Millstone Power Station Unit 2
Safety Analysis Report

Chapter 1: Introduction And Summary

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1.1 INTRODUCTION

This Final Safety Analysis Report (FSAR) was initially submitted in support of the application of The Connecticut Light and Power Company (CL&P), The Hartford Electric Light Company (HELCO), Western Massachusetts Electric Company (WMECO), and Northeast Nuclear Energy Company (NNECO), for a license to operate the second nuclear powered generating unit at the site of the Millstone Power Station. Since the initial licensing of the unit, unless otherwise indicated, the FSAR has been updated a number of times to reflect current design and analysis information. On the basis of the information presented in the FSAR and referenced material at the time of application for operating license, the applicants concluded that Millstone Unit 2 is designed and constructed and will be operated without undue risk to the health and safety of the public.

Construction of Millstone Unit 2 was authorized by the United States Atomic Energy Commission (AEC) when it issued Provisional Construction Permit CPPR-76 on December 11, 1970. Commercial operation of Millstone Unit 2 commenced in December 1975 at a gross electrical output of 865 megawatts.

Millstone Unit 2 is located Millstone Point in the Town of Waterford, Connecticut. It is located immediately to the north of the first unit (Millstone Unit 1) and south of the third unit (Millstone Unit 3). Commercial operation of Millstone Unit 1 was authorized by the AEC by issuing Provisional Operating License DPR-21 on October 7, 1970. Commercial operation of Millstone Unit 1 commenced in December, 1970. Commercial operation of Millstone Unit 3 was authorized by the United States Nuclear Regulatory Commission (NRC) (formerly the AEC) by issuing the Low Power License on November 25, 1985, and the Full Power License on January 31, 1986. Commercial operation of Millstone Unit 3 commenced in April 1986. A licensing history for the Millstone Unit 2 plant is presented in Table 1.1-1.

Millstone Unit 2 utilizes a pressurized water nuclear steam supply system (NSSS). The unit is similar, in this respect, to the former Yankee Atomic Electric Company generating plant in Rowe, Massachusetts, (NRC Docket Number 50-29), the former Haddam Neck Plant operated by the Connecticut Yankee Atomic Power Company on the Connecticut River at Haddam, Connecticut (NRC Docket Number 50-213), and the Maine Yankee Atomic Power Company plant at Wiscasset, Maine (NRC Docket Number 50-309). The NSSS for Millstone Unit 2 is supplied by Combustion Engineering, Inc. (CE) which also supplied the steam supply system for the Maine Yankee plant. The Millstone Unit 2 NSSS is similar to the systems supplied by CE for the initial two units of the Baltimore Gas and Electric Calvert Cliffs Nuclear Power Plant (NRC Docket Numbers. 50-317 and 50-318).

Millstone Unit 2 has been designed to operate safely under all normal operating conditions and anticipated transients. Although the unit produces small amounts of radioactive waste, the offsite disposal of these wastes is rigidly controlled and maintained below established limits.

In 2001, Millstone Units 1, 2 and 3 operating licenses were transferred from Northeast Nuclear Energy Company to Dominion Nuclear Connecticut, Inc. (DNC).

DNC is an indirect wholly-owned subsidiary of Dominion Energy, which is in turn owned by Dominion Resources, Inc. (DRI). Virginia Power, which is the licensed owner and operator of the North Anna and Surry nuclear stations, is also a subsidiary of DRI.

The transmission and distribution assets on the site will continue to be owned by Connecticut Light and Power (CL&P) and will be operated under an Interconnection Agreement between CL&P and DNC.

The FSAR will retain references to Northeast Utilities and Northeast Nuclear Energy Company documents/activities when they are used in a historic context and are required to support the plant licensing bases.

Upon license transfer, all records and design documents necessary for operation, maintenance, and decommissioning were transferred to DNC. Some of these drawings are included (or referenced) in this FSAR. These drawings often have title blocks (or drawing numbers) which list Northeast Nuclear Energy Company (et. al) or Northeast Utilities Service Company (et. al). In general, no changes to these title blocks will be made at this time. Based on this general note, these drawings shall be read as if the title blocks list Dominion Nuclear Connecticut, Inc.

Millstone Unit 2 has been designed to operate reliably without accident. Nevertheless, to ensure that no reasonably credible accident could result in dangerous releases of radioactive material, the unit incorporates a number of features designed to minimize the effects of such an accident. The adequacy of these safety features under the conditions of various postulated accidents is discussed in Chapter 14.

The initial license to operate Millstone Unit 2 was at a full power core thermal output of 2560 megawatts. This corresponded to a NSSS thermal rating, which includes core power and other reactor coolant heat sources such as reactor coolant pumps and pressurizer heaters, of 2570 MWt.

Millstone Unit 2 is currently licensed for a steady state reactor core power level of 2700 MWt, corresponding to a NSSS rating of 2715 MWt. All Chapter 14 analyses have been evaluated on the basis of these current values.

Since the construction permit was issued, and during the design and construction of the unit, there have been no major deviations from the information supplied in the Preliminary Safety Analysis Report (PSAR). However, changes in various specific design features have been found desirable and these are covered in the appropriate sections of this report. A summary of the more significant design changes incorporated in the plant since the issuance of the PSAR up to the time of application for an operating license is provided in Section 1.7.

TABLE 1.1-1 LICENSING HISTORY

EVENT	DATE
Construction Permit Issued	December 11, 1970
Final Safety Analysis Report Filed	August 15, 1972
Full Term Operating Licensing Issued	September 26, 1975
Full Power License	September 26, 1975
Initial Criticality	October 17, 1975
100% Power	March 20, 1976
Commercial Operation	December 26, 1975
“Stretch Power”	June 25, 1979
Operating License Extension Requested	December 22, 1986
Operating License Extension Issued	January 12, 1988
Full Term Operating License Expires	December 11, 2010
Operating License Expires	July 31, 2035

1.2 SUMMARY DESCRIPTION

1.2.1 GENERAL

A summary description of Millstone Unit 2 of the Millstone Nuclear Power Station is provided in this section. The description includes the following:

a. Site

b. Arrangement

c. Reactor

d. Reactor coolant system

e. Containment system

f. Engineered safety features systems

g. Protection, control and instrumentation system

h. Electrical systems

i. Auxiliary systems

j. Steam and power conversion system

k. Radioactive waste processing system

l. Interrelation with Millstone Units 1 and 3

m. Summary of Codes and Standards

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

[redacted]

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

The containment houses the NSSS, consisting of the reactor, steam generators, reactor coolant pumps, pressurizer, and some of the reactor auxiliaries. The containment is equipped with a polar crane.

The enclosure building completely envelopes the containment and provides a filtration region between the containment and the environment.

The turbine building houses the turbine generator, condenser, feedwater heaters, condensate and feedwater pumps, turbine auxiliaries and certain of the switchgear assemblies.

Withhold under 10 CFR 2.390 (d) (1)

[redacted]

1.2.4 REACTOR

The reactor is a pressurized light water cooled and moderated type fueled by slightly enriched uranium dioxide. The uranium dioxide is in the form of pellets and is contained in pressurized Zircaloy-4 tubes fitted with welded end caps. These rods are arranged into fuel assemblies each consisting of 176 fuel rods arranged on a 14 rod square matrix. Space is left in the fuel rod array to allow for the installation of five guide tubes. These guide tubes provide for the smooth motion of control element assembly fingers. The assembly is fitted with end fittings and spacer grids to maintain fuel rod alignment and to provide structural support. The end fittings are also drilled with flow holes to provide for the flow of cooling water past the fuel tubes.

The reactor is controlled by a combination of chemical shim and solid absorber. The solid absorber is boron carbide pellets or stainless steel contained in tubular Inconel elements. Some earlier elements had used stainless steel as the absorber material. Five absorber elements are connected together by a spider yoke in a square matrix with a center element. The five elements constitute a control element assembly (CEA). The 73 CEAs are connected, either singly or dually, through extension shafts, to 61 magnetic jack type control element drive mechanisms (CEDMs) which are mounted on nozzles on the reactor vessel head. Each CEA is aligned with and can be inserted into the guide tubes of fuel assemblies. The dual CEAs are utilized for shutdown rods. The single CEAs are divided into regulating groups. The eight part length control rods of Cycle One were replaced by dummy flow plugs. Two of the flow plugs were replaced by reactor vessel level indication system detectors, then in Cycle Twelve, the last six remaining flow plugs were removed. The resulting increase in core bypass flow has been accounted for in the safety analysis.

The replacement head has a total of 78 nozzle penetrations. 67 of these nozzles are suitable for supporting control element drive mechanisms (61 are in use, while the other 6 nozzles are capped with nozzle adapters). Two nozzles are used for heated junction thermocouples, which enable monitoring reactor vessel between the top of the vessel dome and the area directly above the fuel bundles. Eight nozzles are used for nuclear instrumentation and one nozzle is used for the reactor vessel head vent. The location, size and the number of nozzles on the replacement reactor vessel closure head are maintained in the same configuration as before (prior to cycle 16).

Chemical shim control is provided by boric acid dissolved in the coolant water. The concentration of boric acid is maintained and controlled as required by the chemical and volume control system.

