathway involves a release of activity to the primary containment, which is then assumed to leak to the atmosphere as in the design-basis LOCA. In the second pathway, activity is transferred from the primary to the secondary coolant at an assumed 1-gpm primary-to-secondary leak rate. With loss of offsite power and subsequent steam venting, some of the iodine transferred to the shell side can leak to the environment.

In considering the consequences of this postulated event, the staff calculated the doses as if all the activity were totally released by way of each of the above pathways. The staff would expect the actual consequences to be some combination from these pathways. The assumptions used in calculating the radiological consequences are presented in Table 15.1, and the resultant doses for each pathway are given in Table 15.2 of this SER.

The staff has reviewed the applicant's analysis for the radiological consequences following a postulated control rod ejection accident. The staff concludes that the distances to the exclusion area boundary (EAB) and the low population (LPZ) boundary for Millstone Unit 3, in conjunction with the operation of the dose-mitigating engineered safety features (ESF) systems, are sufficient to provide reasonable assurance that the calculated radiological consequences would be well within the exposure guidelines (less than 25%) in 10 CFR 100.11.

The staff's conclusion is based on (1) the staff's review of the applicant's analysis of the radiological consequences, (2) the staff's independent dose calculation using the recommendations of Appendix B of RG 1.77 and the atmospheric dispersion factors discussed in Section 2 of this report, and (3) the Westinghouse Standard Technical Specifications (NUREG-0452, Rev. 4) for the primary-to-secondary leakage in the steam generators.

15.4.9 Spectrum of Rod Drop Accidents (BWR)

This section is not applicable to Millstone Unit 3.

15.5 Increase in Reactor Coolant System Inventory

The applicant's analysis of events that result in an increase in the primary system inventory are discussed below.

15.5.1 Inadvertent Actuation of the Emergency Core Coolant System During Power Operation

See Section 15.5.2.

15.5.2 Chemical and Volume Control System Malfunction That Increases Reactor Coolant Inventory

Emergency core cooling system operation could be initiated by a spurious signal or operator error. Two cases were examined, one in which the reactor trip occurs simultaneously as a result of the safety injection signal, the other in the which the reactor trip occurs later in the transient on reactor coolant system low pressure. The reactor pressure decreases during the initial phase of the transient and reaches a minimum pressure of 1,850 psia at 100 sec into the transient, then recovers slightly to approximately 2,000 psia. The DNBR never drops below its initial value for this transient.

In Revision 1 to the FSAR, the applicant stated that the most limiting single failure for this event was the loss of one of the redundant protection trains. The staff finds the results acceptable.

15.6 Decreases in Reactor Coolant Inventory

The postulated design-basis accidents analyzed by the applicant to determine their offsite radiological consequences are the same as those analyzed for previously licensed PWRs. To evaluate the effectiveness of the ESFs proposed for Millstone Unit 3 and to ensure that the radiological consequences of these accidents meet the applicable dose criteria, the staff has analyzed the LOCA (Section 15.6.5), the fuel-handling (Section 15.7.4), the steamline break (Section 15.6.4), the steam generator tube rupture (SGTR) (Section 15.6.3), the small-line break (Section 15.6.2), and the control rod ejection accidents (Section 15.4.8). Because it is the staff's position that the Standard Technical Specifications be used, the staff has evaluated the radiological consequences of such accidents having releases through the secondary system, assuming the values given in the Westinghouse Standard Technical Specifications (NUREG-0452, Rev. 4) for primary and secondary coolant activity concentrations and primary-to-secondary leakage. The staff will review the proposed Millstone Unit 3 Technical Specifications to ensure that these limits are met. The calculated doses for these accidents are given in Table 15.2.

The applicant has analyzed the following events that result in a decrease in reactor coolant inventory.

15.6.1 Inadvertent Opening of a Pressurizer Safety or Relief Valve

In FSAR Section 15.6.1, the applicant provided the results of an analysis for inadvertent opening of a pressurizer safety or relief valve. During this event, nuclear power remains at the initial value until reactor trip occurs on low pressurizer pressure. The DNBR decreases initially as a result of the reduction in reactor coolant system (RCS) pressure, but increases rapidly following the trip. The minimum DNBR of 1.5 occurs 24 sec into the transient. The RCS pressure decreases throughout the transient.

In Revision 1 to the FSAR, the applicant stated that the most limiting single failure for this event was the loss of one of the redundant protection trains. The staff finds the results acceptable.

15.6.2 Failure of a Small Line Carrying Primary Coolant Outside Containment

The applicant has provided an analysis of an accidental break in the chemical and volume control system (CVCS) letdown line outside containment, but downstream of the containment isolation valves. The applicant has postulated that the most severe pipe rupture with regard to radiological consequences outside containment would be a complete severance of a 3-in. letdown line in the CVCS. The staff concurs in this assumption. This break would release up to 140 gpm of primary coolant to the auxiliary building, providing a release pathway to the environment. The break would cause a low level in the volume control tank, and the operator could diagnose the break and shut the appropriate isolation valve to isolate the leak. The staff has performed an independent assessment of the radiological consequences of this accident.

The staff assumed that 30 min is required for receipt of the low-level signal and operator action to isolate the break. Thus, a total of 4,200 gal of primary coolant could be released. The staff estimated that 39% of the hot reactor coolant would flash into steam on entering the auxiliary building atmosphere and assumed that an equal fraction of the dissolved iodine fission products would become airborne as gas and particulates. In the absence of ESFs designed to detect and mitigate the consequences of such releases, the staff assumed that the airborne iodine could escape directly to the environment at ground level without delay or effective filtration. Other assumptions are given in Table 15.3.

The staff concludes that the distances to the EAB and the LPZ boundary for Millstone Unit 3 are sufficient to provide reasonable assurance that the calculated radiological consequences of a postulated small-line failure outside the containment (assuming the primary coolant equilibrium iodine concentration permitted by the Standard Technical Specifications) in combination with an accident-generated iodine spike, do not exceed a small fraction of the exposure guidelines in 10 CFR 100.11. The results of the staff's calculations are given in Table 15.2.

The staff's conclusion is based on (1) the staff's review of the applicant's classification and identification of small lines in accordance with GDC 55 and RG 1.11, (2) the staff's review of the applicant's analysis of radiological consequences, (3) the independent dose calculation by the staff using Position C.1.b of RG 1.11 and conservative atmospheric dispersion factors discussed in Section 2 of this report, and (4) the Westinghouse Standard Technical Specifications for the equilibrium iodine concentrations in the primary coolant system. The staff will review the Millstone Unit 3 Technical Specifications to ensure that primary coolant operating conditions that could result in a small-line-break accident with radiological consequences that exceed the dose guidelines for such events are not allowed to occur.

15.6.3 Steam Generator Tube Rupture

A steam generator tube rupture (SGTR) accident releases primary coolant to the secondary side of the steam generator, thus providing a pathway for radiological releases to the environment. This section contains the evaluation of the aspects of the SGTR analysis that pertain to the system.

The accident examined in the FSAR involves a complete severance of a single steam generator tube. The applicant's description of the accident, including the sequence of events, bases for operator action, and the effects of loss of offsite power, was reviewed. The accident scenario involves reactor and turbine trip and subsequent safety injection (SI) actuation initiated by low pressurizer pressure. Emergency feedwater system startup, initiated by SI, was also examined. If offsite power is not available, the turbine bypass valves would close and the steam would discharge to the atmosphere via the steam generator atmospheric relief and/or safety valves.

The applicant states that the operator is expected to determine that an SGTR has occurred and to identify and isolate the faulty steam generator in time to minimize contamination of the secondary system and ensure termination of radioactive release to the atmosphere from the faulty unit. The applicant further states that "consideration of the indications provided on the control board together with the magnitude of the break flow, leads to the conclusion that the accident diagnostics and isolation procedure can be completed within 30 minutes of initiation for the design basis events." SGTR events at operating reactors generally indicate that a longer time period than 30 min is needed for event identification and pressure equalization (e.g., 3 hours at Ginna). The applicant subsequently provided additional information in response to the staff's questions on this subject. The staff has reviewed the applicant's response and concludes that the applicant's analysis is incomplete with regard to demonstrating that the acceptance criteria in SRP Section 15.6.3 are fully met and that there are discrepancies in the applicant's submittal. The staff requires the following additional information to fully evaluate this analysis:

(1) The applicant should submit an evaluation of operator actions necessary to effect pressure equalization, and a conservative, justifiable time estimate for each action, as well as initial delay time. Relevant simulator experience should be cited as part of the justification.

(2) FSAR Section 15.6.3 indicates equalization of primary and secondary pressure 30 min after the SGTR event, with consequent termination of steam generator tube leakage. However, FSAR Figures 15.6-3A and 15.6-3C indicate a pressure differential of 950 psi at 1,800 sec. The applicant should explain this discrepancy and modify the analysis of this event accordingly, using the evaluation of operator actions discussed above.

(3) The applicant is requested to discuss (a) whether, as a result of possible modification of the analysis, including consideration of longer leak times, liquid can enter the main steamlines, and (b) what would the effects be on the integrity of the steam piping and supports, considering both the liquid deadweight and the possibility of waterhammer. Unless the applicant can demonstrate that the incident will be terminated within a time period sufficiently short to avoid steam generator overfill, the applicant

should submit the results of an analysis that demonstrates that the integrity of the steamlines and supports will be maintained.

(4) The applicant should verify that all primary components that are credited in the analysis to mitigate the consequences of the SGTR, including the component power and motive sources, are classified as safety related, meet applicable GDC (including GDC 1, 2, and 4), are seismically and environmentally qualified, and have sufficient capability to equalize primary and secondary pressure within the time period postulated in the response to Items 1 and 3 above. The applicant should ensure that the plant Technical Specifications for these components accurately reflect their assumed availability and operability in the safety analysis (e.g., if the PORV is assumed available to provide the depressurization function, then the Technical Specifications must be written so it cannot be removed from service).

In response to the staff's request, the applicant has committed to comply with the conclusion and recomendations of the generic study currently being conducted by Westinghouse Electric Company for the owner's group. Therefore, the radiological consequences of an SGTR accident will be addressed in a future supplement along with progress on the items above.

15.6.4 Main Steamline Break Outside Containment

Both the staff and the applicant have evaluated the radiological consequences of a postulated steamline-break accident occurring outside containment and upstream of the main steam isolation valve. Although the contents of the secondary side of the affected steam generator would be vented initially to the atmosphere as an elevated release, the staff has conservatively assumed that the entire release throughout the course of the accident occurs at ground level conditions. During the course of the accident, the shell side of the affected steam generator was assumed to stay dry because the auxiliary feedwater flow to the affected steam generator would be blocked off under the conditions of this accident. As a result of the assumed dryout condition in the affected steam generator, all iodine transported to the secondary side by leakage (at the Technical Specification limit of 1 gpm) was assumed available for release to the atmosphere with no reduction for holdup or attenuation.

As part of its review, the staff investigated three cases. For case 1, the most reactive control rod is assumed to be stuck in the fully withdrawn position. The applicant has indicated, and the staff agrees, that no departure from nucleate boiling is expected to occur and, therefore, no fuel cladding failure is to be assumed in the calculation. With no fuel failures, the radiological consequences for case 1 are identical to those for case 2.

For case 2, the staff assumed that an iodine spike occurred as a result of the power and pressure transient caused by the accident. Before the accident, Millstone Unit 3 was assumed to be operating at the equilibrium primary coolant limit of 1 μCi/gm dose equivalent iodine-131 (DEI-131) specified in the Westinghouse Standard Technical Specifications. The iodine spike generated during the accident is assumed to increase because the iodine release rate from the fuel increases by a factor of 500. This increase in the release rate results in an increasing iodine concentration in the primary coolant during the course of the accident. The radiological consequences for this case have been calculated using assumptions given in Table 15.4, and the consequence values are given in Table 15.2 of this SER.

For case 3, the staff assumed that previous reactor operation had resulted in a primary coolant concentration equal to the maximum transient full-power Westinghouse Standard Technical Specification limit (60 μCi/gm DEI-131). As in case 2, the radiological consequences were calculated using assumptions found in Table 15.4, and the consequence values are given in Table 15.2.

On the basis of its findings, the staff concludes that the distances to the EAB and the LPZ boundary for Millstone Unit 3 are sufficient to provide reasonable assurance that the calculated radiological consequences of a postulated main steamline failure outside the containment of Millstone Unit 3 do not exceed (1) the exposure guidelines in 10 CFR 100.11 for the case where the failure occurs with a primary coolant iodine concentration corresponding to a preaccident iodine spike and (2) 10% of these exposure guidelines for the case where the failure occurs with a primary coolant activity corresponding to the maximum equilibrium concentration for continued full-power operation as stated in the Westinghouse Standard Technical Specifications (NUREG-0452, Rev. 4). The staff concludes that the proposed design and operation of Millstone Unit 3 will be effective in controlling the release of fission products following a postulated main steamline-break accident.

15.6.5 Loss-of-Coolant Accident

The applicant has selected and analyzed a hypothetical design-basis LOCA and has shown that the distances to the EAB and the LPZ boundary are sufficient to provide reasonable assurance that the radiological consequences of such an accident are within the guidelines in 10 CFR 100.11 (a)(1) and (2). The analysis has included the following sources and radioactivity transport paths to the atmosphere:

(1) contribution from containment leakage
(2) contribution from post-LOCA leakage from ESF systems outside containment

The staff's review confirms the applicant's finding on the basis of the following:

(1) the applicant's provisions for and design of the containment system, the containment spray system, and the containment enclosure building with its associated supplemental leak collection and release system (SLCRS) and the acceptability of these systems and structures as described in Sections 3 and 6 of this report

(2) the staff's independent analysis of the radiological consequences of a hypothetical design-basis LOCA as described below

The applicant has analyzed the double-ended cold-leg guillotine (DECLG) break as the most limiting large-break LOCA. The analysis is done using three different flow coefficients. The results show that the DECLG with a Moody break discharge coefficient of 0.6 is the worst case. In this analysis, the peak cladding temperature reached is 1960°F. For the small-break LOCA the applicant has determined that a cold-leg rupture less than 10 in. in diameter is the most limiting. The analysis was performed for a 3-in.-, 4-in.-, and 6-in.-diameter break. The results show that the 4-in.-diameter break is the worst case and results in a peak cladding temperature of 1495°F. Both of these accidents are terminated by emergency core cooling system operation. Only safety-grade equipment is used to mitigate the accident.

The staff has reviewed the LOCA analysis according to SRP Section 15.6.5 and concluded that the loss-of-coolant analysis, based on a spectrum of postulated piping breaks within the reactor coolant pressure boundary, is acceptable and meets the relevant requirements of 10 CFR 50.46, Appendix K to 10 CFR 50, GDC 35, and 10 CFR 100. This conclusion is based on the following:

The applicant has performed analyses of the performance of the emergency core cooling system (ECCS) in accordance with the Commission's regulations (10 CFR 50.46 and Appendix K to 10 CFR 50). The analyses considered a spectrum of postulated break sizes and locations and were performed with an evaluation model that had been previously reviewed and approved by the staff and described in NUREG-0390, NUREG-0011 (Sequoyah SER), and NUREG-0422 (McGuire SER). The results of the analyses show that the ECCS satisfies the following criteria:

(1) The calculated maximum fuel rod cladding temperature does not exceed 2200°F.

(2) The calculated maximum local oxidation of the cladding does not exceed 17% of the total cladding thickness before oxidation.

(3) The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 1% of the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

(4) Calculated changes in core geometry are such that the core remains amenable to cooling.

(5) After any calculated successful initial operation of the ECCS, the calculated core temperature is maintained at an acceptably low value and decay heat is removed for the extended period of time required by the long-lived radioactivity.

(6) The applicant has met the requirements of TMI Action Plan Items II.E.2.3, II.K.3.5, II.K.3.25, II.K.3.30, and II.K.3.31.

15.6.5.1 Containment Leakage Contribution

Millstone Unit 3 includes a subatmospheric containment designed to minimize the leakage of fission products from a postulated design-basis LOCA. The containment consists of a posttensioned concrete primary containment with a carbon steel liner and a containment spray system designed to bring the containment atmosphere into subatmospheric condition within 60 min after the onset of a LOCA. Another ESF is the containment enclosure building and its associated SLCRS, which enhances the removal of iodine leaked from the containment to the enclosure building following a LOCA.

The staff used the conservative assumptions of Positions C.1.a through C.1.e of RG 1.4, Revision 2, to calculate the consequences of the hypothetical LOCA. The primary containment was assumed to leak at a rate of 0.9%/day for the first hour, and because the containment pressure becomes subatmospheric at the end of 60 min, the leak rate was 0%/day after 1 hour. The fraction of core inventory available for release was assumed to be 25% for iodine and 100% for noble gases. The analysis took into account radiological decay during holdup in the containment and iodine decontamination by the SLCRS. A list of assumptions used in the calculation of the LOCA doses is given in Table 15.5.

15.6.5.2 Post-LOCA Leakage From ESF System Outside Containment

As part of the LOCA analysis, the staff has also evaluated the consequences of leakage of containment sump water, which is to be circulated by the ESF systems after the posulated accident. During the recirculation mode of operation, the sump water is circulated outside containment to the auxiliary building. If a leak should develop, such as that from a pump seal failure, a fraction of the iodine in the water could become airborne in the auxiliary building and be released to the atmosphere. For Millstone Unit 3, the area in the auxiliary building containing the ESF systems which recirculate contaminated fluids is served by an ESF air filtration system (the auxiliary building exhaust system); therefore, doses from passive failures were not considered (as specified in SRP Section 15.6.5, Appendix B).

In FSAR Table 15.6-9, the applicant has identified a value of 5,000 cc/hr as the routine amount of leakage from ECCS equipment following an accident. Using the guidance of Appendix B of SRP Section 15.6.5, the staff evaluated the potential radiological consequences from this release pathway assuming a routine leakage rate of twice the applicant's value (10,000 cc/hr). The resultant estimated radiological consequences for this pathway were less than 0.1 rem to the thyroid at both the EAB and LPZ boundary. The staff also evaluated the potential radiological consequences from normal ECCS leakage at a leak rate of 1 gpm. The resultant estimated radiological consequences were 2.2 rems to the thyroid at the EAB and 1.5 rems to the thyroid at the LPZ boundary.

15.6.5.3 Conclusions

The staff's calculated thyroid and whole-body doses from the hypothetical LOCA are given in Table 15.2. The staff concludes that the distances to the EAB and the LPZ boundary for Millstone Unit 3, in conjunction with the ESFs of the Millstone Unit 3 design, are sufficient to provide reasonable assurance that the total radiological consequences of a postulated LOCA will be within the exposure guidelines in 10 CFR 100.11. This conclusion is based on the staff's review of the applicant's analyses and on an independent analysis performed by the staff to verify that the total calculated doses are within the guidelines.