The reactor core rests on the core support plate assembly which is supported by the core support barrel. The core support barrel is a right circular cylinder supported from a machined ledge on the inside surface of the vessel flange forging. The support plate assembly transmits the entire weight of the core to the core support barrel through a structure made of beams and vertical columns. Surrounding the core is a shroud which serves to limit the coolant which bypasses the core. An upper guide structure, consisting of upper support structure, control element assembly shrouds, a fuel alignment plate and a spacer ring, serves to support and align the upper ends of the fuel assemblies, prevents lifting of the fuel assemblies in the event of a loss-of-coolant accident (LOCA) and maintains spacing of the CEAs. Chapter 3 contains more detailed information on the reactor.

1.2.5 REACTOR COOLANT SYSTEM

The reactor coolant system consists of two closed heat transfer loops in parallel with the reactor vessel. Each loop contains one steam generator and two pumps to circulate coolant. An electrically heated pressurizer is connected to one loop hot leg. The coolant system is designed to operate at a thermal power level of 2715 MWt to produce steam at a nominal pressure of 880 psia.

The reactor vessel, loop piping, pressurizer and steam generator plenums are fabricated of low alloy steel, clad internally with austenitic stainless steel. The pressurizer surge line and coolant pumps are fabricated from stainless steel and the steam generator tubes are fabricated from Inconel.

Overpressure protection is provided by power-operated relief valves and spring-loaded safety valves connected to the pressurizer. Safety and relief valve discharge is released under water in the quench tank where the steam discharge is condensed.

The two steam generators are vertical shell and U-tube steam generators each of which produces 5.9×10^6 lb/hr of steam. Steam is generated in the shell side of the steam generator and flows upward through moisture separators. Steam outlet moisture content is less than 0.2 percent.

The reactor coolant is circulated by four electric motor-driven, single-suction, centrifugal pumps. Each pump motor is equipped with a non reverse mechanism to prevent reverse rotation of any pump that is not being used during operation with less than four pumps energized. Chapter 4 contains more detailed information on the reactor coolant system.

1.2.6 CONTAINMENT SYSTEM

A double containment system is used for Unit 2. The containment system consists of a prestressed concrete cylindrical structure referred to as the containment, which is completely enclosed by the enclosure building (EB). The enclosure building filtration region (EBFR) includes the region between the containment and the enclosure building, the penetration rooms and engineered safety feature equipment rooms. In the unlikely event of a LOCA the EBFR is maintained at a slightly negative pressure by the enclosure building filtration system (EBFS). Air in the EBFR would be processed through charcoal filters and released through the 375 foot Millstone stack during a LOCA.

The containment uses a prestressed post-tensioned concrete design. The containment is a vertical right cylindrical structure with a dome and a flat base. The interior is lined with carbon steel plate to further ensure leak tightness.

Inside the containment, the reactor and other NSSS components are shielded with concrete. Access to portions of the containment during power operation is permissible.

The containment, in conjunction with the engineered safety features, is designed to withstand the highest internal pressure and coincident temperature resulting from the main steam line break accident (Section 14.8.2). The structural design conditions are for an internal pressure of 54 psig and a coincident equilibrium temperature of 289°F.

The enclosure building is a limited leakage steel framed structure partially supported off the containment and auxiliary building with uninsulated metal siding and an insulated metal roof deck.

1.2.7 ENGINEERED SAFETY FEATURES SYSTEMS

The engineered safety features systems (ESFS) provide protection for the public and plant personnel against the incidental release of radioactive products from the reactor system, particularly as a result of postulated LOCA. These safety features localize, control, mitigate and terminate such accidents to hold exposure levels below the applicable limits of 10 CFR Part 50.67.

The engineered safety features consist of the following systems:

a. Safety injection

b. Containment spray

c. Containment air recirculation and cooling

d. Enclosure building filtration

e. Hydrogen control

f. Auxiliary feedwater automatic initiation system

Each of these systems is divided into two redundant independent subsystems which in turn are powered by the associated redundant independent emergency electrical subsystem (see Section 1.2.9). The first three are cooled by the associated redundant independent reactor building closed cooling water headers (see Section 1.2.10.3).

Following a postulated LOCA, borated water is injected into the reactor coolant system by either high and/or low pressure safety injection pumps and safety injection tanks. This provides cooling to limit core damage and fission product release, and assures an adequate shutdown margin. The safety injection system also provides continuous long term post-accident cooling of the core by recirculating borated water from the containment sump through shutdown cooling heat exchangers and back to the reactor core (see Section 6.2).

Four safety injection tanks are provided, each connected to one of the four reactor inlet lines. The volume of each tank is 2019 cubic feet. Each tank contains about 1100 cubic feet of borated water at refueling concentration and is pressurized with nitrogen at 200 psig. In the event of a LOCA, the borated water is forced into the reactor coolant system by the expansion of the nitrogen. The water from three tanks adequately cools the entire core. Borated water is injected into the same nozzles by two low pressure and three high pressure injection pumps taking suction from the refueling water storage tank (RWST). For maximum reliability, the design capacity from the combined operation one high pressure and one low pressure pump provides adequate injection flow for any LOCA; in the event of a design basis accident (DBA), at least one high pressure and one low pressure pump will receive power from the emergency power sources if preferred power is lost and one of the emergency diesel generators is assumed to fail. When the refueling water storage tank supply is nearly depleted, the high pressure pump suctions automatically transfer to the containment sump and the low pressure pumps are shut down. One high pressure pump has sufficient capacity to cool the core adequately at the start of recirculation. During recirculation, heat in the recirculating water is removed through the shutdown cooling heat exchangers via either the low pressure injection pumps or containment spray pumps.

The safety injection pumps are located outside the containment to permit access for periodic testing during normal operation. The pumps discharge into separate headers which lead to the containment. Test lines are provided to permit running the pumps for test purposes during plant operation.

The safety injection system is designed in accordance with AEC General Design Criteria 35, 36, and 37 in Appendix A to 10CFR50 and General Criteria as described in Section 6.1. An analysis of the performance of the safety injection system (emergency core cooling system) following a postulated LOCA is given in Section 14.6.

Two independent, full capacity systems are provided to remove heat from the containment atmosphere by containment sprays and/or air recirculation and cooling after the postulated LOCA.

a. The containment spray system supplies borated water to cool the containment atmosphere. The spray system is sized to provide adequate cooling with two containment spray pumps. The pumps take suction from the refueling water storage tank. When this supply is nearly depleted, the pump suction is transferred automatically to the containment sump (see Section 6.4).

b. The containment air recirculation and cooling system is designed to cool the containment atmosphere. The cooling coils and fans are sized to provide adequate containment cooling with three of the four units in service (see Section 6.5).

c. A combination of one containment spray pump aligned with the shutdown cooling heat exchanger and two containment air recirculation units provides adequate cooling of the containment. Each spray pump and two associated containment air recirculation units are cooled by one of two associated redundant reactor building cooling water and service water subsystems. They are powered by the associated emergency electrical subsystem.

The enclosure building filtration system would collect and filter all potential containment leakage and minimize environmental radioactivity levels resulting from the discharge of all sources of containment leakage into the enclosure building filtration region in the unlikely event of a LOCA. The enclosure building filtration system would also collect and filter any radioactive releases in the unlikely event of a fuel handling accident inside the containment or spent fuel pool areas (see Section 6.7).

The hydrogen control system is provided to mix and monitor the concentration of hydrogen gas within the containment. This system consists of the post-accident recirculation system for mixing the containment environment and the hydrogen monitoring system for continuous monitoring of the post-accident containment atmosphere. The hydrogen purge system and hydrogen recombiners which are not credited in accident analyses are provided for reducing containment hydrogen concentrations.

The auxiliary feedwater automatic initiation system, (AFAIS), is provided to ensure delivery of sufficient feedwater to the steam generators in event of the loss of main feedwater. This system automatically actuates two motor driven auxiliary feedwater pumps (see Section 10.4.5.3), and opens the two auxiliary feedwater flow control valves via the automatic initiation control circuitry (see Section 7.3.2.2.h). The AFAIS is actuated upon completion of a 2-out-of-4 logic matrix initiated by a low steam generator level. Upon receipt of an actuation signal both pumps are started and the flow control valves to both steam generators are opened (see Section 7.3).

1.2.8 PROTECTION, CONTROL AND MONITORING INSTRUMENTATION

Various instrumentation systems provide protection, control, and monitoring functions for the safe and efficient operation of Millstone Unit 2.

Protection instrumentation systems function to shut down the reactor and activate safety systems if continuously monitored key plant process parameters exceed predetermined limits. Specific protection instrumentation systems include the Reactor Protective System (RPS) and the Engineered Safety Features Actuation System (ESFAS). The RPS functions to shut down or trip the reactor if any two of four safety channels generate coincident trip signals. An RPS trip removes power from the reactor control rods, allowing them to drop into the reactor, and shut it down. The ESFAS functions to actuate the engineered safety features systems described in FSAR Section 1.2.7. The exception to this is the containment purge valve isolation where one of four containment air radiation detectors can generate a trip signal. Actuation of the ESFS occurs if any two of four safety channels generate coincident trip signals.

Control instrumentation systems function to maintain plant parameters within operational limits during both steady state and normal operating transients. Major control systems include the Control Element Drive System (CEDS), the Reactor Regulating System (RRS), Pressurizer Level Regulating System (PLRS), Reactor Coolant Pressure Regulating System (RCPRS), Feed Water Regulating System (FWRS), and Turbine Generator Control System (TGCS).

Indications are provided to monitor normal and abnormal plant operation. Indicators are located within the control room and throughout the plant. The indicators are used to monitor the status and operation of the protective and control systems, and the status of other support systems. Major indication systems include the Control Element Assembly (CEA) Position Indication, Nuclear Instrumentation (NI), In-Core Instrumentation (ICI), Radioactivity Monitoring System (RMS), Integrated Computer System (ICS), Control Room Annunciation, and Post Accident Monitoring Instrumentation (PAMI).

Details of the above and other protective, control, and monitoring instrumentation systems are provided in Chapter 7.

1.2.9 ELECTRICAL SYSTEMS

The Millstone Nuclear Power Station consists of Millstone Unit 1 which is no longer generating power, Millstone Unit 2 with a 1011-MVA, 0.90 power factor generator, and Millstone Unit 3 with a 1354.7-MVA, 0.925 power factor generator (see Chapter 8).

The Millstone Unit 2 generator output is fed through a step up transformer bank to the 345 kV switchyard. The switchyard is connected to the high voltage transmission system through four 345 kV transmission lines. The switchyard, in addition to carrying the electrical output of the station, also provides a means of supplying power to the units from external sources. Startup power and reserve auxiliary power for Millstone Unit 2 are taken from the 345 kV switchyard through the reserve station service transformer. Normal station service power is taken from the generator main leads through the normal station service transformer. A second source of off site power for the engineered safety features is provided from normal station service transformer 15G-3SA or reserve station service transformer 15G-23SA, both associated with Millstone Unit 3 via a 4160V crosstie connection. Two diesel generators provide the on site emergency power for Millstone Unit 2. The 4160V crosstie from Unit 3 can also be configured (by operator action) to supply power directly from the Unit 3 Alternate AC (SBO) diesel generator to provide an alternate AC source for Unit 2 Appendix R and Station Blackout requirements.

Auxiliary power for Millstone Unit 2 is provided at 6900, 4160, 480, and 120/208 volts. Direct current 125 volt systems are also available for emergency power, engineered safety feature control, and essential nuclear instrumentation, control and relaying.

The preferred and on site emergency sources of electrical power are each adequate to permit prompt shutdown and maintain safe conditions under all credible circumstances. The on site emergency power source consists of two separate and redundant diesel generators. Each diesel is capable of carrying all required auxiliary loads following postulated LOCA without exceeding its continuous rating.

Each of the two separate and redundant station batteries is capable of carrying essential 125 volt DC and 120 volt AC inverter loads associated with a postulated LOCA.

The redundant channel wiring associated with these emergency electrical sources is physically separated.

1.2.10 AUXILIARY SYSTEMS

1.2.10.1 Chemical and Volume Control System

The chemistry of the reactor coolant is controlled by purification of a regulated letdown stream of reactor coolant. Water removed from the reactor coolant system is cooled in the regenerative heat exchanger. The fluid pressure is then reduced and flow is regulated by the letdown control valves. Temperature is reduced further in the letdown heat exchanger. From there, the flow passes through a filter and a purification ion exchanger to remove corrosion and fission products. A small fraction of the flow is diverted prior to entering the ion exchanger. This stream of coolant flows through a process radiation monitor. Upon leaving the ion exchanger, the coolant flows through a strainer and another filter and is then sprayed into the volume control tank.

Coolant is returned to the reactor coolant system by the charging pumps, through the regenerative heat exchanger. Prior to entering the charging pumps, the coolant boron concentration is adjusted to meet the reactor reactivity requirements. In addition, provision is made to inject chemical additives to the suction of the charging pumps for coolant chemistry control.

The volume control system automatically controls the rate at which coolant must be removed from the reactor coolant system to maintain the pressurizer level within the prescribed control band, thereby compensating for changes in volume due to coolant temperature changes. Using the volume control tank as a surge tank decreases the quantity of liquid and gaseous wastes which would otherwise be generated.

Reactor coolant system makeup water is taken from the primary water storage tank and the two concentrated boric acid storage tanks. The boric acid solution is maintained at a temperature which prevents crystallization. The makeup water is pumped through the regenerative heat exchanger into the reactor coolant loop by the charging pumps.

Boron concentration in the reactor coolant system can be reduced by diverting the letdown flow away from the volume control tank to the radioactive waste processing system. Demineralized water is then used for makeup.

When the boron concentration in the reactor coolant system is low, the feed and bleed procedure previously described would generate excessive volumes of waste to be processed. Therefore, the chemical and volume control system is equipped with a deborating ion exchanger which reduces boron concentration late in cycle life. A complete description is given in Section 9.2.

1.2.10.2 Shutdown Cooling System

The shutdown cooling system (see Section 9.3) is used to reduce the reactor coolant temperature, at a controlled rate, from 300°F to a refueling temperature of approximately 130°F. It also maintains the proper reactor coolant temperature during refueling. Once entry conditions are met, the shutdown cooling system can provide long term cooling capability in the event of a LOCA after the reactor coolant system has refilled (see Section 14.6.5.3).

The shutdown cooling system utilizes the low pressure safety injection pumps to circulate the reactor coolant through two shutdown cooling heat exchangers. It is returned to the reactor coolant system through the low pressure safety injection header.

The reactor building closed cooling water system (RBCCW) supplies cooling water for the shutdown heat exchangers.

1.2.10.3 Reactor Building Closed Cooling Water System

The RBCCW system consists of two separate independent headers, each of which includes a RBCCW pump, a service water (seawater)-cooled RBCCW heat exchanger, interconnecting piping, valves and controls. A third RBCCW pump and a third RBCCW heat exchanger are provided as installed spares. The corrosion inhibited, demineralized water in this closed system is circulated through the RBCCW heat exchanger where it is cooled to 85°F by seawater which has a maximum design inlet temperature of 80°F (see Section 9.4).

The RBCCW system removes heat from the containment atmosphere, engineered safety feature components and various auxiliary system/components handling the reactor coolant. Items cooled by the RBCCW system include:

Containment air recirculation and cooling unit
Reactor vessel support concrete cooling coils
Containment spray pump seal coolers
High and low pressure safety injection pump seal coolers
Shutdown cooling heat exchangers
Engineered safety feature room air recirculation coils
Reactor coolant pump thermal barrier and oil coolers
Primary drain and quench tanks heat exchanger
CEDM coolers
Letdown heat exchanger
Degasifier effluent cooler
Degasifier vent condenser
Sample coolers
Spent fuel pool heat exchangers
Waste gas compressor aftercoolers
Steam generator blowdown quench heat exchanger

Each of the independent headers supply cooling water to components in the associated redundant safety related sub-systems (see Section 1.2.7). The RBCCW heat exchangers, connected to each independent RBCCW headers, are cooled by the associated independent service water header (see Section 1.2.10.6). Components in each independent RBCCW header, the associated safety related subsystems, and the associated service water header are powered from the associated redundant independent emergency electrical power subsystem (see Section 1.2.9).

Remote manually operated valves allow the spare RBCCW pump and/or heat exchanger to be operated with either of the two independent headers. The RBCCW surge tank absorbs the volumetric changes caused by temperature changes of the water within the RBCCW headers.

A chemical addition system is provided for the RBCCW system to maintain the corrosion inhibitor concentration as required.

During normal plant operation and normal shutdown, both of the independent RBCCW headers are in service.

Following a postulated LOCA, each of the RBCCW headers, in conjunction with the associated service water header and electrical subsystem, would provide the necessary cooling capacity to the associated engineered safety feature subsystems.

1.2.10.4 Fuel Handling and Storage

The fuel handling systems provide for the safe handling of fuel assemblies and control element assemblies and for the required assembly, disassembly, and storage of the reactor vessel head and internals. These systems include a refueling machine located inside the containment above the refueling pool, the fuel transfer carriage, the upending machines, the fuel transfer tube, a fuel handling machine over the spent fuel pool, a new fuel elevator in the spent fuel pool, a spent fuel cask crane, a new fuel inspection machine in the fuel handling area of the auxiliary building, and various devices used for handling the reactor vessel head and internals (see Section 9.8).

New fuel is stored dry in vertical racks within a storage vault near the spent fuel pool in the auxiliary building. Storage space is provided for approximately one-third of a core.

The vault is designed to avoid criticality by spacing fuel assemblies at 20.5 inches, center to center. The spent fuel pool, located in the auxiliary building, is constructed of reinforced concrete lined with stainless steel. The spent fuel storage racks are separated into four regions, designated Regions 1, 2, 3, and 4. Section 9.8.2.1 contains a detailed description of spent fuel storage design and components.

Cooling and purification equipment is provided for the spent fuel pool water (see Section 9.5). This equipment can also be used to clean up the refueling water during and after its use in the refueling pool. Backup cooling methods are also available.

1.2.10.5 Sampling System

The sampling system consists of Sampling Stations 1 and 2, the Post Accident Sampling System (PASS), the Corrosion Monitoring Sample Station, and the Waste Gas Sample Sink. These provide the means for determining physical, chemical and radioactive conditions of process fluids, waste gas and containment air. The system is supplemented by independent sampling of nonradioactive fluids in numerous locations within the unit, including sampling of the chlorinated water. (See Section 9.6.)

1.2.10.6 Cooling Water Systems

The exhaust steam from the main turbine and steam generator feedwater pump turbines is condensed in the condenser, which is cooled, in turn, by circulating water flowing through the condenser tubes, (see Section 9.7.1).

Four circulating water pumps, with 548,800 gpm total capacity, take suction from and discharge to Long Island Sound. The circulating water system is designed to maintain condenser back pressure at 2 inches Hg absolute with a 60.8°F inlet circulating water temperature.