15.7 Radioactive Releases From a Subsystem or Component

15.7.1*

15.7.2*

*The Standard Review Plan (NUREG-0800) does not include sections addressing FSAR sections that consist of background or design data used in the review of other sections. The section numbers are retained in this SER to provide continuity and ensure a close correlation between subsequent SER sections and their associated SRP sections.

15.7.3 Postulated Radioactive Releases Resulting From Liquid Tank Failures

The applicant's analysis of the radioactive liquid waste tank failure accident is provided in FSAR Section 15.7.3. The staff has reviewed the applicant's analysis and has performed an independent evaluation of this accident in accordance with SRP Section 15.7.3 (NUREG-0800).

The principal criteria governing acceptance in the review are (1) GDC 60, as it relates to the radioactive waste management systems being designed to control releases of radioactive materials to the environment and (2) 10 CFR 20, as it relates to effluents released to unrestricted areas. Tanks and associated components containing radioactive liquids outside containment are considered acceptable, according to the criteria of SRP Section 15.7.3, if failure does not result in radionuclide concentrations in excess of the limits in 10 CFR 20, Appendix B, Table II, Column 2, at the nearest potable water supply in an unrestricted area.

The applicant's analysis postulated the spill of 120,000 gal of radioactive material as a result of the failure of a boron recovery tank. The accidental releases were conservatively assumed to spill from a boron recovery tank at the radwaste enclosure, enter instantaneously to the groundwater as a slug release, and flow toward the Niantic Bay.

The staff also assumed that the contents of the same tank did enter the groundwater system instantaneously. The nuclides were then assumed to travel by way of the backfilled trench containing the circulating and service water pipelines to the bay with the travel time controlled by the groundwater velocity.

The groundwater gradient for this part of the Millstone Unit 3 site is toward the adjacent saltwater of Niantic Bay and away from areas of groundwater usage. The staff has conservatively estimated the travel time from the site to the waters of the bay to be approximately 6.6 years. For those nuclides that are affected by ion-exchange processes, the travel time would be longer. Using conservative values of groundwater travel time and dilution by the mixing process associated with the tidal-, wind-, and wave-induced currents of Long Island Sound, the staff concludes that the concentration of all radionuclides present at the tidal water will be sufficiently below the 10 CFR 20 limits.

Therefore, the staff concludes that accidental releases of liquid radioactivity from accidents within the design basis would not pose a threat to public health and safety and that the plant meets the requirements of 10 CFR 20 and 100.

15.7.4 Fuel-Handling Accident

For the analysis of the fuel-handling accident in the fuel pool, the staff has assumed that a fuel assembly was dropped in the fuel pool during refueling operations and that all of the fuel rods in the dropped assembly were damaged, plus 50 rods in the second assembly (as proposed by the applicant), thereby releasing the volatile fission gases from the fuel rod gaps into the pool. The fuel building ESF-grade ventilation system and filters would be in operation during fuel handling. Further, the radioactive materials that escaped from the fuel pool were assumed to be released to the environment as a puff release with the iodine activity reduced by filtration through the fuel building ventilation system. The radiological consequences following the postulated accident are given in Table 15.2, and the assumptions and parameters used in the analysis are given in Table 15.6. The dose model and dose conversion factors used in the analysis were the same as those in RG 1.25, "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."

The staff also evaluated the consequences of a fuel-handling accident inside containment. The applicant states that the purge exhaust system is equipped with two isolation valves in series. These valves are fast-acting isolation valves designed to close within 1.5 sec after the receipt of a high radiation signal from the radiation monitors. The applicant has estimated that the isolation valves would be closed within 2.5 sec following a fuel-handling accident and that at least 5 sec would be required for radioactivity to reach the purge system under normal conditions. Therefore, the design capability for rapid isolation of the containment provides assurance that virtually all the radioactive releases would be contained in the primary containment, and no doses are reported in this SER.

The staff finds that the applicant has provided an adequate system to mitigate the radiological consequences of a postulated fuel-handling accident inside the containment and in the spent fuel pool area. The staff concludes that the fuel-handling area ventilation system meets the relevant requirements of GDC 61. The staff further concludes that the distances to the EAB and the LPZ boundary for Millstone Unit 3, in conjunction with the operation of dose-mitigating ESFs and implementation of plant procedures, are sufficient to provide reasonable assurance that the calculated offsite radiological consequences of a postulated fuel-handling accident are well within the 10 CFR 100 exposure guidelines.

The staff's conclusion is based on (1) the staff's determination that the design features and plant procedures at Millstone Unit 3 meet the requirements of GDC 61 with respect to radioactivity control; (2) the staff's review of the applicant's assumptions and analyses of the radiological consequences of the fuel-handling accident; (3) the staff's independent analyses using the assumptions in RG 1.25, Positions C.1.a through C.1.k; and (4) the Millstone Unit 3 Technical Specifications relating to fuel-handling and ventilation system operation.

15.7.5 Spent Fuel Cask Drop Accident

The applicant has stated that the spent fuel cask will not be lifted more than 30 ft above any surface during the entire transfer operation under normal operating conditions. On the basis of this commitment, no radiological release is anticipated from such a drop, and, therefore, no doses need to be evaluated in accordance with SRP Section 15.7.5.

15.8 Anticipated Transients Without Scram

Anticipated transients without scram (ATWS) are events in which the scram system (reactor trip system) is postulated to fail to operate as required. This subject has been under generic review by the Commission staff for several years.

Volume 3 of NUREG-0460, "Anticipated Transients Without Scram for Light Water Reactors," was issued in December 1978. It describes the proposed plant modifications the staff believed are necessary to reduce the risk from anticipated transients with failure to scram to an acceptable level. The staff issued requests for the industry to supply generic analyses to confirm the ATWS mitigation capability described in Volume 3 of NUREG-0460. The staff subsequently presented recommendations on plant modifications to the Commission in September 1980. The staff has recommended to the Commission that rulemaking be used to determine the required modifications to resolve ATWS concerns as well as the required schedule for implementation of such modifications. Millstone Unit 3 is subject to the Commission's decision in this matter.

The bases for operation of Millstone Unit 3 at full power while final resolution of ATWS is being considered by the Commission are discussed below.

NUREG-0460, Volume 3, states:

> The staff has maintained since 1973 (for example, see pages 69 and 70 of WASH-1270) and reaffirms today that the present likelihood of severe consequences arising from an ATWS event is acceptably small and presently there is no undue risk to the public from ATWS. This conclusion is based on engineering judgement in view of: (a) the estimated arrival rate of anticipated transients with potentially severe consequences in the event of a scram failure; (b) the favorable operating experience with current scram systems; and (c) the limited number of operating reactors.

In view of these considerations and its expectation that the necessary plant modifications will be implemented in 1 to 4 years following a Commission decision on ATWS, the staff has generally concluded that PWR plants can continue to operate because the risk from ATWS events during this period is acceptably small. As a prudent course, to further reduce the risk from ATWS events during the interim period before the plant modifications determined by the Commission to be necessary are completed, the staff has required that emergency procedures be developed to assist operators in the recognition and mitigation of an ATWS event. These procedures shall include consideration of scram indicators, rod position indicators, flux monitors, pressurizer level and pressure indicators, pressurizer relief valve and safety valve indicators, and any other alarms annunciated in the control room, with emphasis on alarms not processed through the electric portion of the reactor scram system. When implemented, these procedures will provide an acceptable basis for interim operation of Millstone Unit 3 based on the staff's understanding of the plant's response to postulated ATWS events.

As noted in Section 13.5.2, the applicant has committed to implement the Westinghouse Emergency Response Guidelines (Westinghouse, Nov. 30, 1981; July 21, 1982, Jan. 4, 1983) that have been endorsed by the staff. These guidelines include instructions for responding to ATWS. On the basis of the applicant's commitment to implement procedures based on NRC-approved guidelines, the staff concludes that the applicant's commitment in this area is acceptable on an interim basis for full-power operation. The Commission will, by rulemaking, determine any future modifications necessary to resolve the ATWS concerns and the required schedule for implementation of such modifications.

15.9 TMI Action Plan Requirements

15.9.1 II.B.1 Reactor Coolant System Vents

In response to Item II.B.1, the applicant stated that a redundant safety-grade vent system will be provided for the reactor coolant system (RCS) and reactor vessel high point. The redundant system design is in accordance with the guidelines of NUREG-0737. The venting system consists of two parallel flow paths with redundant isolation valves in each flow path. The venting operation uses only one of these flow paths at any one time.

The system design with two valves in series in each flow path minimizes the possibility of reactor coolant pressure boundary leakage. The isolation valves are powered by a separate vital power supply. The system is designed to remove noncondensable gases or steam from the RCS via remote manual operations from the control room. The system discharges to the pressurizer relief tank. Additionally, a letdown flow path is provided from the reactor vessel head vent to the excess letdown heat exchanger in the chemical and volume control system.

All piping and equipment from the vessel head vent up to and including the second isolation valve in each flow path are designed according to the requirements of ASME Code, Section III, Class 1. The piping and equipment in the flow paths from the isolation valve to the excess letdown heat exchanger are designed according to the requirements of ASME Code, Section III, Class 2.

On the basis of the above, the staff concludes that the applicant's response meets the criteria of Item II.B.1 of NUREG-0737.

15.9.2 II.K.1.5 Review ESF Value Positions, Controls, and Related Test and Maintenance Procedures To Assure Proper ESF Functioning

In response to Item II.K.1.5, the applicant stated that independent position verification of safety-related components/systems will be performed before the component/system is returned to service, and all proposed operating and maintenance procedures will be completed at least 3 months before fuel loading. On the basis of the above, the staff concludes that the applicant's commitment meets the guideline of this item and is acceptable.

15.9.3 II.K.1.10 Review and Modify Procedures for Removing ESF Equipment From Service To Assure Operability Status is Known

In response to Item II.K.1.10, the applicant stated that administrative control procedures are developed and in use for Millstone Units 1 and 2. Changes necessary to support Millstone Unit 3 will be made in accordance with the Technical Specifications for Millstone Unit 3 at least 6 months before fuel loading. In addition, station procedures will be issued to cover aspects of plant operation. The plant procedures will provide a checklist to ensure the availability and proper alignment of all systems essential to the prevention and mitigation of operational transients and accidents. Items that must be verified on the control console and criteria for acceptable status are included on the checklist. The control room operator on arriving and departing and the shift supervisor on arriving must complete and sign the above checklists. Also checklists of logs are provided for completion by the auxiliary operators and technicians as they arrive and leave. The checklists or logs include any equipment undergoing maintenance or testing that by itself could degrade a system critical to the prevention and mitigation of operational transients and accidents or initiate an operational transient. On the basis above, the staff concludes that the Millstone Unit 3 procedures meet the requirement of this item and are acceptable.

15.9.4 II.K.2.13 Thermal Mechanical Report: Effect of High Pressure Injection On Vessel Integrity for Small-Break LOCA With No Auxiliary Feedwater

Staff review of this item is covered under NRC Unresolved Safety Issue A-49, "Pressurized Thermal Shock," in Appendix C of this SER.

15.9.5 II.K.2.17 Potential for Voiding in the Reactor Coolant System During Transients

Westinghouse has performed a study that addresses the potential for void formation in Westinghouse-designed nuclear steam supply systems during natural circulation cooldown/depressurization transients. This study has been submitted to the NRC by the Westinghouse Owners Group. As stated in a memorandum from R. W. Houston to G. C. Lainas, dated December 6, 1983, the results of this study have been accepted by the staff.

15.9.6 II.K.2.19 Sequential Auxiliary Flow Analysis

The sequential auxiliary feedwater flow analytical requirement is only of concern to plants with a once-through steam generator design. Because Westinghouse uses inverted U-tube steam generator designs, requirements set forth by Item II.K.2.19 are not applicable.

15.9.7 II.K.3.1 Installation and Testing of Automatic PORV Isolation System

In response to Item II.K.3.1, the applicant referenced a generic Westinghouse Owners Group submittal (WCAP-9804). Should the staff's generic review of this material determine that modifications are necessary, the applicant will be required to consider modification of Millstone Unit 3.

15.9.8 II.K.3.3 Reporting SV and PORV Challenges and Failures

The applicant states in FSAR Table 1.10-1 that any failure of a PORV or safety valve to close will be reported promptly to NRC. All challenges to the PORVs or safety valves will be documented in the annual report. On the basis of the above, the staff concludes that the Millstone Unit 3 procedures meet the requirements of this item and are acceptable.

15.9.9 II.K.3.5 Automatic Trip of RCPs During LOCA

In response to Item II.K.3.5, the applicant stated that, according to Westinghouse analyses provided in response to Generic Letter 83-10, sufficient time is available for manual tripping of the pumps; therefore, automatic reactor coolant pump (RCP) trip is not necessary.

The staff is currently reviewing the Westinghouse Owners Group submittals concerning RCP trip during a LOCA. If, as a result of its review, the staff concludes that manual trip of the RCPs is unacceptable, the applicant will be required to review its method of RCP trip in accordance with the resolution of this issue.

15.9.10 II.K.3.10 Proposed Anticipatory Trip Modification

In response to Item II.K.3.10, the applicant stated that an analysis has been performed using realistic yet conservative values for the core physics parameters and a conservatively high initial power, average reactor temperature and pressurizer pressure level. The transient was initiated from 50% of the reactor fuel power level plus 2% for power measurement uncertainty. The applicant concluded that, on the basis of the analysis results, the peak pressure reached in the pressurizer would be 2,302 psia. The transient will not cause the pressurizer PORVs to be challenged because the setpoint for these PORVs is 2,350 psia. The applicant indicated that methodology approved by the staff was used to obtain the analysis results. The staff concludes that Millstone Unit 3 analysis meets the requirements of this item and is acceptable.

15.9.11 II.K.3.17 Report on Outages of ECCS

In response to Item II.K.3.17, the applicant stated that a plan will be developed and implemented for Millstone Unit 3 to compile inforamtion on ECCS components involved in outages. The plan shall require a periodic report that contains (1) ECCS or components involved, (2) outage dates and duration of outage, (3) cause of the outage, and (4) corrective action taken. Test and maintenance outages shall be included. The report shall be reviewed and changes proposed to improve the availability of ECCS equipment, if needed. This plan will be developed before full-power operation. The staff finds the applicant's response acceptable.

15.9.12 II.K.3.25 Effect of Loss of Alternating Current Power On RCP Seals

In response to Item II.K.3.25, the applicant stated that, in the event of loss of offsite power, the RCP motor is deenergized and both of these cooling supplies are terminated. However, the diesel generators are automatically started and either seal injection flow or component cooling water to the thermal barrier heat exchanger is automatically restored within seconds.

Either of these cooling supplies is adequate to provide seal cooling and prevent seal failure resulting from loss of seal cooling during loss of offsite power for at least 2 hours. On the basis of the above, the staff concludes that the design meets the requirement of this item and is acceptable.

15.9.13 II.K.3.30 Revised Small-Break LOCA Methods To Show Compliance With 10 CFR 50, Appendix K

In response to Item II.K.3.30, the applicant stated that Westinghouse is committed to revise its small-break LOCA analysis model to address NRC concerns. This revised Westinghouse model is currently under staff review.

15.9.14 II.K.3.31 Plant-Specific Calculations To Show Compliance With 10 CFR 50.46

In response to Item II.K.3.31, the applicant stated that a small-break LOCA specific to Millstone Unit 3 has been analyzed using the present Westinghouse small-break evaluation model. The results are in conformance with 10 CFR 50, Appendix K, and 10 CFR 50.46. The staff concludes that the applicant's analysis meets the requirements of this item and is acceptable. Nevertheless, if Westinghouse's new model for small-break LOCA evaluation yields more limiting results than the current approved model, the staff will require the applicant to reanalyze the accident with the new model.

15.9.15 III.D.1.1 Integrity of Systems Outside Containment Likely To Contain Radioactive Material for Pressurized-Water Reactors and Boiling-Water Reactors

The applicant states that a program will be developed and implemented to monitor leakage and to reduce detected leakage to as-low-as-practical levels for systems outside the containment that could contain radioactivity.

A program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids in a postaccident situation will include the following:

(1) Review of system design and construction to ensure that the potential for inadvertent releases of radioactive fluids is eliminated

(2) Implementation of all practical leak reduction measures for all systems that could carry radioactive fluid outside containment

(3) Measurement of actual leak rates

(4) A leak reduction program of preventive maintenance to reduce leakage to as-low-as-practical levels. Pressure testing at system operational pressure and integrated leak tests at intervals not to exceed each refueling cycle are typical demonstrations of system integrity.

The applicant further states that a detailed description of the complete program, as described above, will be provided 4 months before fuel loading. The staff will review this program and the results of the initial leak testing at that time. The results of the review will be presented in a supplement to this SER. This is a confirmatory item.

Table 15.1 Assumptions used for estimating the radiological consequences following a postulated control rod ejection accident

Parameter	Value
Power level, MWt	3,636
Primary-to-secondary leak rate as limited by Technical Specifications, gpm	1.0
Fuel rods experiencing cladding failure, releasing all gap radioactivity (assumed to be 10% of the equilibrium core activity of iodines and noble gases), %*	10
Fuel rods experiencing fuel melting, % **	0.25
Iodine transported to and mixed with the secondary coolant which is lost during the course of the accident as a result of loss of offsite power and subsequent steam venting, %	10
Equalization of primary and secondary system pressures terminating the primary-to-secondary leak, sec	~140
Iodine released into the containment which is plated out instantaneously for the containment pathway calculation, %	50
Primary containment leak rate until the containment becomes subatmospheric at 60 min into the accident at which time all leakage stops (containment leakage pathway), %/day	0.90
Assumed iodine concentration in the secondary coolant, μCi/g dose equivalent I-131	0.1
χ/Q values	
0-2 hour EAB (524 m), sec/m^3	5.3×10^{-4}
0-8 hour LPZ (3,862 m), sec/m^3	2.7×10^{-5}

*The released activity is mixed immediately with the primary coolant.
**All the noble gases and 50% of the iodine in this fraction of fuel are released and mixed immediately with the primary coolant.