The service water system (see Section 9.7.2) provides cooling water to the RBCCW, TBCCW, diesel engine cooling water, chilled water system heat exchangers, vital switchgear room cooling coils and the circulating water pump bearings. Three vertical, centrifugal, half capacity service water pumps have a design flow of 12,000 gpm, each with a total dynamic head of 100 feet of water. These pumps take suction from and discharge to Long Island Sound.

The service water system consists of two redundant, independent cross-connected supply headers with isolation valves to all heat exchangers and two discharge headers for the RBCCW heat exchangers. Two discharge headers exist for the emergency diesel generator cooling water; once underground these headers combine prior to entering the discharge canal. Service Water discharge from the TBCCW, chilled water system and vital switchgear room cooling heat exchangers combine into a common header prior to entering the discharge canal. Each of the supply headers is supplied by one of the service water pumps. During normal operation and shutdown and following a postulated LOCA, the two pumps connected to the two redundant supply headers are in service. However, only one service water pump and header is required to provide cooling of the RBCCW and diesel following a LOCA or for unit shutdown. Remote manually operated valves allow the third service water pump to be connected to either of the redundant headers.

The intake structure consists of four independent bays. The intake structure is equipped with a chlorination system, consisting of two 1800 gallon sodium hypochlorite storage tanks and two injection systems with one supplying sodium hypochlorite to the service water system and the other to the circulating water intake.

1.2.10.7 Ventilation Systems

Normally the containment environment is cooled by the containment air recirculation and cooling system. Following a postulated LOCA, these units reduce the temperature and pressure of the containment atmosphere to a safe level (see Sections 1.2.7, 6.5 and 9.9.1). The containment auxiliary circulation fans maintain uniform containment environmental temperature by mixing the air. Normally, the environment for the control element drive mechanisms is maintain by the CEDM fan-coil units. A forced outside air purge system is provided to maintain a suitable environment within the containment whenever access is desired. The exhaust of this containment air purge system is monitored to assure that radioactive effluents are maintained within acceptable limits.

The auxiliary building is served by separate ventilation systems in the fuel handling area, the radioactive waste area and for the nonradioactive waste area. Each area is provided with a heating and ventilating supply unit and separate exhaust fans. Exhausts from the potentially contaminated areas are filtered through high efficiency particulate air (HEPA) filters, monitored, and discharged through the Unit 2 stack. Exhaust from clean areas is discharged directly to the atmosphere (see Section 9.9.6).

Handling of irradiated fuel or moving a cask over the spent fuel pool does not require fuel handling area integrity or ventilation but it may be desirable to use the main exhaust or EBF systems, if available, as the exhaust discharge paths. If boundary integrity is set then these discharge paths provide a monitored radiological release pathway. If boundary integrity is not assured then suitable radiological monitoring is recommended per the Millstone Effluent Control Program.

The ventilation systems (main exhaust and EBFS) are normally available to provide for a filtered and monitored release pathway for effluents from the fuel handling area. If ventilation is not available, releases from the fuel handling area are monitored per the Millstone Effluent Control Program to ensure appropriate radiological effluent controls are in place.

Two full capacity and redundant air conditioning systems are provided for the control room. In the event of an accident, a bypass through either of the two full capacity and redundant control room filtration systems, which contain charcoal filters, is provided to protect control room operating personnel from exposure to high radiation levels.

The turbine building is equipped with supply and exhaust fans for year round ventilation.

The access control area is air conditioned for year-round comfort. All other areas are provided with ventilation for cooling during summer and unit heaters for heating during the winter.

1.2.10.8 Fire Protection System

The fire protection systems' (see Section 9.10) function to protect personnel, structures, and equipment from fire and smoke. The fire protection systems have been designed in accordance with the applicable National Fire Protection Association (NFPA) Codes and Standards, regulatory requirements, industry standards, and approved procedures. The design of the various fire protection systems has been reviewed by American Nuclear Insurers (ANI).

The fire detection and protection systems are designed such that a fire will be detected, contained, and/or extinguished. This is accomplished through the use of noncombustible construction, equipment separation, fire walls, stops and seals, fire detection systems, and automatic and manual water suppression systems. As a minimum, portable extinguishers, hose stations, and fire hydrants are available for all areas to control or extinguish a fire.

1.2.10.9 Compressed Air Systems

The instrument air system consists of one 640 scfm and two 237 scfm (each) instrument air compressors, receivers, dryers, and after-filters to provide a reliable supply of clean, oil free dry air for the unit pneumatic instrumentation and valves. Station air for normal unit maintenance is provided by a separate 630 scfm station air compressor. Operating pressures for both systems range between 80 to 120 psig depending on how the compressors are aligned and how the systems are interconnected.

The station air is used as a backup to the instrument air with tie-in points at the receiver inlets and inside the containment. The compressed air systems for Units 3 and 2 are interconnected by piping and manually operated valves.

Descriptions of the compressed air systems are given in Section 9.11.

1.2.11 STEAM AND POWER CONVERSION SYSTEM

The turbine generator for Unit 2 is furnished by General Electric Company. It is an 1800 rpm tandem compound, four flow exhaust, indoor unit designed for saturated steam conditions.

Under nominal steam conditions of 870 psia and 528°F at the turbine stop valve inlets and with turbines exhausting against a condenser pressure of 2 inches Hg absolute, the gross electrical output is 935 MWe. Turbine output corresponds to a NSSS thermal power level of approximately 2715 MWt.

The condensate and feedwater system consists of three condensate pumps, one steam packing exhauster, two steam jet air ejectors, two external drain coolers, two trains each having five stages of low pressure feedwater heaters, two turbine-driven steam generator feedwater pumps, two high pressure feedwater heaters as well as the associated piping, valves and instrumentation.

Normally, the steam generator feedwater pump turbines are driven by extraction steam. At low loads, main steam is used to drive the steam generator feedwater pump turbines.

A complete description of the steam and power conversion system is given in Section 10.

1.2.12 RADIOACTIVE WASTE PROCESSING SYSTEM

The radioactive waste processing system provides controlled handling and disposal of liquid, gaseous and solid waste from Unit 2 (see Section 11.1). Gaseous and liquid wastes discharged to the environment are controlled to comply with the limits given in the Technical Specifications and established to meet the requirements of 10 CFR Part 20 Sections 1301 and 1302 and Appendix B and the “as low as reasonably achievable (ALARA)” requirement of 10 CFR Part 50, Appendix I. The radioactive waste processing system consists of the following parts.

a. Clean Liquid Waste Processing System

   The clean liquid waste processing system collects and processes reactor coolant wastes from the chemical and volume control system, primary drain tank and the closed drains system. The system is comprised of pumps, filters, degasifier, demineralizers, receiver tanks, monitor tanks and the necessary instrumentation, piping, controls and accessories.

   The processed clean liquid wastes are collected in monitor tanks, sampled, and monitored prior to discharge to the circulating water system after ensuring that the predetermined limits for release are not exceeded.

b. Aerated Liquid Waste Processing System

   Aerated liquid wastes, consisting of radioactive liquid wastes exposed to the atmosphere, are collected in drain tanks and processed through filters, and demineralizers. The processed wastes are collected in a monitor tank, sampled, and monitored prior to discharge to the circulating water system after ensuring that the predetermined limits for release are not exceeded.

c. Gaseous Waste Processing System

   Radioactive waste gases are collected through the waste gas header into the waste gas surge tank. These gases are drawn from the surge tank by one of two compressors and are pumped into a waste gas decay tank for storage to allow radioactive decay. After decay, the tank contents are sampled and monitored prior to discharge and released through a particulate filter, at a predetermined controlled rate, into the Millstone stack. The discharge is monitored prior to its entering the stack and while in the stack, thus ensuring that the predetermined limits for release are not exceeded. The six waste gas decay tanks which are provided allow a minimum of 60 days storage capacity prior to release.

d. Solid Waste Processing System

Radioactive solid wastes are collected and placed in suitable containers for off site disposal. Spent demineralizer resins are held for radioactive decay prior to being dewatered and placed in a shielded cask for removal. Contaminated filter elements are placed in shielded drums for subsequent storage and off site disposal.

Low activity compactible solid wastes such as contaminated rags, paper, etc., are compacted at the Millstone Radwaste Reduction Facility prior to being shipped for disposal. Noncompactible solid wastes may be shipped to an off site processor for volume reduction prior to disposal.

1.2.13 INTERRELATION WITH MILLSTONE UNITS 1 AND 3

A number of the facilities of the Millstone Nuclear Power Station are common to Millstone Units 1, 2, and 3. The safe shutdown of any unit will not be impaired by the failure of the facilities and systems which are shared. A list of these facilities and systems follows:

a. Facilities

Radiochemistry laboratory
Radioactive and clean change facilities, including showers, lockers, clothing storage, and toilets
Radiation Protection offices
Instrument repair room
Warehouse machine shop
Millstone stack (for Unit 2 waste gas), main condenser air ejector and enclosure building filtration system discharge
General offices
First aid station
Lunch room
Visitors gallery
345 kV switch yard
Millstone Unit 3 normal station service transformer 15G-3SA
Millstone Unit 3 reserve station service transformer 15G-23SA
Millstone Unit 3 SBO diesel generator system
Makeup water treatment (Millstone Units 2 and 3 only)
Bulk storage chemical ton (Millstone Units 2 and 3 only)

Millstone Unit 2 Control Room (for monitoring and controlling Millstone Unit 1 systems)

b. Systems

Low pressure nitrogen storage
Fire protection (water supply and fire detection)
Auxiliary steam
Makeup water treatment
Building heating
Sanitary sewers
Plant water
Communications
Station Air (A system cross-tie between Unit 3 service air and Unit 2 station air headers is provided)

Operating and maintenance personnel are employed in all three units as described in Section 12.1.

Both units have a double containment system with rectangular outer envelopes.

The 40 CFR 190 off site radiation dose limits will not be exceeded by simultaneous operation of Millstone Units 1, 2, and 3.

The Millstone Station Physical Security Plan has been implemented in accordance with 10 CFR 73.55 “Requirements for Physical Protection of Licensed Activities in Nuclear Power Reactors Against Industrial Sabotage” to prohibit unauthorized access to vital areas.

This plan includes measures to deter or prevent malicious actions that could result in the release of radioactive materials into the environment through sabotage. Section 12.7 contains a description of the Security Plan.

1.2.14 SUMMARY OF CODES AND STANDARDS

To ensure the integrity and operability of pressure-containing components important to safety, established codes and standards are used in the design, fabrication and testing. Table 1.2-1 lists these codes and standards for components relied upon to prevent or mitigate the consequences of incidents and malfunctions originating within the reactor coolant pressure boundary, to permit shutdown of the reactor, and to maintain the reactor in a safe shutdown condition.

TABLE 1.2-1 SUMMARY OF CODES AND STANDARDS FOR COMPONENTS OF WATER-COOLED NUCLEAR POWER UNITS (1)

CODE CLASSIFICATION

Component	Group A	Group B	Group C	Group D
Pressure Vessels	ASME Boiler and Pressure Vessel Code, Section III, Class A, 1968 Edition, Addenda through Summer 1969	ASME Boiler and Pressure Vessel Code, Section III, Class C (1968 Edition including Addenda through Summer 1969)	ASME Boiler and Pressure Vessel Code, Section VIII, Division 1	ASME Boiler and Pressure Vessel Code, Section VIII, Division 1 or Equivalent
	Reactor Vessel (2)	Safety Injection Tanks (4)	Reactor Building Closed Cooling Water Heat Exchangers (3)	Service Water Strainers (3)
	Pressurizer (2)		Reactor Building Closed Cooling Water Surge Tank	Vital Chilled Water System Condensers/ Evaporators
	Steam Generators (3)	Shutdown Heat Exchangers (2) Concentrated Boric Acid Storage Tanks (2)		
		Refueling Water Storage Tank (4)		
0-15 psig Storage Tanks	—	API-620 with the NDT Examination Requirements in Table NST-1, Class 2	API-620 with the NDT Examination Requirements in Table NST-1, Class 3	API-620 or Equivalent Condenser Storage Tank
Atmospheric Storage Tanks	—	Applicable Storage Tank Codes such as API-650, AWWAD100 or ANSI B96.1 With the NDT Examination Requirements in Table NST-1, Class 2	Applicable Storage Tank Codes Such as API-650 AWWAD100 or ANSI B 96.1 with the NDT Examination Requirements in Table NST-1, Class 3	API-650, AWWAD100 or ANSI B 96.1 or Equivalent Diesel Oil Supply Tanks

TABLE 1.2-1 SUMMARY OF CODES AND STANDARDS FOR COMPONENTS OF WATER-COOLED NUCLEAR POWER UNITS [1] (CONTINUED)

CODE CLASSIFICATION

Component	Group A	Group B	Group C	Group D
Pumps and Valves	1. ASME Standard Code for Pumps and Valves for Nuclear Power, Class 1, March 1970 Draft	Draft ASME Code for Pumps and Valves, Class II, November 1968. See Footnote [5].	Draft ASME Code for Pumps and Valves Class III	Valves - ANSI B 31.1.0 or Equivalent Pumps - Draft ASME Code for Pumps and Valves Class III or Equivalent
	2. ASME Section III, Paragraph N153 in Summer 1969 Addenda			
	3. ASME Section III, Appendix IX Reactor Coolant Pumps and Valves	High Pressure Safety Injection Pumps and Valves	Vital Chilled Water Pump	Vital Chilled Water Valves
		Low Pressure Safety Injection Pumps and Valves ASME Section III 1971 Edition, 1971 Winter Addenda Reactor Coolant System Branch Connection Valves beyond Second Isolation Valves ASME Standard Code for Pumps and Valves, Class 2, March 1970 draft All Containment Penetration Isolation Valves ASME Section III, 1971; Draft ASME Pump and Valve Code, 1980, 1983		Service Water Pumps and Valves Standards of the Hydraulic Institute, ANSI G16.5 Class 1 RBCCW Pumps and Valves Standards of the Hydraulic Institute, ANSI B16.1, ANSI B31.1 Auxiliary Feedwater Pumps ASME Code for Pumps and Valves for Nuclear Power, Class II NEMA Standard SM20-1958 Hydraulic Institute
		Chemical and Volume Control System-Concentrated Boric Acid Service-Pumps Acid Service-Pumps and Valves Draft ASME Code for Pump and Valves, Class II, November 1968		
		Containment Spray Pumps and Valves		
Pressurizer Safety Valves	1. ASME Section III, Class A, 1968 Edition, Addenda through summer of 1970. Code Case 1344-1.			
Pressurizer Power Operated Relief Valves	ASME Section III Class 1, 1977 Edition through winter 1979 Addenda			

TABLE 1.2-1 SUMMARY OF CODES AND STANDARDS FOR COMPONENTS OF WATER-COOLED NUCLEAR POWER UNITS [(1)] (CONTINUED)

CODE CLASSIFICATION

Component	Group A	Group B	Group C	Group D
Piping	1. ANSI B 31.7 Class I, 1969 Edition	ANSI B 31.7, Class II 1969 Edition	ANSI B 31.7, Class III 1969 Edition	ANSI B 31.1.0 or Equivalent
	2. ASME Section III, Paragraph N153 in Summer 1969 Addenda 3. Code Case 70 to B31.7			
	Primary Coolant Piping and Surge Line	High Pressure Safety Injection Piping		
		Low Pressure Safety Injection Piping		
	4. Other Reactor Coolant Pressure Pressure Boundary Class I Piping–ASME Section III Code - 1971 Edition, Class I.	Reactor Coolant System Branch Piping beyond Second Isolation Valves Chemical and Volume Control System Concentrated Boric Acid Service Piping ANSI B31.1.0 modified (inside Containment) Containment Spray Piping		Service Water Piping RBCCW Piping
		All Containment Piping Penetrations		
		1. ANSI B-31.1, Piping Code, ANSI B31.7 Nuclear Piping Code, Class I or II as a minimum, 1969 Edition.		
		3. ASME Section III, Class 1 or 2, 1971 Edition		

1 This table summarizes the Codes and Standards used for major pressure retaining components. Not all components are listed. Later codes and standards may be employed for plant modifications if permitted by applicable design and regulatory requirements in effect at the time of the modification.

2 The reactor vessel head and the replacement pressurizer are constructed in accordance with ASME Boiler & Pressure Vessel Code, Section III, Subsection NB 1998 Edition, through 2000 Addenda.

3 Including ASME Code Case N-416.

4 1971 ASME Boiler and Pressure Code, Section III, Class 3.

5 All pressure-retaining cast parts shall be radiographed (or ultrasonically tested to equivalent standards). Where size or configuration does not permit effective volumetric examination, magnetic particle or liquid penetrant examination may be substituted. Examination procedures and acceptance standards shall be at least equivalent to those specified in the applicable class in the code.

[redacted]

FIGURE 1.2–1 SITE LAYOUT

Withhold under 10 CFR 2.390 (d) (1)

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 1.2–2 PLOT PLAN

[redacted]

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 1.2–3 GENERAL ARRANGEMENT, TURBINE BUILDING PLAN AT OPERATING FLOOR ELEVATION 54 FEET 6 INCHES

[redacted]

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 1.2–4 GENERAL ARRANGEMENT, TURBINE BUILDING PLAN AT MEZZANINE FLOOR ELEVATION 31 FEET 6 INCHES

[redacted]

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 1.2–5 GENERAL ARRANGEMENT, TURBINE BUILDING PLAN AT GROUND FLOOR ELEVATION 14 FEET 6 INCHES

[redacted]

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 1.2–6 GENERAL ARRANGEMENT CONTAINMENT PLAN AT FLOOR ELEVATION 14 FEET 6 INCHES AND ELEVATION 36 FEET 6 INCHES

[redacted]

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 1.2–7 GENERAL ARRANGEMENT AUXILIARY BUILDING PLAN AT ELEVATION 36 FEET 6 INCHES AND ELEVATION 38 FEET 6 INCHES

[redacted]

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 1.2–8 GENERAL ARRANGEMENT AUXILIARY BUILDING SECTIONS “G-G” AND “H-H”

[redacted]

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 1.2–9 GENERAL ARRANGEMENT AUXILIARY BUILDING GROUND FLOOR ELEVATION 14 FEET 6 INCHES AND CABLE VAULT ELEVATION 25 FEET 6 INCHES

[redacted]

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 1.2–10 GENERAL ARRANGEMENT CONTAINMENT AND AUXILIARY BUILDING PLAN AT ELEVATION (-)5 FEET 0 INCHES AND ELEVATION (-)3 FEET 6 INCHES

[redacted]

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 1.2–11 GENERAL ARRANGEMENT CONTAINMENT AND AUXILIARY BUILDING PLAN AT ELEVATION (-)25 FEET 6 INCHES AND ELEVATION (-)22 FEET 6 INCHES

[redacted]

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 1.2–12 GENERAL ARRANGEMENT CONTAINMENT AND AUXILIARY BUILDING PLAN AT ELEVATION (-)45 FEET 6 INCHES