Table 15.2 Radiological consequences of design-basis accidents*

Postulated accident	Exclusion area boundary, rem		Low population zone, rem	
	Thyroid	Whole body	Thyroid	Whole body
<u>Loss of coolant</u>				
Containment leakage				
0-2 hours	158	21	-	-
0-8 hours			8.0	1.1
8-24 hours			0.0	0.0
24-96 hours			0.0	0.0
96-720 hours			0.0	0.0
Total containment leakage	158	21	8.0	1.1
ECCS component leakage	<0.1	<0.1	<0.1	<0.1
Totals	158	21	8.0	1.1
<u>Steamline break outside secondary containment</u>				
Long-term operation case (cases 1 and 2)	14.5	<0.1	2.3	<0.1
Short-term operation case (case 3)	14.6	<0.1	2.5	<0.1
<u>Control rod ejection</u>				
Containment leakage pathway	11.8	0.7	2.2	<0.1
Secondary system release pathway	0.2	<0.1	<0.1	<0.1
<u>Fuel-handling accident</u>				
Fuel-handling area	1.8	0.6	<0.1	<0.1
<u>Small-line break</u>	9.6	<0.1	0.5	<0.1
<u>Steam generator tube rupture</u>				
Case 1 (DEI-131 at 60 μCi/gm)	*	*	*	*
Case 2 (DEI-131 at 1 μCi/gm)	*	*	*	*

*The short-term diffusion estimates (χ/Qs) used in this analysis are those presented and discussed in SER Section 2.3.4 and summarized in Tables 15.1 and 15.3 through 15.6. The meteorological models described in regulatory guides referenced in these analyses are modified by those presented in RG 1.145. See SER Section 2.3.4 for further discussion of the meteorological models.

Table 15.3 Assumptions used in accidents involving small-line breaks outside containment

Parameter and unit of measure	Quantity
Coolant released, lb_m	35,000
Fraction of coolant released flashed to steam, %	39
Coolant contaminant concentration, μCi/g	1.0
Spiking factor (iodine release rate multiplier)	500

Table 15.4 Assumptions used to evaluate the radiological consequences following a postulated main steamline break accident outside containment

Parameter	Value
Power level, MWt	3,636
Preaccident dose-equivalent I-131 in primary coolant (case 2), μCi/g	1.0
Preaccident dose-equivalent I-131 in primary coolant (case 3), μCi/g	60.0
Primary-to-secondary leak rate, as limited by Technical Specifications, gpm	1.0
Amount of the 1-gpm leak which occurs in the affected steam generator	All
Amount of iodine transported to the shell side of the steam generator by the leakage which is lost to the environment without decay	All
Increase in iodine release rate from fuel as a result of the accident (case 2)	Factor of 500
χ/Q values	
0-2 hour EAB (524 m), sec/m^3	5.3×10^{-4}
0-8 hour LPZ (3,862 m), sec/m^3	2.7×10^{-5}

Table 15.5 Assumptions used in the calculation of LOCA doses

Parameter and unit of measure	Quantity
Containment leakage	
Power level, Mwt	3,636
Operating time, years	3
Fraction of core inventory available for containment leakage, %	
Iodine	25
Noble gases	100
Initial iodine composition in containment, %	
Elemental	91
Organic	4
Particulate	5
Containment leak rate, %/day	
First hour	0.9
After first hour	0.0
Bypass leakage, %	1.03
Containment volume, ft^3	2.3×10^6
Supplementary leak collection and release system filter efficiencies, %	
Elemental	99
Organic	99
Particulate	99
Relative concentration values, seconds per cubic meter	
0-2 hours at exclusion area boundary	5.3×10^{-4}
0-8 hours at low population zone (LPZ) boundary	2.7×10^{-5}
8-24 hours at LPZ boundary	1.9×10^{-6}
24-96 hours at LPZ boundary	8.4×10^{-6}
96-720 hours at LPZ boundary	2.7×10^{-6}
Emergency core cooling system leakage outside containment	
Power, Mwt	3,636
Sump volume, gal	1,322,000
Flash fraction	0.1
Leak rate (twice maximum operational leakage defined in FSAR Table 15.6-9), gph	2.64
Leak duration, hr	720
Delay time, hr	0.06
Auxiliary building exhaust vent system filter efficiency for iodine, %	
Elemental and particulate	99
Organic	99

Table 15.6 Assumptions used for estimating the radiological consequences following a postulated fuel-handling accident

Parameter and unit of measure	Value
Power level, MWt	3,636
Number of fuel rods damaged	314
Total number of fuel rods in core	50,952
Radial peaking factor of damaged rods	1.65
Shutdown time, hr	100
Inventory released from damaged rods, % iodines and noble gases	10
Pool decontamination factors	
Iodines	100
Noble gases	1
Iodine fractions released from pool, %	
Elemental	75
Organic	25
Iodine removal efficiencies for fuel building exhuast system, %	
Elemental	99
Organic	99
χ/Q values	
0-2 hour EAB (524 m), sec/m^3	5.3×10^{-4}
0-8 hour LPZ (3,862 m), sec/m^3	2.7×10^{-5}

16 TECHNICAL SPECIFICATIONS

The Technical Specifications in a license define certain features, characteristics, and conditions governing operation of a facility that cannot be changed without prior approval of the staff. The finally approved Technical Specifications will be made a part of the operating license. Included will be sections covering definitions, safety limits, limiting safety system settings, limiting conditions for operation, surveillance requirements, design features, and administrative controls.

The Technical Specifications for Millstone Unit 3 will be based on "Standard Technical Specifications for Westinghouse Pressurized Water Reactors" (NUREG-0452, Rev. 4). This document has been updated from earlier revisions as a result of continued discussion with Westinghouse and other licensees with Westinghouse PWRs.

The staff is working with the applicant to prepare a draft of the Technical Specifications for Millstone Un.t 3. On the basis of its review to date, the staff concludes that normal plant operation within the limits of the Technical Specifications will not result in offsite exposure in excess of the 10 CFR 20 limits. Furthermore, the limiting conditions for operation and surveillance requirements will ensure that necessary engineered safety features will be available in the event of malfunctions within the facility.

During its review of Millstone Unit 3, the staff identified certain issues that must be included in the Technical Specifications as a condition of staff acceptance. These issues are identified in Table 16.1 and are discussed further in sections of this report as indicated in parentheses after each item. Most of the issues that the staff has identified as being required to be included in the Millstone Unit 3 Technical Specifications are already addressed in NUREG-0452. Those issues that are not included in NUREG-0452 will be added to the Technical Specifications being prepared for Millstone Unit 3.

Table 16.1 Technical Specification items

Item	SER Section
(1) Securing of watertight docks into service water cubicles	2.4.5
(2) Intake water temperature monitoring	2.4.11.2
(3) Containment isolation valves in purge/vent system test every 6 months	6.2.6
(4) Periodic testing to ensure control room leaktightness	6.4
(5) ESF atmosphere cleanup system flow rate	6.5.1
(6) Periodic surveillance of the battery float charge	8.3.2.1
(7) Deenergizing of 12 safety-related motor-operated valves during normal plant operation	8.3.3.1.1
(8) Periodic testing and calibration of interrupting devices	8.3.3.3.15
(9) Fuel oil quality and tests conformance to Regulatory Guide 1.137	9.5.4.2
(10) Lubricating oil storage	9.5.7
(11) Capability to transfer lubricating oil from storage to diesel generator	9.5.7
(12) Lubricating oil inventory in storage	9.5.7
(13) Opening of access hatch in emergency diesel generator combustion exhaust system	9.5.8
(14) Open access hatch once per year	9.5.8
(15) Main steam stop and control valves and reheat valves	10.2

17 QUALITY ASSURANCE

17.1 General

The description of the quality assurance (QA) program for the operations phase of Millstone Unit 3 is contained in FSAR Section 17.2, which includes a reference to the latest NRC-accepted revision of the report entitled "Northeast Utilities Quality Assurance Program Topical Report" (NU-QA-1, Rev. 5A). The staff's evaluation of this QA program is based on a review of this information, discussions with representatives from Northeast Utilities Services Company, and Northeast Utilities' responses, contained in FSAR Amendment 3, to questions generated by the staff.

The staff assessed Northeast Utilities' QA program for the operations phase to determine if it complies with the requirements of 10 CFR 50, Appendix B, the applicable QA-related regulatory guides listed in Table 17.1; and SRP Section 17.2 (NUREG-0800).

17.2 Organization

The structure of the Northeast Utilities' organization responsible for the operation of Millstone Unit 3 and for the establishment and implementation of the QA program for the operations phase is shown in Figure 17.1. The President and Chief Operating Officer has ultimate responsibility for the establishment and execution of the operational QA program. Authority for the establishment and execution of this QA program is delegated to the Executive Vice President, Engineering and Operations. He is responsible for engineering, construction, operation, maintenance, modification, and QA within Northeast Utilities. Authority for the nuclear engineering, operation, maintenance, modification, and QA for Millstone Unit 3 is delegated to the Senior Vice President, Nuclear Engineering and Operations. He has the specific responsibility for the program management in accordance with the Nuclear Engineering and Operations Policy Statement, "Quality Assurance Program," and he resolves disputes arising from a difference of opinion between QA personnel and other department personnel that are not resolved by lower management. The three vice presidents reporting to the Senior Vice President, Nuclear Engineering and Operations, are:

(1) The Vice President, Generation Engineering and Construction, is responsible for modification, backfit, and betterment projects during the operations phase of Millstone Unit 3. Within his organization is a Construction Quality Control Branch that performs inspections of activities performed as part of these projects.

(2) The Vice President, Nuclear Operations, is responsible for the operation and maintenance of Millstone. Within his organization for Millstone is the Millstone QA Supervisor who reports through the Quality Services Supervisor and the Station Services Superintendent to the Millstone Station Superintendent, independent of the Unit Superintendent, Millstone Unit 3, who is responsible for Millstone Unit 3 operation. Overall responsibility for implementing the requirements of the Millstone Unit 3 operational QA

program is assigned to the Unit Superintendent. The Millstone QA Supervisor is responsible for first-line verification of implementation of the requirements. He is supported by a staff of quality assurance/quality control (QA/QC) engineers and technicians. He is present or represented at daily Millstone Unit 3 work schedule and status meetings and is thus aware of day-to-day assignments throughout the plant and can provide adequate QA/QC coverage. He provides nuclear generation facility management and the Manager, Quality Assurance, with objective evidence of the implementation of the QA program within the facility. He has the authority and organizational freedom and is sufficiently removed from undue cost and schedule influences to perform QA functions effectively, including responsibilities (a) to stop unsatisfactory work and control further processing, delivery, or installation of nonconforming materials as delineated in writing; (b) to identify quality problems; (c) to initiate, recommend, or provide solutions; and (d) to verify implementation of solutions.

(3) The Vice President, Nuclear and Environmental Engineering, is responsible for nuclear engineering and operations services, which include QA. The Manager, Quality Assurance, reports through the Director Nuclear Engineering and Operations Services, to the Vice President, Nuclear and Environmental Engineering. The Manager, Quality Assurance, is responsible for the preparation and issuance of the "QA Program Topical Report" and verification of the implementation of its requirements. Verification is performed by a planned program for audits, inspections, and surveillances. He provides management with objective evidence of the performance of activities affecting quality, independently of the individual or group directly responsible for performing the specific activity. He has the authority and organizational freedom to ensure all necessary quality-affecting activities are performed. He is independent of undue influences and responsibilities for schedules and costs. He has the responsibility and authority, delineated in writing, to stop unsatisfactory work and control further processing, delivery, or installation of nonconforming materials.

The staff finds the applicant's organization for QA acceptable.

17.3 Quality Assurance Program

In addition to descriptive material contained in the Northeast Utilities' topical report on quality assurance, the operations phase of the QA program has detailed company procedures. A summary of the topics addressed in these procedures and their relationship to the QA requirements of Appendix B to 10 CFR 50 is presented in the topical report.

Procedures and work instructions necessary to implement the requirements of the operations phase program are developed by the organization responsible for the activity. Lower tier procedures and instructions are contained in manuals, station procedures and directives, administrative instructions, and/or other documents. Onsite implementation of procedures and work instructions is the responsibility of the Unit Superintendent of Millstone Unit 3. QA personnel verify that the procedures are followed by means of inspections, audits, and other surveillance. Procedures for such inspections, audits, and surveillance are developed, approved, and implemented by the QA organization.

Inspections are performed using preplanned checklists in accordance with written and approved inspection plans. The qualifications of inspectors (and their current status) to conduct inspections, tests, and examinations are based on applicable codes, standards, and Northeast Utilities' training programs.

The QA organizations are responsible for the content and control of the audit program. Audits are performed in accordance with written procedures or checklists by appropriately trained QA personnel who do not have direct responsibility in the area being audited. The audit activities described in the topical report are conducted at least annually, or on a more frequent basis as determined by the QA organization. These include an objective evaluation of QA practices, procedures, and instructions; work areas, activities, processes, and items; effectiveness of implementation of the QA program; and compliance with policy directives.

The QA program requires that both documentation of audit results and formal notification of the audit findings be provided to the Manager Quality Assurance, and to the management of the audited function. Audit findings, which indicate quality trends and the effectiveness of the QA program, are also reported to the Senior Vice-President, Nuclear Engineering and Operations. Management for the area audited implements any corrective action needed. Followup audits are performed to determine that nonconformances are effectively corrected and that the corrective action precludes repetitive occurrences.

An indoctrination and training program is established to ensure that persons involved in quality-related activities are knowledgeable in QA instructions and requirements and demonstrate a high level of competence and skill in the performance of their quality-related activities. A program for retraining of such persons is provided to ensure that they maintain their proficiency.

17.4 Conclusion

On the basis of its detailed review and evaluation of the QA program description in FSAR Section 17.2 and NU-QA-1 as referenced therein, the staff concludes that

(1) The Northeast Utilities' organization gives QA personnel sufficient (a) independence from cost and schedule (when opposed to safety consideration), (b) authority to effectively carry out the operations QA program, and (c) access to management at a level necessary to perform their QA functions.
(2) The QA program describes requirements, procedures, and controls that, when properly implemented, comply with the requirements of Appendix B to 10 CFR 50 and with the acceptance criteria contained in SRP Section 17.2 (NUREG-0800).

Accordingly, the staff concludes that Northeast Utilities' description of the QA program for operations is in compliance with applicable NRC regulations, meets the requirements of Appendix B to 10 CFR 50, and is acceptable.

The staff review of the list of items to which the QA program applies is incomplete and this is an open item. The list of items is being reviewed by the staff technical review branches to ensure that safety-related items within their scope of review are under the QA program controls. Differences between the staff and the applicant regarding the list will be resolved to the staff's satisfaction before closing this open item. The list includes safety-related items reflected in NUREG-0737.

In responding to staff questions regarding QA, Northeast Utilities committed to revise either the FSAR or its topical report on QA to include the applicable responses. This has not yet been done and is a confirmatory item. The staff also will require that the FSAR reference Rev. 5A of NU-QA-1 instead of Rev. 4A before closing this confirmatory item.

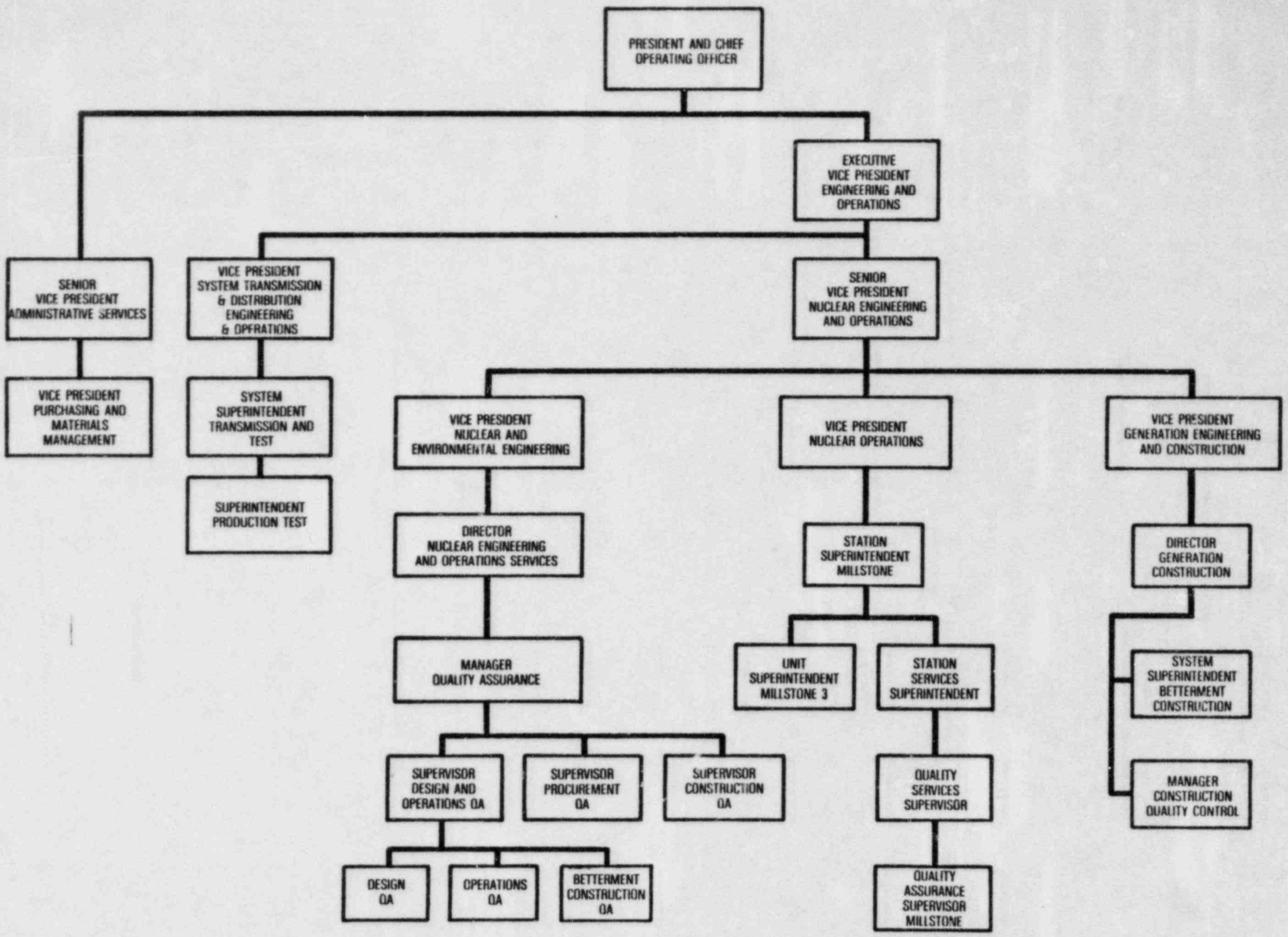

Figure 17.1 Organization for operations quality assurance

Table 17.1 Regulatory guidance applicable to quality assurance program

Regulatory Guide, revision, and date	Title
1.8 Rev. 1-R May 1977	Personnel Selection and Training
1.30 Aug. 11, 1972	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electrical Equipment
1.33 Rev. 2 Feb. 1978	Quality Assurance Program Requirements (Operation)
1.37 Mar. 16, 1973	Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants
1.38 Rev. 2 May 1977	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants
1.39 Rev. 2 Sept. 1977	Housekeeping Requirements for Water-Cooled Nuclear Power Plants
1.58 Rev. 1 Sept. 1980	Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel
1.64 Rev. 2 June 1976	Quality Assurance Requirements for the Design of Nuclear Power Plants
1.74 Feb. 1974	Quality Assurance Terms and Definitions
1.88 Rev. 2 Oct. 1976	Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records
1.94 Rev. 1 April 1976	Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants
1.116 Rev. 0-R May 1977	Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems
1.123 Rev. 1 July 1977	Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants

Table 17.1 (Continued)

Regulatory Guide, revision, and date	Title
1.144 Rev. 1 Sept. 1980	Auditing of Quality Assurance Programs for Nuclear Power Plants
1.146 Aug. 1980	Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants

18 HUMAN FACTORS ENGINEERING

All licensees and applicants for an operating license are required to conduct a detailed control room design review (DCRDR) and to provide a safety parameter display system (SPDS) in response to NRC Task Action Plan Items I.D.1 and I.D.2 (NUREG-0660, May 1980; NUREG-0737, November 1980 as supplemented by Generic Letter 82-33, December 17, 1982). The purpose of the DCRDR is to identify and correct human engineering discrepancies (HEDs) which might affect the operator's ability to prevent or cope with an accident. DCRDRs should be conducted using the guidance provided in NUREG-0700, "Guidelines for Control Room Design Reviews," dated September 1981. The purpose of the SPDS is to continuously display information from which the plant safety status can be readily and reliably assessed. The principal function of the SPDS is to aid the control room personnel during abnormal and emergency conditions in determining the safety status of the plant and in assessing whether abnormal conditions warrant corrective action by operators to avoid a degraded core. A written SPDS safety analysis shall be prepared describing the basis on which the selected parameters are sufficient to assess the safety status of each identified function for a wide range of events, which include symptoms of severe accidents.