[redacted]

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 1.2–13 GENERAL ARRANGEMENT CONTAINMENT AND AUXILIARY BUILDING SECTION "A-A"

[redacted]

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 1.2–14 GENERAL ARRANGEMENT CONTAINMENT AND AUXILIARY BUILDING SECTION "B-B"

[redacted]

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 1.2–15 GENERAL ARRANGEMENT TURBINE BUILDING SECTIONS “C-C” AND “E-E”

[redacted]

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 1.2–16 GENERAL ARRANGEMENT TURBINE BUILDING SECTIONS "D-D" AND "F-F"

[redacted]

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 1.2–17 GENERAL ARRANGEMENT INTAKE STRUCTURE AUXILIARY STEAM BOILER ROOM PLAN AND SECTION

[redacted]

1.3 COMPARISON WITH OTHER PLANTS

Table 1.3-1 presents a summary of the characteristics of the Millstone Unit 2 Nuclear Power Plant at the time of application for operating license. The table includes similar data for Calvert Cliffs Units 1 and 2, Maine Yankee Unit Number 1, Turkey Point Units Numbers. 3 and 4 and Palisades Unit Number 1. Bechtel Corporation and Combustion Engineering (CE), Inc. are identified as contractors in Section 1.6. The Palisades plant is included in the table because its coolant system is similar to that of Millstone Unit 2, because both Bechtel Corporation and CE, Inc. are Palisades contractors and because it is an example of a CE, Inc. nuclear steam supply system which is operating. Calvert Cliffs and Maine Yankee were selected because their cores are similar to that of Millstone Unit 2 and the most contemporaneous plants for which operating licenses have been issued with which CE is associated. Turkey Point is included because it is another comparable plant with which Bechtel Corporation is associated.

TABLE 1.3-1 COMPARISON WITH OTHER PLANTS

HYDRAULIC and THERMAL DESIGN PARAMETERS

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT (1) UNITS 3 AND 4	PALISADES (1) UNIT 1	CALVERT CLIFFS (1) UNITS 1 AND 2	MAINE YANKEE (1)
Total Core Heat Output, MWt	3.5	2,560	2,200	2,200	2,560	2,440
Total Core Heat Output, Btu/hr	3.5	$8{,}737 \times 10^6$	$7{,}479 \times 10^6$	$7{,}509 \times 10^6$	$8{,}740 \times 10^6$	$8{,}328 \times 10^6$
Heat Generated in Fuel, %	3.5	97.5	97.4	97.5	97.5	97.5
Maximum Overpower, %	3.5	12	12	12	12	12
System Pressure, Nominal, psia	3.5	2,250	2,250	2,100	2,250	2,250
System Pressure, Minimum Steady State, psia	3.5	2,200	2,200	2,050	2,200	2,200
Hot Channel Factors, Overall Heat Flux, F_q	3.5	3.00	3.23	3.8	3.00	2.89
Hot Channel Factors, Enthalpy Rise, $F_{\Delta H}$	3.5	1.65	1.77	2.51	1.65	1.62
DNB Ratio at Nominal Conditions	3.5	2.30	1.81	2.00	2.18	2.45
Coolant Flow: Total Flow Rate, lb/hr	3.5	134×10^6	101.5×10^6	125×10^6	122×10^6	122×10^6
Coolant Flow: Effective Flow Rate for Head Transfer, lb/hr	3.5	130×10^6	97.0×10^6	121.25×10^6	117.5×10^6	117.5×10^6
Coolant Flow: Effective Flow Area for Heat Transfer, ft^2	3.5	53.5	41.8	58.7	53.5	53.5
Coolant Flow: Average Velocity along Fuel Rods, ft/sec	3.5	16	14.3	12.7	13.6	13.9
Coolant Flow: Average Mass Velocity, lb/hr-ft^2	3.5	2.4×10^6	2.32×10^6	2.07×10^6	2.20×10^6	2.29×10^6
Coolant Temperatures, °F: Nominal Inlet	3.5	542	546.2	545	543.4	538.9
Coolant Temperatures, °F: Maximum Inlet due to Instrumentation Error and Deadband, °F	3.5	544	550.2	548	548	546
Coolant Temperatures, °F: Average Rise in Vessel, °F	3.5	45	55.9	46	52	51.1
Coolant Temperatures, °F: Average Rise in Core, °F	3.5	46	58.3	47	54	53.1
Coolant Temperatures, °F: Average in Core, °F	3.5	565	575.4	568.5	570.4	565.4
Coolant Temperatures, °F: Average in Vessel	3.5	564	574.2	568	569.5	564.4
Coolant Temperatures, °F: Nominal Outlet of Hot Channel	3.5	640	642	642.8	643	636

TABLE 1.3-1 Comparison with Other Plants (Continued)

HYDRAULIC and THERMAL DESIGN PARAMETERS

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT (1) UNITS 3 AND 4	PALISADES (1) UNIT 1	CALVERT CLIFFS (1) UNITS 1 AND 2	MAINE YANKEE (1)
Average Film Coefficient, Btu/hr-ft^2-F	3.5	5270	5400	4860	5240	5300
Average Film Temperature Difference, °F	3.5	34.5	31.8	30	33.5	33
Heat Transfer at 100% Power: Active Heat Transfer Surface Area, ft^2	3.5	48,400	42,460	51,400	48,416	47,000
Heat Transfer at 100% Power: Average Heat Flux, Btu/hr-ft^2	3.5	176,600	171,600	142,400	176,000	170,200
Heat Transfer at 100% Power: Maximum Heat Flux, Btu/hr-ft^2	3.5	527,800	554,200	541,200	527,900	502,300
Heat Transfer at 100% Power: Average Thermal Output, kw/ft	3.5	5.94	5.5	4.63	5.94	5.74
Heat Transfer at 100% Power: Maximum Thermal Output, kw/ft	3.5	16.6	17.6 (2)	17.6 (2)	17.8	16.9
Maximum Clad Surface Temperature at Nominal Pressure, °F	3.5	657	657	648	657	657
Fuel Center Temperature, °F: Maximum at 100% Power	3.5	3,780	4,030	4,040	3,780	3,640
Fuel Center Temperature, °F: Maximum at Over Power	3.5	4,070	4,300	4,350	4,070	3,940
Thermal Output, kw/ft at Maximum Over Power	3.5	19.6	20.0	19.7 (2)	20.0	19.0

CORE MECHANICAL DESIGN PARAMETERS

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT (1) UNITS 3 AND 4	PALISADES (1) UNIT 1	CALVERT CLIFFS (1) UNITS 1 AND 2	MAINE YANKEE (1)
Fuel Assemblies: Design	3.3	CEA	RCC	Cruciform	CEA	CEA
Fuel Assemblies: Rod Pitch, inches	3.3	0.58	0.563	0.550	0.58	0.580
Fuel Assemblies: Cross-Section Dimensions, inches	3.3	7.98 x 7.98	8.426 x 8.426	8.1135 x 8.1135	7.98 x 7.98	7.98 x 7.98
Fuel Assemblies: Fuel Weight (as UO_2), pounds	3.3	207,035	176,200	210,524	207,269	203,934
Fuel Assemblies: Total Weight, pounds	3.3	282,500	226,200	295,800	282,570	279,235

TABLE 1.3-1 Comparison with Other Plants (Continued)

CORE MECHANICAL DESIGN PARAMETERS

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT [1] UNITS 3 AND 4	PALISADES [1] UNIT 1	CALVERT CLIFFS [1] UNITS 1 AND 2	MAINE YANKEE [1]
Fuel Assemblies: Number of Grids per Assembly	3.3	8	7	8	8	8
Fuel Rods: Number	3.3	36,896	32,028	43,168	36,896	36,352
Fuel Rods: Outside Diameter, inches	3.3	0.44	0.422	0.4135	0.44	0.440
Fuel Rods: Diametral Gap, inches	3.3	0.0085	0.0065	0.0065	0.0085	0.0085
Fuel Rods: Clad Thickness, inches	3.3	0.026	0.0243	0.022	0.026	0.026
Fuel Rods: Clad Material	3.3	Zircaloy	Zircaloy	Zircaloy	Zircaloy	Zircaloy
Fuel Pellets: Material	3.3	UO_2 Sintered	UO_2 Sintered	UO_2 Sintered	UO_2 Sintered	UO_2 Sintered
Fuel Pellets: Diameter, inches	3.3	0.3795	0.367	0.359	0.3795	0.3795
Fuel Pellets: Length, inches	3.3	0.650	0.600	0.600	0.650	0.650
Control Assemblies: Neutron Absorber	3.3	B_4C / S.S. Cd-In-Ag	Cd-In-Ag (5-15-80%)	Cd-In-Ag (5-15-80%) Cruciform	B_4C / S.S. / Cd-In-Ag	B_4C / S.S. / Cd-In-Ag
Control Assemblies: Cladding Material	3.3	NiCrFe Alloy (Inconel 625)	304 SS-Cold Worked Welded to 13.5 inch span	304 SS Tubes, E.B.	NiCrFe Alloy	NiCrFe Alloy
Control Assemblies: Clad Thickness	3.3	0.040	0.109	0.016	0.040	0.040
Control Assemblies: Number of Assembly, full / part length	3.3	73	53	41 / 4 Cruciform Rods	77 / 8	77 / 8
Control Assemblies: Number of Rods per Assembly	3.3	5	20	117 Tubes per Rod	5	5
Core Structure: Core Barrel ID / OD, inches	3.3.2.2	148 / 151.5	133.875 / 137.875	149.75 / 152.5	148 / 149.75	148 / 149.75
Core Structure: Thermal Shield ID / OD, inches	3.3.2.5	156.75 / 162.75	142.625 / 148.0	None	None	156 / 162