In response to Generic Letter 82-33 and in additional information submitted on November 18, 1983, the applicant provided a program plan and stated that a summary report will be submitted in June 1984. Following review of the Millstone Unit 3 DCRDR Program Plan, the staff will conduct an in-progress audit and on the basis of an assessment of the DCRDR Summary Report, the staff will decide if a preimplementation onsite audit is required. This decision will be made, and the licensee informed, within 2 weeks after the staff's receipt of the report. If required, the audit will be accomplished within 1 month after receipt of the report. Within a month after the onsite audit, the staff will issue a supplement to this SER.

The staff will review the applicant's safety analysis and SPDS implementation plan to confirm (1) the adequacy of the parameters selected to be displayed to detect critical safety functions, (2) that means are provided to ensure that the data displayed are valid, (3) the adequacy of the design and installation of the system from a human factors perspective, and (4) the adequacy of the verification and validation program to ensure a highly reliable SPDS.

19 REPORT OF THE ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

The Millstone Unit 3 application for an operating license will be reviewed by the Advisory Committee on Reactor Safeguards. The NRC staff will issue a supplement to this Safety Evaluation Report after the Committee's report to the Commission is available. The supplement will include a copy of the Committee's report, will address comments made by the Committee, and will describe steps taken by the NRC staff to resolve any issues raised as a result of the Committee's review.

20 COMMON DEFENSE AND SECURITY

The application reflects that the activities to be conducted will be within the jurisdiction of the United States and that all of the directors and principal officers of the applicant are citizens of the United States. Northeast Nuclear Energy Company, the applicant, is not owned, dominated, or controlled by an alien, a foreign corporation, or a foreign government. The activities to be conducted do not involve any restricted data, but in accordance with the requirements of 10 CFR 50, the applicant has agreed to safeguard any such data that might become involved. The applicant will rely on obtaining fuel as it is needed from sources of supply available for civilian purposes so that no diversion of special nuclear material for military purposes is involved. For these reasons, and in the absence of any information to the contrary, the staff finds that the activities to be performed will not be inimical to the common defense and security.

21 FINANCIAL QUALIFICATIONS

The NRC regulations that relate to financial data and information required to establish financial qualifications for applicants for a facility operating license are in 10 CFR 50.33(f) and in Appendix C to 10 CFR 50. To ensure that the staff has the latest information on which to base its determination of the financial qualifications of applicants, it is the current practice of the staff to review this information during the later stages of its review of an application. Thus, the staff is continuing its review of the applicant's financial qualifications and will report the results of its evaluations in a supplement to this SER.

22 FINANCIAL PROTECTION AND INDEMNITY REQUIREMENTS

22.1 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 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 50.

22.2 Preoperational Storage of Nuclear Fuel

The Commission's regulations in 10 CFR 140 require that each holder of a construction permit under 10 CFR 50 who is also the holder of a license under 10 CFR 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 an operating license is issued under 10 CFR 50) shall, during the interim storage period before 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 before, and the indemnity agreement executed as of, the effective date of the 10 CFR 70 license. Payment of an annual indemnity fee is required.

The applicant will furnish to the Commission proof of financial protection in the amount of $1,000,000 in the form of a Nuclear Energy Liability Insurance Association Policy (Nuclear Energy Liability Policy, Facility Form NF-256). 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.

22.3 Operating Licenses

Under the Commission's regulations in 10 CFR 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 that must be maintained for Millstone Unit 3 (which has a rated capacity in excess of 100,000 electrical kilowatts) is the maximum amount available from private sources (i.e., the combined capacity of the two nuclear liability insurance pools; this amount is currently $570 million).

Accordingly, a license authorizing operation of Millstone Unit 3 will not be issued until proof of financial protection in the requisite amount has been received and the requisite indemnity agreement executed.

The staff expects that, in accordance with the usual procedure, the nuclear liability insurance pools will provide in writing before the operating license document is issued, on behalf of the applicant, evidence that the present coverage has been appropriately amended so that the policy evidence limits have been increased to meet the requirements of the Commission's regulations for reactor operation. Similarly, an operating license 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 NRC regulations.

On the basis of the above considerations, the staff concludes that the currently applicable requirements of 10 CFR 140 have been satisfied and that, before an operating license is issued, the applicant will be required to comply with the provisions of 10 CFR 140 applicable to operating licenses, including those as to proof of financial protection in the requisite amount and as to execution of an appropriate indemnity agreement with the Commission.

23 CONCLUSIONS

On the basis of its evaluation of the application as set forth above, the staff has determined that, upon favorable resolution of the outstanding matters described herein, it will be able to conclude that

(1) The application for a facility license filed by the applicant complies with the requirements of the Atomic Energy Act of 1954, as amended (Act), and the Commission's regulations set forth in 10 CFR Chapter I, except as duly exempted therefrom.

(2) The facility will operate in conformity with the application as amended, the provisions of the Act, and the rules and regulations of the Commission.

(3) There is reasonable assurance (a) that the activities authorized by the operating license can be conducted without endangering the health and safety of the public and (b) that such activities will be conducted in compliance with regulations of the Commission set forth in 10 CFR Chapter I.

(4) The applicant is technically and financially qualified to engage in the activities authorized by the licenses, in accordance with the regulations of the Commission set forth in 10 CFR Chapter I.

(5) The issuance of these licenses will not be inimical to the common defense and security or to the health and safety of the public.

Before an operating license is issued, 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 level must be verified by the NRC before the license is issued.

Before on operating license is issued, the applicant will be required to satisfy the applicable provisions of 10 CFR 140.

APPENDIX A

CHRONOLOGY OF THE NRC STAFF RADIOLOGICAL REVIEW OF THE MILLSTONE NUCLEAR POWER STATION, UNIT 3

October 29, 1982	Letter from applicant concerning application for operating license. The application (general information), FSAR, Environmental Report - Operating License stage and information for antitrust review were received.
November 1, 1982	Letter from applicant transmitting the primary features of the physical security plan as part of the operating license application for Millstone Unit 3.
November 9, 1982	Letter from applicant transmitting the piping and instrumentation diagrams (electrical, instrumentation and control drawings) with the application for operating license.
December 9, 1982	Letter from applicant asking to make a minor change to the physical security plan letter submitted as part of the operating license application for Millstone Unit 3.
January 7, 1983	Letter from applicant concerning guidelines for preparing and implementing inplant drill programs for nuclear power plants.
January 24, 1983	Letter to applicant transmitting Federal Register Notice for publication on February 7, 1983, with an intervention date of March 9, 1983. Also forwarding legal notice that will be printed in NRC-approved trade journals.
January 31, 1983	Letter to applicant advising that the operating license application is acceptable for docketing.
February 2, 1983	Letter from applicant transmitting required number of copies to docket operating license application. Application received in docket files on February 2, 1983, the docketing date.
February 2, 1983	Letter from applicant transmitting the application for an operating license for the Millstone Nuclear Power Station, Unit No. 3. Application docketed on February 2, 1983.

February 11, 1983	Letter from applicant transmitting an affidavit of distribution of application to Federal, State, and local officials.
February 24, 1983	Letter to applicant advising of newspapers to which the staff sent a display ad and transmitting the Federal Register Notice to be published on March 4, 1983.
February 28, 1983	Letter from applicant transmitting Amendment 1 to the Environmental Report - Operating License (ER-OL) stage and Final Safety Analysis Report (FSAR).
March 23, 1983	Letter to applicant concerning management meeting on Millstone Unit 3 operating license application.
March 31, 1983	Letter from applicant transmitting a response to the requests for additional information that resulted from the acceptance review.
March 31, 1983	Letter from applicant transmitting references in response to acceptance review requests for additional information.
April 5, 1983	Letter from applicant transmitting FSAR Figures 12.3-1 through 12.3-4 and 12.3-6 through 12.3-9.
April 12, 1983	Representatives from NRC and Northeast Utilities meet in Bethesda, Maryland, to discuss the plans for the review of the docketed OL application for Millstone Unit 3. (Summary issued May 12, 1983)
April 15, 1983	Letter from applicant transmitting Amendment 2 to the FSAR and Environmental Report.
April 26, 1983	Letter from applicant transmitting the 1982 Annual Financial Report.
May 3, 1983	Letter to applicant transmitting a request for additional information; questions are a result of the review of the information in the FSAR.
May 31, 1983	Letter to applicant concerning request for additional information.
May 31, 1983	Letter from applicant transmitting a revision to the NRC service list.
June 2, 1983	Representatives from NRC and Northeast Utilities meet in Bethesda, Maryland, to discuss the NRC staff's safety review questions concerning certain portions of the FSAR. (Summary issued October 11, 1983)

June 9, 1983	Representatives from NRC and Northeast Utilities meet in Bethesda, Maryland, to discuss concerns of the NRC staff.
June 27, 1983	Letter to applicant concerning NRC staff environmental review site visit.
June 29, 1983	Letter to applicant requesting additional information.
June 29, 1983	Representatives from NRC and Northeast Utilities meet in Bethesda, Maryland, to discuss specific design information needed by the staff to complete its review of reactor containment pressure boundary materials GDC 51 compliance).
June 30, 1983	Letter from applicant concerning responses related to the operating license application for Millstone Unit 3.
July 7, 1983	Letter from applicant transmitting a response to request for additional information (Question 451.1). (Section 2.3.3)).
July 7, 1983	Representatives from NRC and Northeast Utilities meet in Waterford, Connecticut (site), and Berlin, Connecticut (corporate offices), to discuss applicant's responses to staff's safety review questions concerning quality assurance.
July 20 and 21, 1983	Representatives from NRC and Northeast Utilities meet at site for a tour on July 20, 1983, and an open meeting for the public on environmental matters on July 21, 1983. (Summary issued August 18, 1983)
July 22, 1983	Letter from applicant concerning a response to selected requests for additional information.
July 26, 27, and 28, 1983	Representatives from NRC, Northeast Utilities, and Stone & Webster meet at Stone & Webster offices in Boston, Massachusetts, to discuss applicant's responses concerning instrumentation and control systems items. (Summary issued September 13, 1983)
July 27, 1983	Letter from applicant transmitting Amendment 3 to the Environmental Report.
July 27, 1983	Letter from applicant transmitting a probabilistic safety study (PSS).
August 1, 1983	Letter from applicant concerning response to selected requests for additional information.

August 17, 1983	Letter from applicant transmitting an application for construction permit amendment concerning ownership change in shares.
August 19, 1983	Letter from applicant concerning comments on proposed requirements related to the steam generator.
August 29, 1983	Letter from applicant transmitting Amendment 3 to the FSAR.
August 29, 1983	Letter from applicant concerning response to selected requests for additional information.
August 31, 1983	Letter to applicant transmitting Amendment 11 to construction permit CPPR-113 concerning a change in ownership shares.
August 31, 1983	Letter from applicant transmitting Amendment 3 to the Environmental Report.
September 1, 1983	Letter from applicant concerning deferral of responses to two acceptance review questions and an enclosure.
September 2, 1983	Letter from applicant concerning comments on NUREG-0897, "Containment Emergency Sump Performance"; Standard Review Plan Section 6.2.2, Revision 4, "Containment Heat Removal Systems"; and NUREG-0869, "USI A-43 Resolution Positions."
September 7, 1983	Representatives from NRC and Northeast Nuclear Energy Company (NNECO) meet in Bethesda, Maryland, to hear technical presentation by NNECO on probabilistic safety study methodology and results. (Summary issued November 23, 1983)
September 13, 1983	Letter from applicant concerning NRC staff instrumentation and control systems review meeting.
September 15, 1983	Representatives from NRC and Northeast Utilities meet in Waterford, Connecticut, to observe hydrologic conditions at selected locations at the Millstone site. (Summary issued October 20, 1983)
September 27, 1983	Letter from applicant transmitting Amendment 4 to the Environmental Report.
September 27, 1983	Letter from applicant transmitting remaining responses to the 640-series questions.
September 30, 1983	Letter to applicant concerning use of Code Case N-310-1.

October 3, 1983	Letter to applicant requesting additional information from staff on geosciences and instrumentation and control systems items.
October 7, 1983	Letter to applicant transmitting a request for additional information.
October 11, 1983	Letter from applicant concerning the deferral of responses to three acceptance review questions and Enclosure 4, Item 7.
October 11 and 12, 1983	Representatives from NRC and Northeast Utilities meet at the Millstone Unit 3 site to tour the plant in support of the NRC staff's review of the probabilistic safety study.
October 18, 1983	Letter to applicant concerning NRC staff visit to Millstone Unit 3 to assess the status of construction.
October 19, 20, and 21, 1983	Representatives from NRC, Stone & Webster, and Northeast Utilities meet at the Millstone site to perform mechanical engineering confirmatory pipe audit.
October 24, 1983	Letter from applicant concerning Amendment 11 to construction permit CPPR-113 requesting corrections to pages of the Amendment 11 package.
October 24, 1983	Letter from applicant concerning response to request for additional information (Question 230.6 (Standard Review Plan Sections 2.5.2.2, 2.5.2.3, and 2.5.2.4)).
October 25, 26, and 27, 1983	Representatives from NRC and Northeast Utilities meet in Berlin, Connecticut, at the Northeast Utilities general offices and in Waterford, Connecticut, at the site to review construction progress to collect data for assessing projected fuel loading date.
October 26, 1983	Letter to applicant concerning clarification of required actions based on generic implications of Salem anticipated transient without scram (ATWS) events (Generic Letter 83-28).
November 7, 1983	Letter from applicant transmitting telephone conversation references requested in Question 291.19.
November 7, 1983	Letter from applicant transmitting responses to requests for additional information.
November 7, 1983	Letter to applicant concerning clarification of required actions based on generic implications of Salem ATWS events (Generic Letter 83-28).

November 10, 1983	Letter from applicant concerning control room design review implementation plan.
November 21, 1983	Letter from applicant submitting seismic studies referenced in the response to Question 230.6 and requesting the withholding from public disclosure as proprietary the two-volume report entitled "Seismological and Geological Studies: Miramichi Area New Brunswick and Central New Hampshire."
November 29, 1983	Representatives from NRC and Northeast Utilities meet in Bethesda, Maryland, to discuss applicant's response to Question 230.6. (Summary issued December 14, 1983)
November 29, 1983	Letter from applicant transmitting Amendment 5 to the FSAR and response to selected requests for additional information.
December 1, 1983	Representatives from NRC and Northeast Utilities meet in Bethesda, Maryland, to discuss additional information to be provided by the applicant concerning Attachment II to a letter from W. G. Counsil to B. J. Youngblood, dated September 13, 1983. (Summary issued December 14, 1983)
December 5, 1983	Letter to applicant requesting additional information on mechanical engineering items.
December 8, 1983	Letter from applicant concerning response to acceptance review Question 440.7.
December 13 and 14, 1983	Representatives from NRC, NRC staff consultants, and Northeast Utilities personnel meet on December 13, 1983, at the Northeast Utilities general offices in Berlin, Connecticut, and on December 14, 1983, at the Millstone Unit 3 site in Waterford, Connecticut, to discuss the status of review of the Millstone Unit 3 probabilistic safety study (PSS) and to tour plant in support of PSS review.
December 14, 1983	Letter from applicant transmitting additional information requested by staff on geosciences items.
December 14, 1983	Representatives from NRC and Northeast Utilities meet at the Millstone Unit 3 site in Waterford, Connecticut, to observe certain aquatic and terrestrial conditions at the site - environmental site tour.
December 14, 1983	Letter from applicant transmitting viewgraphs presented in applicant's November 29, 1983, meeting with staff.
December 14, 1983	Letter from applicant transmitting additional information on right-of-way development and management plan for the transmission line.