TABLE 1.3-1 Comparison with Other Plants (Continued)

NUCLEAR DESIGN DATA

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT (1) UNITS 3 AND 4	PALISADES (1) UNIT 1	CALVERT CLIFFS (1) UNITS 1 AND 2	MAINE YANKEE (1)
Structural Characteristics: Core Diameter, inches (Equivalent)	3.3.1	136	119.5	136.71	136.0	136.0
Structural Characteristics: Core Height, inches (Active Fuel)	3.3.1	136.7	144	132	136.7	136.7
H_2O/U, Unit Cell (Cold)	3.4.1	3.50	4.18	3.50	3.44	3.44
Number of Fuel Assemblies	3.3	217	157	204	217	217
UO_2 Rods per Assembly, Unshimmed / Shimmed		-	204	212 / 208	-	-
UO_2 Rods per Assembly, Unshimmed / Shimmed: Batch A	3.3	176	-	-	176	176
UO_2 Rods per Assembly, Unshimmed / Shimmed: Batch B	3.3	164	-	-	164	160
UO_2 Rods per Assembly, Unshimmed / Shimmed: Batch C	3.3	(176 / 164 / 164)	-	-	(176 / 164 / 164)	(176 / 164 / 160)
Performance Characteristics Loading Technique	3.4.1	3 Batch Mixed Central Zone	3 Regions Non-uniform	3 Batch Mixed Central Zone	3 Batch Mixed Central Zone	3 Batch Mixed Central Zone

TABLE 1.3-1 Comparison with Other Plants (Continued)

NUCLEAR DESIGN DATA

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT [1] UNITS 3 AND 4	PALISADES [1] UNIT 1	CALVERT CLIFFS [1] UNITS 1 AND 2	MAINE YANKEE [1]
Fuel Discharge Burnup, Mwd/MTU: Average First Cycle	3.4.1	12,770	13,000	10,180	13,775	13,795
Fuel Discharge Burnup, Mwd/MTU: First Core Average	3.2.1	22,000	14,500	17,600	22,550	30,000
Feed Enrichment (weight percent): Region 1	3.4.1	1.93	1.85	1.65	2.05	2.01
Feed Enrichment (weight percent): Region 2	3.4.1	2.33	2.55	2.08 / 2.54	2.45	2.40
Feed Enrichment (weight percent): Region 3	3.4.1	2.82	3.10	2.54 / 3.20	2.99	2.95
Feed Enrichment (weight percent): Equilibrium		-	-	2.54 / 3.20	-	-
Control Characteristics Effective Multiplication (beginning of life): Cold, No Power, Clean	3.4.1	1.170	1.180	1.212	1.194	1.170
Control Characteristics Effective Multiplication (beginning of life): Hot, No Power, Clean	3.4.1	1.129	1.38	1.175	1.152	1.129
Control Characteristics Effective Multiplication (beginning of life): Hot, Full Power, Xe Equilibrium	3.4.1	1.078	1.077	1.111	1.094	1.075
Control Assemblies: Material	3.3	B_4C / S.S. Cd-In-Ag	Cd-In-Ag (5-15-80%)	Cd-In-Ag (5-15-80%)	B_4C / S.S.-Cd-In-Ag	B_4C / S.S.-Cd-In-Ag
Control Assemblies: Number of Control Assemblies	3.4.1	73	53	45 Cruciform	85	85
Number of Absorber Rods per RCC (or CEA) Assembly	3.3	5	20	117 Tubes Welded to Form 13.5 inches span	5	5
Total Rod Worth (Hot), % $\Delta\rho$	3.4.1	11.0	7	8.6	$\geq$ 9.6	$\geq$ 9.9
Boron Concentrations – To shut reactor down with no rods inserted, clean, ppm: Cold / Hot, ppm	3.4.1	945 / 935	1,250 / 1,210	1,180 / 1,210	1,120 / 1,095	945 / 935
Boron Concentrations – To shut reactor down with no rods inserted, clean, ppm: To control at power with no rods inserted, clean / equilibrium xenon, ppm	3.4.1	820 / 590	1,000 / 670	1,070 / 830	960 / 725	820 / 590
Kinetic Characteristics, Range Over Life: Moderator Temperature Coefficient [3] $\Delta\rho$/°F	3.4.1	-0.4×10^{-4} to -2.1×10^{-4}	$+.3 \times 10^{-4}$ to -1.96×10^{-4} -3.5	-0.08×10^{-4} to -2.25×10^{-4}	$-.20 \times 10^{-4}$ to -1.96×10^{-4}	-0.40×10^{-4} to -2.20×10^{-4}

TABLE 1.3-1 Comparison with Other Plants (Continued)

NUCLEAR DESIGN DATA

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT (1) UNITS 3 AND 4	PALISADES (1) UNIT 1	CALVERT CLIFFS (1) UNITS 1 AND 2	MAINE YANKEE (1)
Kinetic Characteristics, Range Over Life: Moderator Pressure Coefficient (3) $\Delta\rho$/psi	3.4.1	-0.65×10^{-6} to $+2.39 \times 10^{-6}$	-0.3×10^{-6} to $+3.4 \times 10^{-6}$	$+0.10 \times 10^{-6}$ to $+1.7 \times 10^{-6}$	$+0.65 \times 10^{-6}$ to $+2.39 \times 10^{-6}$	$+0.65 \times 10^{-6}$ to $+2.39 \times 10^{-6}$
Kinetic Characteristics, Range Over Life: Moderator Void Coefficient (3) $\Delta\rho$/% Void	3.4.1	-0.41×10^{-3} to -1.43×10^{-3}	$+0.5 \times 10^{-3}$ to -2.5×10^{-3}	-0.06×10^{-3} to -1.0×10^{-3}	-0.41×10^{-3} to -1.43×10^{-3}	-0.41×10^{-3} to -1.43×10^{-3}
Kinetic Characteristics, Range Over Life: Doppler Coefficient (4) $\Delta\rho$/°F	3.4.1	-1.45×10^{-5} to -1.07×10^{-5}	-1.0×10^{-5} to -1.6×10^{-5}	-1.56×10^{-5} to -1.08×10^{-5}	-1.46×10^{-5} to -1.06×10^{-5}	-1.45×10^{-5} to -1.07×10^{-5}

REACTOR COOLANT SYSTEM - CODE REQUIREMENTS

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT (1) UNITS 3 AND 4	PALISADES (1) UNIT 1	CALVERT CLIFFS (1) UNITS 1 AND 2	MAINE YANKEE (1)
Reactor Vessel	4.2.2	ASME III Class A	ASME III Class A	ASME III Class A	ASME III Class A	ASME III Class A
Steam Generator: Tube Side	4.2.2	ASME III Class A	ASME III Class A	ASME III Class A	ASME III Class A	ASME III Class A
Steam Generator: Shell Side	4.2.2	ASME III Class A	ASME III Class C	ASME III Class A	ASME III Class A	ASME III Class A
Pressurizer	4.2.2	ASME III Class A	ASME III Class A	ASME III Class A	ASME III Class A	ASME III Class A
Pressurizer Relief (or Quench) Tank	4.2.2	ASME III Class C	ASME III Class C	ASME III Class C	ASME III Class C	ASME III Class C
Pressurizer Safety Valves	4.2.2	ASME III	ASME III	ASME III	ASME III	ASME III
Reactor Coolant Piping	4.2.2	ANSI B 31.7	USAS B 31.1	USAS B 31.1	USAS B 31.7	USAS 31.1

PRINCIPAL DESIGN PARAMETERS OF THE COOLING SYSTEM

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT (1) UNITS 3 AND 4	PALISADES (1) UNIT 1	CALVERT CLIFFS (1) UNITS 1 AND 2	MAINE YANKEE (1)
Operating Pressure, psig	4.2.1	2235	2235	2085	2235	2235
Reactor Inlet Temperature, °F	4.2.1	539.7	546.2	545	544.5	540
Reactor Outlet Temperature, °F	4.2.1	595.1	602.1	591.1	599.4	592.8
Number of Loops	4.1	2	3	2	2	3
Design Pressure, psig	4.3.4	2,485	2,485	2,485	2,485	2,485

TABLE 1.3-1 Comparison with Other Plants (Continued)

PRINCIPAL DESIGN PARAMETERS OF THE COOLING SYSTEM

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT [1] UNITS 3 AND 4	PALISADES [1] UNIT 1	CALVERT CLIFFS [1] UNITS 1 AND 2	MAINE YANKEE [1]
Design Temperature, °F	4.3.4	650	650	650	650	650
Hydrostatic Test Pressure (cold), psig	4.2.1	3,110	3,110	3,110	3,110	3,110
Total Coolant Volume, cubic feet	4.2.1	11,101	9,088	10,809	11,101	11,026