December 16, 1983	Letter from applicant concerning facility staffing survey.
December 16, 1983	Letter from applicant transmitting a response to Question 420.06.
December 19, 1983	Letter from applicant concerning meeting with staff.
December 23, 1983	Letter to applicant transmitting proprietary report by Weston Geophysical Corporation, "Seismology and Geology Studies: Miramichi Area, New Brunswick and Central New Hampshire."
December 28, 1983	Letter from applicant requesting removal of proprietary status of report submitted in response to Question 230.6.
December 30, 1983	Letter to applicant transmitting a draft Safety Evaluation Report (SER) for Millstone Unit 3 and requesting a written response to the open items before May 15, 1984. The schedule for issuance of the final SER is July 16, 1984.
January 5, 1984	Representatives from NRC and Northeast Utilities meet in Bethesda, Maryland, to discuss applicant's position on the New Brunswick earthquake as it relates to the Millstone Unit 3 site in support of its response to Question 230.6. (Summary issued February 1, 1984)
January 6, 1984	Letter to applicant requesting additional information as a result of the staff's preliminary review of the Millstone Unit 3 probabilistic safety study.
January 10, 1984	Letter from applicant concerning the probabilistic safety study.
January 10, 1984	Letter from applicant concerning certification of compliance of operation with the Connecticut Coastal Area Management Program.
January 13, 1984	Letter from applicant advising that the response to Question 420.6 had several pages missing.
January 16, 1983	Letter to applicant concerning request for additional information on containment systems.
January 16, 1984	Letter from applicant transmitting Amendment 6 to the FSAR and revised responses to selected requests for additional information.
January 17, 18, and 19, 1984	Representatives from NRC, NRC consultant, and Northeast Utilities meet at the Stone & Webster offices in Boston, Massachusetts, to discuss applicant's safety review Questions 210.8-210.43 listed in B. J. Youngblood letter, dated December 5, 1984. (Summary issued March 1, 1984)

January 20, 1984	Letter from applicant transmitting Amendment 5 to the Environmental Report and responses to selected requests for additional information.
January 30, 1984	Letter from applicant concerning scheduled meetings to resolve open items in the draft Safety Evaluation Report.
February 1, 1984	Letter from applicant transmitting responses to requests dated December 5, 1983, for additional information.
February 1, 1984	Letter from applicant transmitting meeting summary and viewgraphs presented in applicant's January 5, 1984, meeting with staff.
February 3, 1984	Letter to applicant requesting comments on preliminary report, "A Review of the Millstone 3 Probabilistic Safety Study."
February 6, 1984	Letter from applicant concerning the probabilistic safety study.
February 10, 1984	Letter from applicant transmitting the boron dilution analysis for failures in the chemical and volume control system.
February 10, 1984	Letter from applicant concerning probabilistic safety study input listings.
February 14, 1984	Representatives from NRC and Northeast Utilities meet in Bethesda, Maryland, to discuss applicant's responses to 241.0-series safety review questions contained in Enclosure 2 of NRC letter, dated January 16, 1984. (Summary issued March 27, 1984)
February 15, 1984	Letter from applicant concerning Westinghouse topical report on sensor response time.
February 16, 1984	Representatives from NRC and Northeast Utilities meet in Bethesda, Maryland, to discuss applicant's responses to fire protection unresolved items contained in draft SER, Section 9.5.1. (Summary issued March 9, 1984)
February 16, 1984	Letter from applicant transmitting a response to Question 492.4 and requesting that it be withheld from public disclosure as proprietary reference CAW-83-65 and respond to R. A. Wiesman, Westinghouse Electric Corporation.
February 17, 1984	Letter to applicant requesting additional information for Millstone Unit 3.

February 21, 1984	Representatives from NRC and Northeast Utilities meet in Bethesda, Maryland, to discuss Millstone Unit 3 program for maintaining safe shutdown capability. (Summary issued May 15, 1984)
February 21, 1984	Letter from applicant concerning licensing schedule for Draft Environmental Statement.
February 22, 1984	Representatives from NRC and Northeast Utilities meet in Bethesda, Maryland, to discuss Millstone simulator examination and training programs including those specifically for Millstone Unit 3 simulator. (Summary issued March 20, 1984)
February 23, 1984	Representatives from NRC and Northeast Utilities meet in Bethesda, Maryland, to discuss applicant's responses to 430.0-series safety review questions contained in Enclosure 3 to letter from B. J. Youngblood to W. Counsil dated February 3, 1984.
February 24, 1984	Letter to applicant concerning Draft Safety Evaluation Report for Millstone Unit 3.
February 27-March 2, 1984	Representatives from NRC and Northeast Utilities meet in Waterford, Connecticut, and Boston, Massachusetts (Stone & Webster offices), to review structural design of Millstone Unit 3 as part of the staff's safety evaluation. (Summary issued March 23, 1984)
March 1, 1984	Representatives from NRC and Northeast Utilities meet at the Millstone station site construction office in Waterford, Connecticut, to tour the plant and discuss applicant's responses to unresolved items contained in Chapter 12 of the Draft Safety Evaluation Report. (Summary issued March 23, 1984)
March 1, 1984	Letter from applicant transmitting Amendment 6 to the ER-OL consisting of revised seismic risk assessment.
March 1, 1984	Letter from applicant concerning meeting with staff.
March 1, 1984	Letter to applicant concerning removal of proprietary status of Weston Geophysical Corporation, "Seismology and Geological Studies: Miramichi Area, New Brunswick and Central New Hampshire," Volumes I and II, August 1983.
March 6, 1984	Letter from Stone & Webster concerning Topical Report No. 7703, "Missile Barrier Interaction."
March 9, 1984	Letter to applicant requesting additional information on geosciences items.

March 9, 1984	Letter from applicant transmitting Amendment 7 to the FSAR consisting of responses to selected requests for additional information.
March 12, 1984	Letter from applicant concerning comments on policy and planning guidance for 1984.
March 13, 1984	Letter from applicant concerning probabilistic safety study - seismic fragilities.
March 13, 1984	Representatives from NRC and Northeast Utilities meet in Bethesda, Maryland, to discuss applicant's responses to instrumentation and control systems unresolved items contained in draft SER, Section 7. (Summary issued April 2, 1984)
March 13, 1984	Letter from applicant transmitting the piping and instrumentation diagrams related to Question 210.42.
March 14, 1984	Representatives from NRC and Northeast Utilities meet in Bethesda, Maryland, to discuss applicant's responses to auxiliary systems unresolved items contained in SER. (Summary issued March 27, 1984)
March 16, 1984	Letter from applicant advising that information concerning any deviations from Supplement 1 to NUREG-0737 will be provided by May 1, 1984.
March 16, 1984	Letter from applicant concerning response to Generic Letter 83-28, Generic Implications of Salem ATWS Events.
March 16, 1984	Letter from applicant transmitting a response to request for additional information on noise.
March 20, 1984	Letter from applicant transmitting preservice inspection plan (Document No. PS12.01, Revision 2).
March 20, 1984	Letter from applicant concerning status of draft SER open items on licensee qualification.
March 20, 1984	Letter from applicant transmitting responses to 480-series questions.
March 21, 1984	Representatives from NRC and Northeast Utilities meet in Bethesda, Maryland, to discuss applicant's responses to procedures and systems unresolved items contained in the draft SER. (Summary issued April 19, 1984)
March 22, 1984	Letter from applicant transmitting response to Enclosure 4, Item 1, and draft SER open item; Q-List.
March 23, 1984	Letter from applicant concerning meeting with staff on radiological assessment.

March 23, 1984	Letter from applicant concerning corrections to Section 18 of the draft SER.
March 23, 1984	Letter from applicant concerning meeting summary of NRC's structural audit transmittal of responses to confirmatory items.
March 27, 1984	Letter from applicant transmitting responses to requests for additional information and draft SER open items (geotechnical issues).
March 27, 1984	Letter from applicant concerning March 14, 1984, meeting with staff on auxiliary systems.
March 28, 1984	Representatives from NRC and Northeast Utilities meet in Bethesda, Maryland, to discuss applicant's responses to unresolved fire protection items contained in Section 9.5.1 of the draft SER. (Summary issued May 14, 1984)
March 30, 1984	Letter from applicant transmitting Amendment 2 to the probabilistic safety study.
April 2, 1984	Letter from applicant concerning March 13, 1984, meeting with staff on instrumentation and control systems.
April 5, 1984	Letter from applicant transmitting Amendment 7 to the Environmental Report.
April 6, 1984	Letter from applicant submitting revised responses to staff questions on power systems.
April 9, 1984	Representatives from NRC and Northeast Utilities meet in Bethesda, Maryland, to discuss applicant's responses to unresolved items concerning Millstone Unit 3 safe shutdown and alternate shutdown capability. (Summary issued May 15, 1984)
April 9, 1984	Letter from applicant concerning probabilistic safety study - seismic fragilities.
April 9, 1984	Letter from applicant transmitting responses to draft SER open item on chemical engineering.
April 9, 1984	Letter from applicant concerning alternative pipe break criteria.
April 9, 1984	Letter from applicant transmitting responses to structural audit items.
April 9, 1984	Letter from applicant concerning N-1 loop operation.
April 11, 1984	Letter from applicant transmitting a response to Question 420.5 and draft SER open item ICSB-20 (#166).

April 12, 1984	Letter from applicant submitting responses to draft SER open items on effluent treatment systems.
April 13, 1984	Letter from applicant transmitting draft of modified amended security plan including Millstone Unit 3 physical security features.
April 13, 1984	Letter from applicant transmitting a response to draft SER open item CPB-8.
April 13, 1984	Letter from applicant transmitting a response to request for additional information and draft SER open item SEB-23, lateral earth pressure coefficient.
April 13, 1984	Letter from applicant concerning open item CPB-7.
April 18, 1984	Representatives from NRC and Northeast Utilities meet in Bethesda, Maryland, to discuss geological and seismological studies performed by Weston Geophysical Corporation for Northeast Utilities in response to NRC Questions 230.6 and 230.7 concerning the 1982 New Brunswick earthquake and its relationship to Millstone Unit 3.
April 18, 1984	Letter from applicant transmitting a response to draft SER open items on materials engineering.
April 19, 1984	Letter from applicant transmitting responses to draft SER open items.
April 19, 1984	Letter from applicant transmitting a response to draft SER open items on chemical engineering.
April 19, 1984	Letter from applicant transmitting a response to draft SER open items on materials engineering.
April 19, 1984	Letter from applicant concerning environmental functions and processes study.
April 25, 1984	Letter from applicant transmitting a response to draft SER open items on materials engineering.
April 25, 1984	Letter from applicant transmitting the 1983 Annual Financial Report.
April 25, 1984	Representatives from NRC and Northeast Utilities meet in Bethesda, Maryland, to discuss applicant's responses to 430-Series questions in support of the staff's safety evaluation. (Summary issued May 17, 1984)
May 2, 1984	Letter from applicant concerning conference call on containment systems.
May 3, 1984	Letter from applicant transmitting a response to open item CPB-11.

May 3, 1984	Letter from applicant concerning probabilistic safety study - success criteria for large loss-of-coolant accident.
May 3, 1984	Letter from applicant concerning staff Question 491.1.
May 3, 1984	Letter from applicant transmitting a response to draft SER open item on materials engineering.
May 3, 1984	Letter from applicant concerning responses to draft SER open items on environmental and hydrologic engineering.
May 4, 1984	Letter from applicant concerning NRC responses to draft SER and open items on structural and geotechnical engineering.
May 4, 1984	Letter from applicant concerning responses to draft SER open items ICSB-1, -3, -6, -7, -19, and -21.
May 7, 1984	Letter from applicant transmitting responses to Questions 480.19 and 480.34.
May 8, 1984	Letter from applicant transmitting the revised Emergency Plan.
May 8, 1984	Letter from applicant transmitting summary/submittal of revised responses to questions on power systems.
May 9, 1984	Representatives from NRC and Northeast Utilities meet in Bethesda, Maryland, to discuss applicant's responses to structural audit unresolved items. (Summary issued May 31, 1984)
May 9, 1984	Representatives from NRC and Northeast Utilities meet in Bethesda, Maryland, to discuss applicant's responses to safety evaluation unresolved items. (Summary issued May 22, 1984)
May 9, 1984	Representatives from NRC and Northeast Utilities meet in Bethesda, Maryland, to discuss applicant's responses to unresolved items in draft SER Section 3.10. (Summary issued May 15, 1984)
May 9, 1984	Letter to Westinghouse withholding from public disclosure the response to Question 492.4 - CAW-83-65.
May 9, 1984	Letter from applicant concerning probabilistic safety study - refueling water storage tank depletion and core uncovery.
May 9, 1984	Letter from applicant transmitting a response to draft SER open item on materials engineering.

May 10, 1984	Letter from applicant concerning response to draft SER open item on chemical engineering.
May 11, 1984	Letter from applicant transmitting a response to draft SER open item CPB-9.
May 11, 1984	Letter from applicant transmitting responses to draft SER open items ASB-9, -16, -20, and -18.
May 11, 1984	Letter from applicant transmitting a response to confirmatory item CPB-12.
May 14, 1984	Representatives from NRC and Northeast Utilities meet in Bethesda, Maryland, to discuss applicant's responses to unresolved items contained in the draft SER. (Summary issued June 12, 1984)
May 15, 1984	Letter from applicant transmitting a response to draft SER open items ASB-10 and ASB-11.
May 15, 1984	Letter from applicant concerning Millstone Unit 3 meeting held on May 4, 1984.
May 15, 1984	Letter from applicant concerning NRC transmittal of revised responses to draft SER open items on procedures and systems review.
May 15, 1984	Letter from applicant transmitting a response to draft SER open items on auxiliary systems.
May 15, 1984	Letter from applicant transmitting responses to open items.
May 15, 1984	Letter from applicant concerning staff review meeting.
May 15, 1984	Letter from applicant transmitting a response to Question 492.5 on core performance.
May 15, 1984	Letter from applicant concerning revised responses to Questions 471.1 and 471.6on radiological assessment.
May 15, 1984	Letter from applicant concerning staff review meeting on May 9, 1984.
May 15, 1984	Letter from applicant transmitting a response to (1) request for additional information and (2) draft SER open items on structural and geotechnical engineering.
May 17, 1984	Letter from applicant transmitting summary/submittal of revised responses to questions on power systems.
May 18, 1984	Letter from applicant transmitting a response to Question 492.6 and draft SER open items CPB-3 and ICSB-14.

May 22, 1984	Letter from applicant transmitting summary/submittal of responses to draft SER open items.
May 22, 1984	Letter from applicant transmitting revised response to Questions 210.31, 210.34, 210.36, 210.37, 210.44 and 210.45 on mechanical engineering.
May 23, 1984	Letter from applicant concerning staff meeting summary.
May 25, 1984	Letter from applicant concerning probabilistic safety study Level 3 review board.
May 25, 1984	Letter to applicant requesting additional information that may be included in the SER, Supplement 1, to be issued on November 15, 1984.
May 31, 1984	Representatives from NRC and Northeast Utilities meet in Bethesda, Maryland, to discuss applicant's responses to unresolved items contained in draft SER. (Summary issued June 12, 1984)
May 31, 1984	Letter from applicant concerning staff request for additional information in response to confirmatory items.
May 31, 1984	Letter to applicant concerning draft SER.
June 4, 1984	Letter from applicant transmitting Amendment 8 to the FSAR and response to selected requests for additional information.
June 7, 1984	Letter from applicant concerning revision to NRC service list.
June 8, 1984	Representatives from NRC, Northeast Utilities, and Stone & Webster meet in Bethesda, Maryland, to discuss applicant's responses to unresolved items contained in draft SER. (Summary issued June 19, 1984)
June 8, 1984	Letter from applicant transmitting a request for acceptance of a new code case and a revised code case.
June 8, 1984	Representatives from NRC and Northeast Utilities meet in Bethesda, Maryland, to discuss applicant's responses to unresolved items contained in draft SER.
June 11, 1984	Representatives from NRC and Northeast Utilities meet in Bethesda, Maryland, to hear applicant's comments on Lawrence Livermore National Laboratory Interim Report.
June 12, 1984	Representatives from NRC and Northeast Utilities meet in Bethesda, Maryland, to discuss Appendix R separation inside containment.

June 13, 1984	Representatives from NRC and Northeast Utilities meet in Waterford, Connecticut, to observe certain plant features in support of the staff safety evaluation.
June 13, 1984	Representatives from NRC and Northeast Utilities meet in Bethesda, Maryland, to discuss applicant's responses to unresolved items contained in draft SER.
June 14 and 15, 1984	Representatives from NRC and Northeast Utilities meet in Bethesda, Maryland, to discuss tornado missiles, ultimate capacity of containment, overpressurization of pressurizer cubicle, and amplified response spectrum.
June 19, 1984	Letter to applicant requesting additional information on probabilistic safety study.
June 21, 1984	Representatives from NRC and Northeast Utilities meet in Bethesda, Maryland, to discuss status of Technical Specifications for Millstone Unit 3 being prepared by Northeast Utilities.
June 22, 1984	Representatives from NRC and Northeast Utilities meet in Berlin, Connecticut, to discuss lessons learned by the applicant as a consequence of performing the probabilistic safety study for Millstone Unit 3.

APPENDIX B

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APPENDIX C

NUCLEAR REGULATORY COMMISSION
UNRESOLVED SAFETY ISSUES

C.1 Introduction

The NRC staff evaluates the safety requirements used in its reviews against new information as it becomes available. Information related to the safety of nuclear power plants comes from a variety of sources including experience from operating reactors; research results; NRC staff and Advisory Committee on Reactor Safeguards (ACRS) safety reviews; and vendor, architect/engineer, and utility design reviews. After the accident at Three Mile Island (TMI) the Office for Analysis and Evaluation of Operational Data was established to provide a systematic and continuing review of operating experience. Each time a new concern or safety issue is identified from one or more of these sources, the need for immediate action to ensure safe operation is assessed. This assessment includes consideration of the generic implications of the issue.

In some cases, immediate action is taken to ensure safety, for example, the derating of boiling water reactors as a result of the channel box wear problems in 1975. In other cases, interim measures, such as modifications to operating procedures, may be sufficient to allow further study of the issue before licensing decisions are made. In most cases, however, the initial assessment indicates that immediate licensing actions or changes in licensing criteria are not necessary. If the issue applies to several or a class of plants the issue is evaluated further as a "generic safety issue." This evaluation considers the safety significance of the issues, the cost to implement any changes in plant design or operation, and other significant and relevant factors to establish a priority ranking of the issue. On the basis of this ranking, resolution of the issue is scheduled for near-term resolution, deferred until resources become available, or dropped from further consideration.

The issues with the highest priority ranking are reviewed to determine whether they should be designated as "unresolved safety issues" (NUREG-0410, "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants," dated January 1, 1978). However, as discussed above, such issues are considered on a generic basis only after the staff has made an initial determination that the safety significance of the issue does not prohibit continued operation or require licensing actions while the longer term generic review is under way.

These longer term generic studies were the subject of a Decision by the Atomic Safety and Licensing Appeal Board of the Nuclear Regulatory Commission. The Decision was issued on November 23, 1977 (ALAB-444) in connection with the Appeal Board's consideration of the Gulf States Utility Company application for the River Bend Station, Units 1 and 2. These issues were also considered in the operating license proceeding, "Virginia Electric and Power Company (North Anna Nuclear Power Station, Unit Nos. 1 and 2)," ALAB-491, issued August 25, 1978). A further discussion of these issues is contained in a decision by the Atomic Safety and Licensing Appeal Board in connection with its considerations of the Pacific Gas and Electric Company operating license application for the Diablo Canyon Nuclear Power Plant, Units 1 and 2 (ALAB-728, issued May 18, 1983). In the ALAB-728 Decision, the Board stated with regard to an operating license proceeding that "it would be helpful to us if the staff would include in an SER supplement an explanation of the unresolved safety issues affecting the facility under review and the reasons the facility could nonetheless safely operate pending resolution of those issues." This appendix is provided in response to the Board's request.

C.2 Unresolved Safety Issues

In a related matter, as a result of Congressional action on the Nuclear Regulatory Commission budget for Fiscal Year 1978, the Energy Reorganization Act of 1974 was amended (PL 95-209) on December 13, 1977 to include, among other things, a new Section 210 as follows:

UNRESOLVED SAFETY ISSUES PLAN

> SEC. 210. The Commission shall develop a plan providing for specification and analysis of unresolved safety issues relating to nuclear reactors and shall take such action as may be necessary to implement corrective measures with respect to such issues. Such plan shall be submitted to the Congress on or before January 1, 1978 and progress reports shall be included in the annual report of the Commission thereafter.

The Joint Explanatory Statement of the House-Senate Conference Committee for the Fiscal Year 1978 Appropriations Bill (Bill S. 1131) provided the following additional information regarding the Committee's deliberations on this portion of the bill:

SECTION 3 - UNRESOLVED SAFETY ISSUES

> The House amendment required development of a plan to resolve generic safety issues. The conferees agreed to a requirement that the plan be submitted to the Congress on or before January 1, 1978. The conferees also expressed the intent that this plan should identify and describe those safety issues, relating to nuclear power reactors, which are unresolved on the date of enactment. It should set forth: (1) Commission actions taken directly or indirectly to develop and implement corrective measures; (2) further actions planned concerning such measures; and (3) timetables and cost estimates of such actions. The Commission should indicate the priority it has assigned to each issue, and the basis on which priorities have been assigned.

In response to the reporting requirements of the new Section 210, the NRC staff submitted NUREG-0410 to Congress on January 1, 1978. This NUREG describes the NRC generic issues program. The NRC program was already in place when PL 95-209 was enacted and is of considerably broader scope than the unresolved safety issues plan required by Section 210. In the letter transmitting NUREG-0410 to the Congress on December 30, 1977, the Commission indicated that "the progress reports, which are required by Section 210 to be included in future NRC annual reports, may be more useful to Congress if they focus on the specific Section 210 safety items."

It is the NRC's view that the intent of Section 210 was to ensure that plans were developed and implemented on issues with potentially significant public safety implications. In 1978, the NRC undertook a review of more than 130 generic issues addressed in the NRC program to determine which issues fit this description and qualify as unresolved safety issues for reporting to the Congress. The NRC review included the development of proposals by the NRC staff and review and final approval by the NRC Commissioners.

The review is described in a report, NUREG-0510, "Identification of Unresolved Safety Issues Relating to Nuclear Power Plants - A Report to Congress," dated January 1979. The report provides the following definition of an unresolved safety issue:

> An Unresolved Safety Issue is a matter affecting a number of nuclear power plants that poses important questions concerning the adequacy of existing safety requirements for which a final resolution has not yet been developed that involves conditions not likely to be acceptable over the lifetime of the plants it affects.

Further, the report indicates that, in applying this definition, matters that pose "important questions concerning the adequacy of existing safety requirements" were judged to be those for which resolution is necessary to (1) compensate for a possible major reduction in the degree of protection of the public health and safety or (2) provide a potentially significant decrease in the risk to the public health and safety. Quite simply, an unresolved safety issue is potentially significant from a public safety standpoint and its resolution is likely to result in NRC action on the affected plants.

All of the issues addressed in the NRC program were systematically evaluated against this definition as described in NUREG-0510. As a result, 17 unresolved safety issues addressed by 22 tasks in the NRC program were identified.

An in-depth and systematic review of generic safety concerns identified between January 1979 and March 1981 was performed by the staff to determine if any of these issues should be designated as unresolved safety issues. The candidate issues originated from concerns identified in NUREG-0660, "NRC Action Plan as a Result of the TMI-2 Accident"; from ACRS recommendations; from abnormal occurrence reports; and from other operating experience. The staff's proposed list was reviewed and commented on by the ACRS, the Office of Analysis and Evaluation of Operational Data (AEOD), and the Office of Policy Evaluation. The ACRS and AEOD also proposed that several additional unresolved safety issues be considered by the Commission. The Commission considered the above information and approved the four Unresolved Safety Issues (USIs) A-45 through A-48. A description of the review process for candidate issues, together with a list of the issues considered, is presented in NUREG-0705, dated March 1981. An expanded discussion of each of the new unresolved safety issues is also in NUREG-0705. In addition to the four issues identified above, in December 1981 the Commission approved another issue, A-49, "Pressurized Thermal Shock," as an unresolved safety issue.

The issues are listed below. The number(s) of the generic task(s) (e.g., A-1) in the NRC program addressing each issue is indicated in parentheses following the title.

(1) Waterhammer (A-1)
(2) Asymmetric Blowdown Loads on the Reactor Coolant System (A-2)
(3) Pressurized Water Reactor Steam Generator Tube Integrity (A-3, A-4, A-5)
(4) BWR Mark I and Mark II Pressure Suppression Containments (A-6, A-7, A-8, A-39)
(5) Anticipated Transients Without Scram (A-9)
(6) BWR Nozzle Cracking (A-10)
(7) Reactor Vessel Materials Toughness (A-11)
(8) Fracture Toughness of Steam Generator and Reactor Coolant Pump Supports (A-12)
(9) Systems Interaction in Nuclear Power Plants (A-17)
(10) Environmental Qualification of Safety-Related Electrical Equipment (A-24)
(11) Reactor Vessel Pressure Transient Protection (A-26)
(12) Residual Heat Removal Requirements (A-31)
(13) Control of Heavy Loads Near Spent Fuel (A-36)
(14) Seismic Design Criteria (A-40)
(15) Pipe Cracks at Boiling Water Reactors (A-42)
(16) Containment Emergency Sump Reliability (A-43)
(17) Station Blackout (A-44)
(18) Shutdown Decay Heat Removal Requirements (A-45)
(19) Seismic Qualification of Equipment in Operating Plants (A-46)
(20) Safety Implications of Control Systems (A-47)
(21) Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment (A-48)
(22) Pressurized Thermal Shock (A-49)

Of the 22 tasks identified with the unresolved safety issues, 10 are not applicable to Millstone Unit 3, and 6 of these 10 tasks (A-6, A-7, A-8, A-10, A-39, and A-42) are peculiar to boiling water reactors (BWRs). Tasks A-4 and A-5 address steam generator tube problems in Combustion Engineering and Babcock and Wilcox (B&W) plants. A-46 deals with seismic qualification of equipment in operating plants and does not apply to Millstone Unit 3. Millstone Unit 3 was designed on the basis of current seismic design criteria, and commitments for seismic equipment qualification are in accordance with the latest codes and standards (see Sections 3.9.2 and 3.10 of this SER). Also, Task A-48 is related to pressurized water reactor (PWR) plants with ice condenser containments or BWRs with pressure-suppression-type containments. With regard to the remaining tasks that are applicable to this facility, the NRC staff has issued NUREG reports providing its proposed resolution of eight of these issues (Table C.1). Each of these has been addressed in this Safety Evaluation Report (SER) or will be addressed in a future supplement. Table C.1 lists those issues and the section of this SER in which they are discussed.

The remaining issues applicable to this facility are

(1) Westinghouse Steam Generator Tube Integrity (A-3)
(2) Systems Interaction in Nuclear Power Plants (A-17)
(3) Seismic Design Criteria (A-40)
(4) Containment Emergency Sump Reliability (A-43)
(5) Station Blackout (A-44)
(6) Shutdown Decay Heat Removal Requirements (A-45)
(7) Safety Implications of Control Systems (A-47)
(8) Pressurized Thermal Shock (A-49)

The task action plans for the generic tasks up to and including A-40 above are included in NUREG-0649, "Task Action Plans for Unresolved Safety Issues Related to Nuclear Power Plants." Task action plans for later tasks were issued individually as indicated in Table C.2.

Each task action plan provides a description of the problem; the staff's approach to its resolution; a general discussion of the bases on which continued plant licensing or operation can proceed pending completion of the task; the technical organizations involved in the task and estimates of the manpower power required; a description of the interactions with other NRC offices, the ACRS, and outside organizations; estimates of funding required for contractor-supplied technical assistance; prospective dates for completing the task; and a description of potential problems that could alter the planned approach or schedule.

In addition to the task action plans, the staff issues the "Office of Nuclear Reactor Regulation Unresolved Safety Issues Summary, Aqua Book" (NUREG-0606) on a quarterly basis; this report provides current schedule information for each of the unresolved safety issues. It also includes information relative to the implementation status of each unresolved safety issue for which technical resolution is complete.

The staff has reviewed the unresolved safety issues listed above as they relate to Millstone Unit 3. Discussion of each of these issues, including references to related discussions in the SER, is in Section C.3. On the basis of its review, the staff concludes for the reasons set forth in Section C.3 that there is reasonable assurance that Millstone Unit 3 can be operated before the ultimate resolution of these generic issues without endangering the health and safety of the public. Tasks A-43, A-45, and A-47 are accepted subject to the resolution of those confirmatory items identified in Section C.3. The resolution of these items will be reported in a supplement to the SER.

C.3 Discussions of Unresolved Safety Issues as They Relate to Millstone Unit 3

This section provides the NRC staff's evaluation of Millstone Unit 3 for each of the applicable unresolved safety issues. This includes the staff's bases for licensing before ultimate resolution of these issues.

A-3 Steam Generator Tube Integrity

The primary concern is the capability of steam generator tubes to maintain their integrity during normal operation and postulated accident conditions.

Westinghouse steam generators have experienced tube degradation in several forms. These are wastage, intergranular attack, stress corrosion cracking, and denting. Each of these forms of degradation is discussed below, and specific measures to prevent their occurrence at Millstone Unit 3 are included:

(1) Wastage is characterized by general loss of metal from the tube wall as a result of a chemical corrosive reaction. Wastage has occurred only in steam generators that used sodium phosphate as a chemical additive. The Millstone Unit 3 steam generators will use a water treatment consisting of

hydrazine and ammonium hydroxide (this is called all volatile treatment or AVT). Wastage has not been observed in steam generators using all volatile chemistry control.

(2) Intergranular attack is a chemical reaction wherein the grain boundaries of the Inconel 600 tubes are attacked by chemical solutions. Significant intergranular attack has occurred only in steam generators that have an open crevice in the tube to tubesheet area. In the Millstone Unit 3 steam generators, there is no open crevice in the tubesheet area, and hence intergranular attack should be eliminated.

(3) Stress corrosion cracking (SCC) refers to intergranular cracking of stressed tubes, without reference to a causative chemical agent. This term is used either to encompass a number of known SCC mechanisms or when the chemical causing the corrosion is not known. SCC resistance of Millstone Unit 3 steam generator tubes has been improved by a special thermal treatment. Primary-side SCC has also occurred in a number of Westinghouse steam generators in the narrow-radius U-bend area of the tubes in the bundle interior. The Inconel tubing of the inner 8 row has received a stress relief heat treatment that has demonstrated improved resistance to primary-side SCC.

(4) Denting is the most serious degradation problem encountered in Westinghouse steam generators. It is caused by rapid corrosion of the tube support plates at the holes where the tubes pass through the support plates. For Millstone Unit 3, the tube support plates were manufactured from ferritic stainless steel material, which has been shown in laboratory tests to be corrosion resistant to the operating environment. The tube support plates were designed and manufactured with broached holes rather than drilled holes. The broached-hole design promotes high velocity flow along the tubes thereby sweeping impurities away from the support plate locations.

In addition, Millstone Unit 3 will use full-flow demineralizers and the secondary system will be operated within improved guidelines for chemistry control. Also, modified baffling is used to improve the circulation ratio; thus, sludge deposition and consequent tube dryout and corrosion are avoided.

Pending completion of Task A-3, the measures taken at this facility should minimize the steam generator tube problems encountered. Further, the inservice inspection and Technical Specification requirements will ensure that the applicant and the NRC staff are alerted to tube degradation should it occur. Appropriate actions such as tube plugging, increased and more frequent inspections, and power derating could be taken if necessary. Because the improvements that will result from Task A-3 are expected to be procedural (i.e., improved inspection of the steam generators), they can be implemented by the applicant after operation of this facility begins, if necessary.

On the basis of the foregoing, the staff has concluded that Millstone Unit 3 can be operated before final resolution of this generic issue without undue risk to the health and safety of the public.

A-17 <u>Systems Interaction in Nuclear Power Plants</u>

The staff's systems interaction program was initiated in May 1978 with the definition of A-17 and was intensified by TMI Action Plan (NUREG-0660)

Item II.C.3, "Systems Interaction." The concern arises because the design, analysis, and installation of systems are frequently the responsibility of teams of engineers with functional specialties such as civil, electrical, mechanical, or nuclear. Experience at operating plants has led to questions of whether the work of these functional specialists is sufficiently integrated to enable them to minimize adverse interactions among systems. Some adverse events that occurred in the past might have been prevented if the teams had ensured the necessary independence of safety systems under all conditions of operation.

The applicant has not described a comprehensive program that separately evaluates all structures, systems, and components important to safety for the three categories of adverse systems interactions: (1) spatially coupled, (2) functionally coupled, (3) and humanly coupled. However, the plant has been evaluated against current licensing requirements that are founded on the principle of defense-in-depth. Adherence to this principle and conformance to the regulations (e.g., general design criteria) result in requirements such as physical separation and independence of redundant safety systems as well as protection against hazards such as high-energy-line ruptures, missiles, high winds, flooding, seismic events, and fires. These design provisions are subject to review against the Standard Review Plan (NUREG-0800), which requires interdisciplinary reviews of safety-grade equipment and addresses different types of potential systems interactions. Also, the quality assurance program that is followed during the design, construction, and operational phases for each plant contributes to the prevention of introducing adverse systems interactions.

The NRC staff's current review procedures assign primary responsibility for review of various technical areas to specific organizational units and secondary responsibility to other units where there is a functional interface. Designers follow somewhat similar procedures and provide the analyses of systems and interface reviews. Under Task A-17 methods that could identify adverse systems interactions that were not uncovered by current review procedures are being investigated.

The development of systematic ways to identify, rank, and evaluate systems interactions could further reduce the likelihood of intersystem failures resulting in the loss of plant safety functions. A comprehensive program may use analytical methods, visual inspections, and experience feedback to identify hidden dependencies. On the basis of the foregoing discussion, the staff concludes that there is reasonable assurance that Millstone Unit 3 can be operated pending ultimate resolution of this generic issue without endangering the health and safety of the public.

A-40 <u>Seismic Design Criteria - Short-Term Program</u>

NRC regulations require that nuclear power plant structures, systems, and components important to safety be designed to withstand the effects of natural phenomena such as earthquakes. Detailed requirements and guidance regarding the seismic design of nuclear plants are provided in the NRC regulations and in regulatory guides issued by the Commission. However, there are a number of plants with construction permits and operating licenses issued before the NRC's current regulations and regulatory guidance were in place. For this reason, reviews of the seismic design of various plants are being undertaken to ensure that these plants do not present an undue risk to the public. Task A-40 is, in effect, a compendium of short-term efforts to support such reevaluation efforts of the NRC staff, especially those related to older operating plants. In addition, some revisions to sections of the Standard Review Plan (SRP) and regulatory guides to bring them more in line with the state of the art will result.

Safety-related structures, systems, and components for Millstone Unit 3 are designed to withstand the effects of earthquakes in accordance with current NRC regulations, regulatory guides, and the Standard Review Plan, as discussed in Sections 3.7, 3.9, and 3.10 of the FSAR. Specifically, the five subjects identified in the NRC's issue description for Task A-40, that is, magnitude of earthquakes (safe shutdown earthquake (SSE)), free-field motion (SSE), soil-structures interactions, motion of plant equipment, and load combination are discussed therein. Design of structures for protection against natural phenomena such as earthquakes is described in FSAR Section 3.8. Should the resolution of A-40 indicate that a change is needed in these licensing requirements, all operating reactors, including Millstone Unit 3, will be reevaluated on a case-by-case basis.

Accordingly, the staff concludes that there is reasonable assurance that Millstone Unit 3 can be operated before ultimate resolution of this generic issue without endangering the health and safety of the public.

A-43 <u>Containment Emergency Sump Reliability</u>

Following a postulated loss-of-coolant accident (LOCA), water would be collected in the containment emergency sump for use in the long-term recirculation mode, thus maintaining core cooling. This water could also be circulated through the containment spray cooling system for removal of heat and fission products within containment. The principal safety concern is loss of the ability to draw water from the containment emergency sump under post-LOCA conditions, thus leading to the degradation of, or disability of, the long-term recirculation safety train and impairment of decay heat removal.

Two major concerns have been postulated: (1) adverse hydraulic conditions in the sump (e.g., air ingestion, break flow effects, and vortex formation) thereby leading to loss of residual heat removal pumping and (2) severe sump screen blockages resulting from LOCA-generated insulation debris, which could cause loss of net positive suction head requirements.

The evaluation of such safety concerns has been carried out, and the technical findings are reported in NUREG-0897. The result has been a recommended revision to Regulatory Guide (RG) 1.82 that reflects these findings. The destruction of plant insulation by the LOCA jet is viewed as a potential safety concern relative to screen blockage. The evaluation of debris blockage is a plant-specific requirement resulting from design differences and types of insulation used. Air ingestion and vortex formation are not as serious as previously hypothesized. NUREG-0897, NUREG-0869 (which includes the proposed RG 1.82, Rev. 1), and SRP Section 6.2.2 were issued for public comment in May 1983. The requirements that may result from A-43 are expected to be primarily procedural (i.e., an assessment of sump blockage following a posulated LOCA). Plant modifications, if necessary, can be implemented after operation of the facility begins.

The staff has reviewed the applicant's sump design; the results of this review are documented in Section 6.2.2 of this SER. As noted in Section 6.2.2, the staff is reviewing additional information to ensure that the Millstone sump design will meet all current design criteria.

On the basis of the above, the staff concludes that, subject to the satisfactory resolution of the confirmatory items identifed in Section 6.2.2, there is reasonable assurance that Millstone Unit 3 can be operated before the ultimate resolution of this generic issue without undue risk to the health and safety of the public.

A-44 Station Blackout

Electrical power for safety systems at nuclear power plants must be supplied by at least two redundant and independent divisions. The systems used to remove decay heat to cool the reactor core following a reactor shutdown are included among the safety systems that must meet these requirements. Each electrical division for safety systems includes two offsite alternating current (ac) power connections, a standby emergency diesel generator ac power supply, and direct current sources.

Task A-44 involves a study of whether or not nuclear power plants should be designed to accommodate a complete loss of all ac power, that is, a loss of both the offsite and the emergency diesel generator ac power supplies. This issue arose because of operating experience regarding the reliability of ac power supplies. A number of operating plants have experienced a total loss of offsite electrical power, and more occurrences are expected in the future. In almost every one of these loss-of-offsite-power events, the onsite emergency ac power supplies were available immediately to supply the power needed by vital safety equipment. However, in some instances, one of the redundant emergency power supplies has been unavailable. In a few cases there has been a complete loss of ac power, but during these events, ac power was restored in a short time without serious consequences. In addition, there have been numerous instances of emergency diesel generators failing to start and run in operating plants during periodic surveillance tests.

A loss of all ac power was not a design-basis event for the Millstone Unit 3 facility. Nonetheless, a combination of design, operating, and testing requirements has been imposed to ensure that this unit will have substantial resistance to a loss of all ac source and that, even if a loss of all ac power should occur, there is reasonable assurance the core will be cooled. These design, operating, and testing requirements are discussed below.

A loss of offsite ac power involves a loss of both the preferred and backup sources of offsite power. The staff's review and basis for acceptance of the design, inspection, and testing provisions for the offsite power system are described in Section 8.2 of this SER.

If offsite ac power is lost, the diesel generators and their associated distribution systems will deliver emergency power to safety-related equipment. The staff's review of the design, testing, surveillance, and maintenance provisions for the onsite emergency diesel generators is described in Sections 8.3 and 9.6 of this SER. Staff requirements include preoperational testing to ensure the reliability of the installed diesel generators in accordance with the provisions of RG 1.108. In addition, the applicant has implemented certain recommendations of NUREG/CR-0660 for enhancing diesel generator reliability.

If both offsite and onsite ac power is lost, cooling water can still be provided to the steam generator by the auxiliary feedwater system using a steam turbine-driven pump that does not rely on ac power for operation. The auxiliary feedwater system design and operation is described in SER Section 10.4.9.

In addition to the above, the Commission has determined that some interim measures should be taken at all plants to accommodate a station blackout pending resolution of the issue. Consequently, the NRC requested (Generic Letter 81-04, dated February 25, 1981) a review of plant operation to determine the applicant's capability to mitigate a station blackout event and properly implement, as necessary, emergency procedures and training programs for station blackout events. Appropriate review of the procedures and training programs for station blackout events will be completed before fuel loading. The applicant will use the Westinghouse Owners Group Emergency Response Guidelines for development of emergency operating procedures.

On the basis of the above considerations, the staff concludes that there is reasonable assurance that Millstone Unit 3 can be operated before the ultimate resolution of this generic issue without endangering the health and safety of the public.

A-45 <u>Shutdown Decay Heat Removal Requirements</u>

Under normal operating conditions, power generated within a reactor is removed as steam to produce electricity through a turbine generator. Following a reactor shutdown, a reactor produces insufficient power to operate the turbine; however, the radioactive decay of fission products continues to produce heat (so-called "decay heat"). Therefore, when the reactor is shut down, other measures must be available to remove decay heat from the reactor to ensure that high temperatures and pressures do not develop that could jeopardize the reactor and the reactor coolant system. It is evident, therefore, that all light-water reactors (LWRs) share two common decay-heat-removal functional requirements (1) to provide a means of transferring decay heat from the reactor coolant system to an ultimate heat sink and (2) to maintain sufficient water inventory inside the reactor vessel to ensure adequate cooling of the reactor fuel. The reliability of a particular power plant to perform these functions depends on the frequency of initiating events that require or jeopardize decay heat removal operations and the probability that required systems will respond to remove the decay heat.

The accident at Three Mile Island, Unit 2 (TMI-2) demonstrated how a relatively common fault, with which the operator should have been able to cope easily, could escalate into a potentially hazardous situation, with severe financial losses to the utility, as a result of difficulties arising in the decay heat removal (DHR) process.

Other circumstances, of a more unusual nature (e.g., damage to systems by external events such as floods or earthquakes, or by sabotage), which could make removal of the decay heat difficult, can also be foreseen.

The question arises, therefore, whether current licensing design requirements are adequate to ensure that LWRs do not pose unacceptable risk as a result of a failure to remove shutdown decay heat, and whether, at a cost commensurate with the increase in safety that could be achieved, improvements could be made in the effectiveness of shutdown decay heat removal in one or more transient or accident situations. Resolution of this question is considered to be of sufficient importance to merit raising it to the status of an unresolved safety issue.

To some extent, the effectiveness of the DHR systems is linked to that of the onsite and offsite electrical supplies; the performance and reliability of those supplies is being considered in Task A-44, "Station Blackout." Consequently, the scope of work required in relation to the DHR systems is complementary to Task A-44 above.

The overall purpose of Task A-45 is to evaluate the adequacy of current licensing design requirements to ensure that nuclear power plants do not pose an unacceptable risk because of a failure to remove shutdown decay heat. This will require the development of a comprehensive and consistent set of shutdown cooling requirements for existing and future LWRs, including the study of alternative means of shutdown DHR and of diverse "dedicated" systems for this purpose.

This task will evaluate the benefit of providing alternate means of DHR that could substantially increase the plant's capability to handle a broader spectrum of transients and accidents. The study will include a number of plant-specific DHR systems evaluations and will result in recommendations regarding the desirability of, and possible design requirements for, improvements in existing systems of an alternative DHR method, if the improvements or alternatives can significantly reduce the overall risk to the public in a cost-effective manner.

An integrated systems approach to the problem will be employed. Accordingly, quantitative methods will be used, where possible, to define design requirements for future plants and to measure the effectiveness and acceptability of the shutdown DHR systems in existing plants. The principal means for removing the decay heat in a PWR under normal conditions immediately following reactor shutdown is through the steam generators, using the auxiliary feedwater system. In addition to the WASH-1400 study (NUREG-75/014), later reliability studies and related experience from the TMI-2 accident have reaffirmed that the loss of capability to remove heat through the steam generator is a significant contributor to the probability of a core-melt event. The staff's review of the auxiliary feedwater AFW system design and operation is described in Section 10.4.9 of this SER.

It should be noted, as discussed below, that the NRC required licensees to implement many improvements to the steam generator AFW system following the TMI-2 accident. However, the staff still believes that providing an alternative means of decay heat removal could substantially increase the plant's capability to deal with a broader spectrum of transients and accidents and potentially could, therefore, significantly reduce the overall risk to the public. Consequently, this task will investigate alternative means of decay heat removal in PWR plants, including but not limited to, the use of existing equipment where possible. This study will include a representative sample of plant-specific DHR system evaluations. It will result in recommendations regarding the adequacy of existing DHR requirements and the desirability of, and possible design requirements for, an alternative DHR method, other than that normally associated with the steam generator and secondary coolant system.

The AFW system is a very important safety system in a PWR in terms of providing a heat sink via the steam generators to remove core decay heat. As mentioned above, the TMI-2 accident and subsequent studies have further highlighted the importance of the AFW systems. As discussed below, the NRC staff has required certain upgrading of the AFW systems for all LWRs following the TMI-2 accident. Although this task will investigate alternative means of decay heat removal, the NRC staff concludes that in general (not on a plant-specific basis) if the licensees comply with the upgrading of requirements for the AFW system, the action taken following the TMI-2 accident justifies continued operation and licensing pending completion of this task. Further discussion and the bases for this view are provided below.

° TMI-2 Accident

The accident at TMI-2 on March 28, 1979, involved a main feedwater transient coupled with a stuck-open pressurizer power-operated relief valve and a temporary failure of the AFW system, and subsequent operator intervention to severely reduce flow from the safety injection system. The resulting severity of the ensuing events and the potential generic aspects of the accident for other operating reactors led the NRC to initiate prompt action to (1) ensure that other reactor licensees, particularly those with plants similar in design to TMI-2, took the necessary action to substantially reduce the likelihood for TMI-2-type events, and (2) investigate the potential generic implications of this action for other operating reactors.

The Bulletins and Orders Task Force (BOTF) was established within the NRC Office of Nuclear Reactor Regulation (NRR) in early May 1979 and completed its work on December 31, 1979. This task force was responsible for reviewing and directing the TMI-2-related staff activities associated with the NRC Office of Inspection and Enforcement (IE) bulletins. Commission orders, and generic evaluations of loss-of-feedwater transients and small-break loss-of-coolant accidents for all operating plants to ensure their continued safe operation. NUREG-0645, "Report of the Bulletins and Orders Task Force," summarizes the results of the work performed.

° Generic and Plant-Specific Studies

For B&W-designed operating reactors, an initial NRC staff study was completed and published as NUREG-0560, "Staff Report on the Generic Assessment of Feedwater Transients in Pressurized Water Reactors Designed by the Babcock & Wilcox Company." This study considered the particular design features and operational history of B&W-designed operating plants in light of the TMI-2 accident and related current licensing requirements. As a result of this study, a number of findings and recommendations resulted that are now being pursued.

Generally, the activities involving the B&W-designed reactors are reflected in the actions specified in the Commission orders. Consequently, a number of actions have been specified regarding transient and small-break analyses, upgrading of AFW reliability and performance, procedures for operator action, and operator training. The results of the NRC staff review of the B&W small-break analysis are published in NUREG-0565, "Generic Evaluation of Small-Break Loss-of-Coolant Accident Behavior in Babcock & Wilcox-Designed Operating Plants."

Similar studies have been completed for operating plants designed by Westinghouse (W), Combustion Engineering (CE), and General Electric (GE). Those studies, which also focus specifically on the predicted plant performance under different accident scenarios involving feedwater transients and small-break LOCAs, are published in NUREG-0611, "Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants and Near-Term Operating License Applications."

On the basis of the review of the operating plants in light of the TMI-2 accident, the NRC staff reached the following conclusions:

(1) The continued operation of the operating plants is acceptable provided that certain actions related to the plants' design and operation and training of operators identifed in NUREG-0645 are implemented, consistent with the recommended implementation schedules.

(2) The actions taken by the licensees with operating plants in response to the IE bulletins (including the actions specified in NUREG-0623, "Generic Assessment of Delayed Reactor Coolant Pump Trip During Small-Break Loss-of-Coolant Accidents in Pressurized Water Reactors") provide added assurance for the protection of the health and safety of the public.

In addition, the BOTF independently confirmed the safety significance of those related actions recommended by other NRR task forces as discussed in NUREG-0645.

° Pressurized Water Reactors

The primary method for removal of decay heat from PWRs is via the steam generators to the secondary system. This energy is transferred on the secondary side to either the main feedwater or auxiliary feedwater systems and is rejected to either the turbine condenser or the atmosphere via the secondary coolant system safety/relief valves. Following the TMI-2 accident, the importance of the AFW system was highlighted and a number of improvements were made to improve the reliability of the AFW system (NUREG-0645). It was also required that operating plants be capable of providing the required AFW flow for at least 2 hours from one AFW pump train independent of any ac power source; that is, if both offsite and onsite ac power sources are lost.

Some PWRs potentially have at least one alternate means of removing decay heat if an extended loss of feedwater is postulated. This method is known as "feed and bleed" and uses the high-pressure injection (HPI) system to add water coolant (feed) at high pressure to the primary system. The decay heat increases the system pressure, and energy is removed through the power-operated relief valves (PORVs) and/or the safety valves (bleed), if necessary. It should be noted that some PWRs incorporate HPI pumps that cannot operate at full system pressure (cutoff head about 1,500 psi). For those cases, the PORVs can be manually opened, thereby reducing the system pressure to within the operating range of the HPI. Limited vendor analyses have shown that the core can be adequately cooled by this means, provided the containment pressure can be controlled to a safe level.

When the primary system is at low pressure, the long-term decay heat is removed by the residual heat removal (RHR) system to achieve and maintain cold shutdown conditions. Task A-45 will also consider the adequacy of reliability and performance criteria and standards for RHR systems. The staff's review of the RHR system design and operation is described in Section 5.4.7 of the SER.

° Conclusion

In summary, because of the upgrading of current DHR systems that was required following the TMI-2 accident, the staff concludes that, in general, plants may continue to be licensed and operated before the ultimate resolution of this generic issue without endangering the health and safety of the public. However, licensee compliance with the upgrading of DHR system requirements must be examined by the staff on an individual case basis. For Millstone Unit 3, the staff is reviewing additional information related to this issue, particularly the confirmatory items identified in SER Section 10.4.9, and will report its findings in a supplement to this SER. Consequently, the staff has concluded that, subject to the satisfactory resolution of the items regarding the AFW system, there is reasonable assurance that Millstone Unit 3 can be operated before ultimate resolution of this generic issue without undue risk to the health and safety of the public.

A-47 Safety Implications of Control Systems

This issue concerns the potential for transients or accidents being made more severe as a result of control system failures or malfunctions. These failures or malfunctions may occur independently or as a result of the accident or transient under consideration. One concern is the potential for a single failure - such as a loss of a power supply, short circuit, open circuit, or sensor failure - to cause simultaneous malfunction of several control features. Such an occurrence could conceivably result in a transient more severe than those transients analyzed as anticipated operational occurrences. A second concern is that a postulated accident could cause control system failures that would make the accident more severe than analyzed. Accidents could conceivably cause control system failures by creating a harsh environment in the area of the control equipment or by physically damaging the control equipment. Although it is generally believed that such control system failures would not lead to serious events or result in conditions that safety systems could not safely handle, indepth studies have not been rigorously performed to verify this belief. The potential for an accident that would affect a particular control system, and effects of the control system failures, may differ from plant to plant. Therefore, it is not possible to develop generic answers to all these concerns; it is possible to develop generic criteria that can be used for future plant-specific reviews. The purpose of this task is to verify the adequacy of existing criteria for control systems or propose additional generic criteria (if necessary) that will be used for plant-specific review.

The Millstone Unit 3 safety systems have been designed with the goal of ensuring that control system failures (either single or multiple) will not prevent automatic or manual initiation and operation of any safety system equipment required to trip the plant or to maintain the plant in a safe shutdown condition following any anticipated operational occurrence or accident. This has been accomplished by either providing independence between safety- and nonsafety-grade systems or providing isolating devices between safety- and nonsafety-grade systems. These devices preclude the propagation of nonsafety-grade system equipment faults so that operation of the safety-grade system equipment is not impaired.

A wide range of bounding transients and accidents is being analyzed to ensure that the postulated events will be adequately mitigated by the safety systems. In addition, systematic reviews of safety systems have been performed with the goal of ensuring that the control system failures (single or multiple) will not defeat safety system action.

Also, the applicant has been requested (NRC Information Notice 79-22, "Qualification of Control Systems," September 17, 1979) (1) to review the possibility of consequential control system failures that could exacerbate the effects of high-energy-line breaks (HELBs) and (2) to adopt new operator procedures, where needed, to ensure that the postulated events would be adequately mitigated. As part of the review, the staff is also evaluating the qualification program to ensure that equipment that may potentially be exposed to HELB environments has been adequately qualified or an adequate basis has been provided for not qualifying the equipment to the limiting hostile environment. The staff's evaluation of the applicant's response to Information Notice 79-22 and the adequacy of the qualification program are reported in Sections 7.7.2.2 and 3.11 of this SER, respectively.

With the recent emphasis on the availability of postaccident instrumentation (RG 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants To Assess Plant Conditions During and Following an Accident"), the staff's reviews evaluate the designs to ensure that control system failures will not deprive the operator of information required to maintain the plant in a safe shutdown condition after any anticipated operational occurrence or accident. The applicant was requested to evaluate the Millstone Unit 3 control systems and identify any control systems whose malfunction could impact plant safety. The applicant has been requested to document the degree of interdependence of these identified control systems and identify the use (if any) of common power supplies and the use of common sensors or common sensor impulse lines whose failure could have potential safety significance. The status of these reviews and the staff's evaluation are in Section 7.7.2.1 of the SER.

In addition, IE Bulletin 79-27 ("Loss of Non-Class 1E Instrumentation and Control Power System Bus During Operation," November 30, 1979) was issued to the applicant requesting that evaluations be performed to ensure the adequacy of plant procedures for accomplishing shutdown on loss of power to any electrical bus supplying power for instruments and controls. The results of this review are in SER Section 7.5.2.1.

The subtask of this issue concerning the steam generator overfill transient in pressurized water reactors is currently under review by the staff. Pending ultimate resolution of this item, the applicant has incorporated in the Millstone Unit 3 design a safety-grade high-level initiation trip signal to trip the main feedwater pumps, the feedwater isolation valves, and the main turbine to prevent the occurrence of overfill transients.

On the basis of these considerations and subject to the satisfactory resolution of the open items identified in Section 3.11, the staff concludes that there is reasonable assurance that Millstone Unit 3 can be operated before the ultimate resolution of this generic issue without endangering the health and safety of the public.

A-49 Pressurized Thermal Shock

The issue of pressurized thermal shock (PTS) arises because in PWRs transients and accidents can occur that result in severe overcooling (thermal shock) of the reactor pressure vessel, concurrent with or followed by repressurization. In these PTS events, rapid cooling of the reactor vessel internal surface results in thermal stress with a maximum tensile stress at the inside surface of the vessel. The magnitude of the thermal stress depends on the temperature profile across the reactor vessel wall as a function of time. The effects of this thermal stress are compounded by pressure stresses.

Severe reactor system overcooling events simultaneous with or followed by pressurization of the reactor vessel (PTS events) can result from a variety of causes. These include system transients, some of which are initiated by instruumentation and control system malfunctions (including stuck-open valves in either the primary or secondary system), and postulated accidents such as small-break LOCAs, main steamline breaks, and feedwater line breaks.

The PTS issue is a concern for PWRs only after the reactor vessel has lost its fracture toughness properties and is embrittled by neutron irradiation. The standards and regulatory requirements to which the Millstone Unit 3 reactor vessel was designed and fabricated are described in FSAR Section 5.3.

As long as the fracture resistance of the reactor vessel material is relatively high, overcooling events are not expected to cause vessel failure. However, the fracture resistance of reactor vessel materials decreases with exposure to fast neutrons during the life of a nuclear power plant. The rate of decrease is dependent on the metallurgical composition of the vessel walls and welds. If the fracture resistance of the vessel has been reduced sufficiently by neutron irradiation, severe overcooling events could cause propagation of small flaws that might exist near the inner surface. The assumed initial flaw might be enlarged into a crack through the vessel wall of sufficient extent to threaten vessel integrity and, therefore, core cooling capability.

For the reactor pressure vessel to fail and constitute a risk to public health and safety, a number of contributing factors must be present. These factors are (1) a reactor vessel flaw of sufficient size to initiate and propagate; (2) a level of irradiation (fluence) and material properties and composition sufficient to cause significant embrittlement (the exact fluence depends on materials present; i.e., high copper content causes embrittlement to occur more rapidly); (3) a severe overcooling transient with repressurization; and (4) the crack resulting from the propagation of initial cracks of such size and location that the vessel fails.

As a result of the evaluation of the PTS issue, the staff recommended to the Commission in SECY-82-465 (November 23, 1982) actions to prevent PTS events in operating reactors. The Commission accepted the staff recommendations and the staff has published a Notice of Proposed Rulemaking (49 FR 4498) for a rule that would establish an RT_{NDT} screening criterion (below which PTS risk is considered acceptable), require early analysis and implementation of such flux reduction programs as a reasonably practicable method to avoid reaching the screening criterion, and require plant-specific PTS safety analyses before plants are within 3 calendar years of reaching the screening criterion including analyses of proposed alternatives to minimize the PTS program.

The staff has published such a rule for public comment (49 FR 4498) and believes that the Millstone Unit 3 plant could easily meet the requirements of the proposed rule. The applicant states that the estimated end-of-life RT_{NDT} for the Millstone Unit 3 vessel is 138°F, which is well below the applicable NRC-staff-proposed screening criterion of 270°F.

On the basis of the above considerations, the staff concludes that there is reasonable assurance that the Millstone Unit 3 facility can be operated before ultimate resolution of this generic issue and completion of the proposed rulemaking without undue risk to the health and safety of the public.

C.5 References

Atomic Safety and Licensing Appeal Board, ALAB-444, 6 NRC 760 "Gulf States Utilities Co., River Bend, Units 1 and 2," November 23, 1977.

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PL 95-209, "NRC Authorizations FY 1978" (91 Stat. 1481), December 13, 1977.

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---, NUREG-0313, "Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping," July 1977.

---, NUREG-0410, "NRC Program for the Resolution of Generic Issues Related to Nuclear Power Plants," January 1978.

---, NUREG-0460, "Anticipated Transients Without Scram for Light Water Reactors," Vol. 4, March 1980.

---, NUREG-0510 "Identification of Unresolved Safety Issues Relating to Nuclear Power Plants - A Report to Congress," January 1979.

---, NUREG-0560, "Safety Report on the Generic Assessment of Feedwater Transients in Pressurized Water Reactors Designed by the Babcock and Wilcox Company," May 1979.

---, NUREG-0565, "Generic Evaluation of Small Break Loss-of-Coolant Accident Behavior in Babcock and Wilcox-Designed Operating Plants," January 1980.

---, NUREG-0577, "Potential for Low Fracture Toughness and Lamellar Tearing in PWR Generator and Reactor Coolant Pump Supports," Rev. 1, October 1983.

---, NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," Rev. 1, July 1981.

---, NUREG-0606, "Office of Nuclear Reactor Regulation Unresolved Safety Issues, Aqua Book," issued quarterly.

---, NUREG-0611, "Generic Evaluation of Feedwater Transients and Small-Break Loss-of-Coolant Accidents in Westinghouse-Designed Operating Plants," January 1980.

---, NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," July 1980.

---, NUREG-0623, "Generic Assessment of Delayed Reactor Coolant Pump Trip During Small-Break Loss-of-Coolant Accidents in Pressurized Water Reactors," November 1979.

---, NUREG-0645, "Report of the Bulletins and Orders Task Force of the Office of Nuclear Reactor Regulation," January 1980.

---, NUREG-0649, "Task Action Plans for Unresolved Safety Issues Related to Nuclear Power Plants," February 1980.

---, NUREG-0660, "NRC Action Plan as a Result of the TMI-2 Accident," May 1980.

---, NUREG-0705, "Identification of New Unresolved Safety Issues Relating to Nuclear Power Plants, Special Report to Congress," March 1981.

---, NUREG-0744, "Resolution of the Task A-11 Reactor Vessel Materials Toughness Safety Issue," Vols. I and II, Rev. 1, October 1982.

---, NUREG-0800, "Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants," July 1981 (includes branch technical positions).

---, NUREG-0869, "USI A-43 Resolution Positions," May 1983.

---, NUREG-0897, "Containment Emergency Sump Performance," May 1983.

---, NUREG/CR-0660, "Enhancement of Onsite Emergency Generator Reliability," February 1979.

---, NUREG-0927, "Evaluation of Water Hammer Occurrence in Nuclear Power Plants," March 1984.

---, SECY-82-465, Subject: "Pressurized Thermal Shock," November 23, 1982.

---, "10 CFR Part 50 - Analysis of Potential; Pressurized Thermal Shock Events - A Proposed Rule," 49 FR 4498, February 7, 1984.

Table C.1 Unresolved safety issues applicable to Millstone Unit 3 addressed in this report

Task no.	NUREG report no. and title	SER section
A-1	NUREG-0927, "Evaluation of Water Hammer Occurrence in Nuclear Power Plants"	10.4.9
A-2	NUREG-0609, "Asymmetric Blowdown Loads on PWR Primary Systems"	3.9
A-9	NUREG-0460, "Anticipated Transients Without Scram for Light Water Reactors," Vol. 4	15.3.8
A-11	NUREG-0744, "Resolution of the Task A-11 Reactor Vessel Materials Toughness Safety Issue," Vols. I and II, Rev. 1	5.3
A-12	NUREG-0577, "Potential for Low Fracture Toughness and Lamellar Tearing in PWR Steam Generator and Reactor Coolant Pump Supports," Rev. 1	5.2
A-24	NUREG-0588, "Interim Staff Position on Environmental Qualification of Safety-Related Electrical Equipment," Rev. 1	3.11
A-31	NUREG-0800, SRP Section 5.4.7 and BTP 5-1, "Residual Heat Removal Systems" (incorporate requirements of USI A-31)	5.4.3
A-36	NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants"	9.1.5

Table C.2 Task action plans for selected unresolved safety issue

Task no.	Issue date of task action plan
A-17	January 1984 (Rev.1)
A-43	January 1981
A-44	July 1980
A-45	October 1981
	June 1982 (Rev.1)
A-46	May 1982
A-47	May 1982
A-48	December 1982

APPENDIX D

ABBREVIATIONS

ABVS	auxiliary building ventilation system
A/C	air conditioning
ACI	American Concrete Institute
ACRS	Advisory Committee on Reactor Safeguards
AEC	Atomic Energy Commission
AEOD	Office of Analysis and Evaluation of Operational Data
AFW	auxiliary feedwater
AFWS	auxiliary feedwater system
AISC	American Institute of Steel Construction
ALARA	as low as reasonably achievable
ANS	American Nuclear Society
ANSI	American National Standards Institute
ARMS	area radiation monitoring system
ASCE	American Society of Civil Engineers
ASME	American Society of Mechanical Engineers
ASP	auxiliary shutdown panel
ASTM	American Society for Testing and Materials
ATWS	anticipated transients without scram
AVT	all volatile treatment
BAT	boric acid tank
BNL	Brookhaven National Laboratory
BOL	beginning of life
BOP	balance of plant
BOTF	Bulletins and Orders Task Force
BTP	branch technical position
B&W	Babcock and Wilcox
BWR	boiling water reactor
CAOC	constant axial control mode
CBI	control building isolation
CCHVS	charging pump, component cooling water pump, and heat exchangers exhaust ventilation system
CCN	command control network
CCS	condensate cleanup system
CDA	containment depressurization actuation
CDCCW	condensate deminerlaizer component cooling water
CDS	containment depressurization signal
CE	Combustion Engineering
CET	core exit thermocouple
CFR	Code of Federal Regulations
CHF	critical heat flux
CHRS	containment heat removal system
CIA	containment isolation phase A

CIB	containment isolation phase B
CIS	containment isolation signal
CL&P	Connecticut Light & Power Company
CONVEX	Connecticut Valley Electric Exchange
CPI	central pressure index
CRDM	control rod drive mechanism
CRDS	control rod drive system
CREFS	control room emergency filtration system
CREVS	control room emergency ventilation system
CSE	containment structure enclosure
CVCS	chemical and volume control system
DBA	design-basis accident
DEI-131	dose equivalent iodine-131
DEMA	Diesel Engine Manufacturers Association
DES	Draft Environmental Statement
DHR	decay heat removal
DNB	departure from nucleate boiling
DNBR	departure from nucleate boiling ratio
DOT	Department of Transportation
DWST	demineralized water storage tank
DWT	drop-weight tests
EAB	exclusion area boundary
EAS	essential auxiliary support
ECCS	emergency core cooling system
EGLS	emergency generator load sequencer
EHC	electrohydraulic control
EOF	emergency operations facility
EOP	emergency operating procedures
EPRI	Electric Power Research Institute
ERG	emergency response guideline
ESF	engineered safety feature
ESFAS	engineered safety feature actuation signal
FBES	fuel building exhaust system
FEMA	Federal Emergency Management Agency
FES	Final Environmental Statement
FMEA	failure modes and effects analysis
FSAR	Final Safety Analysis Report
GDC	general design criteri(on)(a)
GE	General Electric Company
GM	Geiger-Mueller
HAZ	heat-affected zone
HED	human engineering discrepancy
HELB	high-energy line break
HEPA	high efficiency particulate air
HJTC	heated junction thermocouple
HMR	Hydrometeorological Report
HPI	high-pressure injection

I&C	instrumentation and control
ICC	inadequate core cooling
IE	Office of Inspection and Enforcement
IEEE	Institute of Electrical and Electronics Engineers
ISEG	Independent Safety Engineering Group
ISI	inservice inspection
JTG	Joint Test Group
LANL	Los Alamos National Laboratory
LCO	limiting condition(s) for operation
LLNL	Lawrence Livermore National Laboratory
LOCA	loss-of-coolant accident
LOP	loss of offsite power
LP	low pressure
LPMS	loose parts monitoring system
LPZ	low population zone
LTOP	low-temperature overpressure protection
LWA	limited work authorization
LWR	light-water reactor
MCC	motor control center
MCES	main condenser evacuation system
MDNBR	minimum departure from nucleate boiling ratio
MMI	modified Mercalli intensity
MPC	maximum permissible concentration
MSIV	main steam isolation valve
MSL	mean sea level
MSLB	main steamline break
NDTT	nil-ductility transition temperature
NEPCO	New England Power Company
NEPTP	New England - Piedmont Tectonic Province
NFPA	National Fire Protection Association
NEUSSN	Northeastern United States Seismic Network
NIS	nuclear instrumentation system
NNECO	Northeast Nuclear Energy Company
NNS	non-nuclear safety
NOAA	National Oceanic and Atmospheric Administration
NPSH	net positive suction head
NRB	Nuclear Review Board
NRC	U.S. Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NSE	nuclear safety engineering
NSSS	nuclear steam supply system
NU	Northeast Utilities
NUSCo	Northeast Utilities Services Company
OBE	operating-basis earthquake
ODCM	Offsite Dose Calculation Manual
OFA	optimized fuel assembly
OL	operating license
O&M	operations and maintenance

PA	public address
PAD	performance analysis and design
PASS	postaccident sampling system
PBX	private branch exhange
PCI	pellet/cladding interaction
PCT	peak cladding temperature
PGP	procedures generation package
PID	proportional integral derivative
P&ID	piping and instrumentation diagram
PIS	process instrumentation system
PMH	probable maximum hurricane
PMP	probable maximum precipitation
PORC	Plant Operations Review Committee
PORV	power-operated relief valve
PSI	preservice inspection
PSNH	Public Service Company of New Hampshire
PSS	Probabilistic Safety Study
PTS	pressurized thermal shock
PVORT	Pump and Valve Operability Review Team
PWR	pressurized-water reactor
QA	quality assurance
QC	quality control
QSS	quench spray system
RCCA	rod cluster control assembly
RCIC	reactor core isolation cooling
RCP	reactor coolant pump
RCPB	reactor coolant pressure boundary
RCS	reactor coolant system
RG	regulatory guide
RHR	residual heat removal
RHRS	residual heat removal system
RITS	reactor inventory tracking system
RO	reactor operator
RPCCW	reactor plant component cooling water
RPM	Radiation Protection Manager
RPS	reactor protection system
RSS	recirculation spray system
RTS	reactor trip system
RVLIS	reactor vessel level intrumentation system
RWP	radiation work permit
RWST	refueling water storage tank
SAFDL	specified acceptable fuel design limit(s)
SCC	stress corrosion cracking
SEP	Systematic Evaluation Program
SER	Safety Evaluation Report
SER-CP	Safety Evaluation Report issued at Construction Permit stage
SFA	standard fuel assembly
SGBS	steam generator blowdown system
SGOG	Steam Generator Owners Group
SGTR	steam generator tube rupture

SHCP	Seismic Hazard Characterization Program
SI	safety injection
SIAS	safety injection actuation signal
SIS	safety injection system
SLCRS	supplementary leak collection and release system
SMM	subcooled margin monitor
SNETCo	Southern New England Telephone Company
SNRB	Site Nuclear Review Board
SNUPPS	Standardized Nuclear Unit Power Plant System
SORC	Station Operations Review Committee
SPDS	safety parameter display system
SQRT	Seismic Qualification Review Team
SRO	senior reactor operator
SRMS	safety-related monitoring system
SRP	Standard Review Plan
SRSS	square root of the sum of the squares
SSC	structures, systems, and components
SSE	safe shutdown earthquake
SSLPS	solid-state logic protection system
STA	shift technical advisor
STS	Standard Technical Specifications
SV	safety valve
SWEC	Stone & Webster Engineering Company
SWMS	solid waste management system
SWS	service water system
TAP	Task Action Plan
TID	technical information document
TLD	thermoluminescent dosimeter
TMI-2	Three Mile Island Unit 2
TSP	transfer switch panel
UHF	ultra high frequency
UHI	upper head injection
UHS	ultimate heat sink
UL	Underwriters Laboratory
USGS	U.S. Geological Survey
USI	unresolved safety issue
VHF	very high frequency
W	Westinghouse
WATS	wide-area telephone system
WOG	Westinghouse Owners Group

APPENDIX E

INTERIM NRC STAFF POSITION ON CHARLESTON EARTHQUAKE FOR LICENSING PROCEEDING

The NRC staff position with respect to the Modified Mercalli Intensity X, 1886 Charleston, South Carolina, earthquake has been that, in the context of the tectonic province approach used for licensing nuclear power plants, this earthquake should be restricted to the Charleston vicinity. This position was based, in part, on information provided by the United States Geological Survey (USGS) in a letter dated December 30, 1980, from J. E. Devine to R. E. Jackson (see Summer Safety Evaluation Report, NUREG-0717). The USGS has been reassessing its position and issued a clarification on November 18, 1982, in a letter from J. E. Devine to R. E. Jackson. As a result of this letter, a preliminary evaluation and outline for NRC action was forwarded to the Commission in a memorandum from W. J. Dircks on November 19, 1982.

The USGS letter states:

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

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

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

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

References

Letter, Nov. 18, 1982, from J. E. Devine (USGS) to R. E. Jackson (NRC).

Memorandum, Nov. 19, 1982, from W. J. Dircks (NRC) to the Commissioners.

U.S. Nuclear Regulatory Commission, NUREG-0717, "Safety Evaluation Report Related to the Operation of the Virgil C. Summer Nuclear Station, Unit No. 1," Jan. 1982.

APPENDIX F

NRC STAFF CONTRIBUTORS AND CONSULTANTS

This Safety Evaluation Report is a product of the NRC staff and its consultants. The NRC staff members listed below were principal contributors to this report.

NRC STAFF

Name	Title	Branch
L. Bell	Nuclear Engineer	Accident Evaluation
K. Dempsey	Nuclear Engineer	Accident Evaluation
N. Fioravante	Mechanical Engineer	Auxiliary Systems
R. Goel	Mechanical Engineer	Auxiliary Systems
W. LeFave	Mechanical Engineer	Auxiliary Systems
A. Singh	Mechanical Engineer	Auxiliary Systems
J. Stang	Fire Protection Engineer	Chemical Engineering
B. Turovlin	Materials Engineer	Chemical Engineering
F. Witt	Chemical Engineer	Chemical Engineering
P. Hearn	Containment Systems Engineer	Containment Systems
R. Palla	Containment Systems Engineer	Containment Systems
M. Dunenfeld	Reactor Fuels Engineer	Core Performance
A. Gill	Nuclear Engineer	Core Performance
H. Richings	Senior Reactor Physicist	Core Performance
P. Robinson	Emergency Specialist	Emergency Preparedness
R. Jachowski	Hydraulic Engineer	Environmental and Hydrologic Engineering
J. Jackson	Mechanical Engineer	Equipment Qualifications
N. Romney	Mechnaical Engineer	Equipment Qualifications
H. Walker	Materials Engineer	Equipment Qualifications
N. Su	Task Manager (USI)	Generic Issues
T. Cardone	Geologist	Geosciences
J. Kimball	Seismologist	Geosciences
J. Hoyt	Human Factors Engineer	Human Factors Engineering
J. Joyce	Senior Reactor Engineer	Instrumentation and Control Systems
H. Li	Reactor Engineer	Instrumentation and Control Systems
F. Allenspach	Nuclear Engineer	Licensee Qualifications
D. Shum	Systems Engineer	Licensee Qualifications
S. Bhatt	Materials Engineer	Materials Engineering
H. Conrad	Senior Materials Engineer	Materials Engineering
B. Elliot	Materials Engineer	Materials Engineering
J. Halapatz	Materials Engineer	Materials Engineering
D. Smith	Materials Engineer	Materials Engineering
R. Kirkwood	Principal Mechanical Engineer	Mechanical Engineering

Name	Title	Branch
R. Li	Mechanical Engineer	Mechanical Engineering
D. Terao	Mechanical Engineer	Mechanical Engineering
J. Lee	Nuclear Engineer	Meteorology and Effluent Treatment
J. Levine	Meteorologist	Meteorology and Effluent Treatment
C. Gaskin	Safeguards Engineer	Power Reactor Safeguards Licensing
R. Giardina	Mechanical Engineer	Power Systems
J. Knox	Senior Electrical Engineer	Power Systems
R. Becker	Nuclear Systems Engineer	Procedures and Systems Review
F. Liederbach	Principal Operational Safety Engineer	Procedures and Systems Review
W. Long	Operational Safety Engineer	Procedures and Systems Review
J. Spraul	Quality Assurance Engineer	Quality Assurance
J. Minns	Health Physicist	Radiological Assessment
V. Leung	Nuclear Engineer	Reactor Systems
N. Fields	Electrical Engineer	Site Analysis
W. Lambe	Antitrust Economist	Site Analysis
A. Sinisgalli	Site Analyst	Site Analysis
F. Anderson	Reactor Engineer	Standardization and Special Projects
J. Chen	Geotechnical Engineer	Structural and Geotechnical Engineering
N. Chokshi	Structural Engineer	Structural and Geotechnical Engineering

CONSULTANTS

Name	Organization
C. Hofmayer	Brookhaven National Laboratory
H. Spaletta	Idaho National Engineering Laboratory
J. Alzheimer	Pacific Northwest Laboratory
W. J. Bailey	Pacific Northwest Laboratory
C. E. Beyer	Pacific Northwest Laboratory
R. Gruel	Pacific Northwest Laboratory
R. Jensen	R. Jensen & Associates, Inc.