PRINCIPAL DESIGN PARAMETERS OF THE REACTOR VESSEL

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT [1] UNITS 3 AND 4	PALISADES [1] UNIT 1	CALVERT CLIFFS [1] UNITS 1 AND 2	MAINE YANKEE [1]
Material	4.3.1, 4.5.6	SA-533, Grade B Class I, low alloy steel plates and SA-508-64, Class 2 forgings, internally clad with Type 304 [5] austenitic SS	SA-302, Grade B, low alloy steel, internally clad with Type 304 austenitic SS	SA-302, Grade B, low alloy steel, internally clad with Type 304 austenitic SS	SA-533, Grade B, Class I, steel, internally clad Type 304 austenitic SS	SA-533, Grade B, forgings-A-508-64 Class 2, cladding weld deposited 304 SS equivalent
Design Pressure, psig	4.3.1	2,485	2,485	2,485	2,485	2,485
Design Temperature, °F	4.3.1	650	650	650	650	650
Operating Pressure, psig	4.2.1	2,235	2,235	2,085	2,235	2,235
Inside Diameter of Shell, inches	4.3.1	172	155.5	172	172	172
Outside Diameter across Nozzles, inches	4.3.1	253	236	254	253	266-5/8
Overall Height of Vessel and Enclosure Head, feet-inches to top of CRD Nozzle	4.3.1	41 feet 11.75 inches	41 feet 6 inches	40 feet 1-13/16 inches	41 feet 11.75 inches	42 feet 1-3/8 inches
Minimum Clad Thickness, inches	4.3.1	1/8	5/32	3/16	1/8	1/8

TABLE 1.3-1 Comparison with Other Plants (Continued)

PRINCIPAL DESIGN PARAMETERS OF THE STEAM GENERATORS

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT (1) UNITS 3 AND 4	PALISADES (1) UNIT 1	CALVERT CLIFFS (1) UNITS 1 AND 2	MAINE YANKEE (1)
Number of Units	4.3.2	2	3	2	2	3
Type	4.3.2	Vertical U-Tube with integral moisture separator	Vertical U-Tube with integral moisture separator	Vertical U-Tube with integral moisture separator	Vertical U-Tube with integral moisture separator	Vertical U-tube with integral moisture separator
Tube Material	4.3.2	Ni-Cr-Fe Alloy	Ni-Cr-Fe-Alloy	Ni-Cr-Fe Alloy	Ni-Cr-Fe Alloy	Ni-Cr-Fe Alloy
Shell Material	4.3.2	SA-533 Gr. B Class 1 and SA-516 gr 70	SA-302	Carbon Steel	SA-533 Gr. B Class 1 and SA-516 gr 70	SA-533 Gr. B Class 1 and SA-516 gr 70
Tube Side Design Pressure, psig	4.3.2	2,485	2,485	2,485	2,485	2,485
Tube Side Design Temperature, °F	4.3.2	650	650	650	650	650
Tube Side Design Flow, lb/hr	4.3.2	61×10^6	33.93×10^6	62.5×10^6	61×10^6	40.67×10^6
Shell Side Design Pressure, psig	4.3.2	1,000	1,085	985	985	985
Shell Side Design Temperature, °F	4.3.2	550	556	550	550	550
Operating Pressure, Tube Side, Nominal, psig	4.3.2	2,235	2,235	2,085	2,235	2,235
Operating Pressure, Shell Side, Maximum, psig	4.3.2	885	1,020	885	885	885
Maximum Moisture at Outlet at Full Load, %	4.3.2	0.2	0.25	0.2	0.2	0.2
Hydrostatic Test Pressure, Tube Side (cold), psig	4.3.2	3,110	3,107	3,110	3,110	3,110
Steam Pressure, psig, at full power	4.3.2	800	730	755	835	800
Steam Temperature, °F, at full power	4.3.2	520.3	510	513.8	525.2	520.3

TABLE 1.3-1 Comparison with Other Plants (Continued)

PRINCIPAL DESIGN PARAMETERS OF THE REACTOR COOLANT PUMP

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT [1] UNITS 3 AND 4	PALISADES [1] UNIT 1	CALVERT CLIFFS [1] UNITS 1 AND 2	MAINE YANKEE [1]
Number of Units	4.3.3	4	3	4	4	3
Type	4.3.3	Vertical, single stage centrifugal with bottom suction and horizontal discharge	Vertical, single stage radial flow with bottom suction and horizontal discharge	Vertical, single stage radial flow with bottom suction and horizontal discharge	Vertical, single stage centrifugal with bottom suction and horizontal discharge	Vertical, single stage centrifugal with bottom suction and horizontal discharge
Design Pressure, psig	4.3.3	2,485	2,485	2,485	2,485	2,485
Design Temperature, °F	4.3.3	650	650	650	650	650
Operating Pressure, nominal psig	4.3.3	2,235	2,235	2,085	2,235	2,235
Suction Temperature, °F	4.3.3	540	546.5	545	543.4	538.9
Design Capacity, gpm	4.3.3	81,200	89,500	83,000	81,200	108,000
Design Head, feet	4.3.3	243	260	260	300	290
Hydrostatic Test Pressure, (cold), psig	4.3.3	3,110	3,107	3,110	3,110	3,110
Motor Type	4.3.3	AC Induction	AC Induction	AC Induction	AC Induction	AC Induction
	4.3.3	Single Speed	Single Speed	Single Speed	Single Speed	Single Speed
Motor Rating, hp	4.3.3	6,500	6,000	6,250	7,200	9,000

PRINCIPAL DESIGN PARAMETERS OF THE REACTOR COOLANT PIPING

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT [1] UNITS 3 AND 4	PALISADES [1] UNIT 1	CALVERT CLIFFS [1] UNITS 1 AND 2	MAINE YANKEE [1]
Material	4.3.4	SA516 - GR 70 with minimum 1/8 304L SS clad	Austenitic SS	SA212B clad with SS	SA516 - gr 70 with nominal 7/32 SS clad	SA516 - gr 70 with SS clad
Hot Leg - ID, inches	4.3.4	42	29	42	42	33.5
Cold Leg - ID, inches	4.3.4	30	27.5	30	30	33.5
Between Pump and Steam Generator - ID, inches	4.3.4	30	31	30	30	33.5

TABLE 1.3-1 Comparison with Other Plants (Continued)

CONTAINMENT SYSTEM PARAMETERS

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT [1] UNITS 3 AND 4	PALISADES [1] UNIT 1	CALVERT CLIFFS [1] UNITS 1 AND 2	MAINE YANKEE [1]
Type	5.2.1	Double containment with steel lined prestressed post tensioned concrete cylinder, curved dome roof completely enclosed by Enclosure Building	Steel lined prestressed post tensioned concrete cylinder, shallow dome roof	Steel lined prestressed post tensioned concrete cylinder, curved dome roof	Steel lined prestressed post tensioned concrete cylinder, curved dome roof	Steel lined reinforced concrete flat bottom and hemispherical dome
Containment Parameters: Inside Diameter, feet	5.2.1	130	116	116	130	135
Containment Parameters: Height, feet.	5.2.1	175	169	190.5	181-2/3	169.5
Containment Parameters: Free Volume, ft^3	5.2.1	1,920,000 [5]	1,550,000	1,640,000	2,000,000	1,855,000
Containment Parameters: Reference Incident Pressure, psig	5.2.1	54	59	55	50	55
Containment Parameters: Concrete Thickness, feet						
Containment Parameters: Vertical Wall	5.2.1	3.75	3.75	3	3.75	4.5
Containment Parameters: Dome	5.2.1	3.25	3.25	2.5	3.25	2.5
Containment Leak Prevention and Mitigation Systems	6.7.2.1	Completely enclosed containment has leaktight penetrations and continuous steel liner. Enclosure Building Filtration region at small negative pressure during LCI. Automatic isolation where required. The exhaust from filtration region passed through charcoal filters to 375 feet Millstone stack following incident.	Leak tight penetration and continuous steel liner, automatic isolation where required	Leak tight penetration and continuous steel liner, automatic isolation where required	Leak tight penetration and continuous steel liner, automatic isolation where required. The exhaust from penetration rooms to vent.	Leak tight penetration and continuous steel liner, automatic isolation where required
Gaseous Effluent Purge	11.1.2.1.3	Discharge through Unit 2 stack	Through particulate filter & monitors part of main exhaust system	Discharge through stack	Discharge through vent	Discharge through stack

TABLE 1.3-1 Comparison with Other Plants (Continued)

ENGINEERED SAFEGUARDS

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT [(1)] UNITS 3 AND 4	PALISADES [(1)] UNIT 1	CALVERT CLIFFS [(1)] UNITS 1 AND 2	MAINE YANKEE [(1)]
Safety Injection System: Number of High Head Pumps	6.3.2.1	3	4 (shared)	3	3	3 (charging)
Safety Injection System: Number of Low Head Pumps	6.3.2.1	2	2	2	2	2
Safety Injection System: Safety Injection Tank, number	6.3.2.1	4	3	4	4	3
Containment Fan Coolers: Number of Units	6.5.1.2	4	3	4	4	6
Containment Fan Coolers: Air Flow capacity, each at emergency condition, cfm	6.5.2.2	34,800	65,000	30,000	55,000	N/A
Post-Incident Filters Inside Containment: Number of Units		None	None	None	None	None
Post-Incident Filters Inside Containment: Type		None	None	None	None	None
Containment Spray Number of Pumps	6.4.2.1	2	2	-	2	3
Emergency Power Diesel Generator Units	8.3.1.1	2	2 total for both units	2	3 total for both units	2
Enclosure Building Filtration System Number of Units	6.7.2.1	2	-	-	-	0

RADIOACTIVE WASTE PROCESSING SYSTEMS

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT [(1)] UNITS 3 AND 4	PALISADES [(1)] UNIT 1	CALVERT CLIFFS [(1)] UNITS 1 AND 2	MAINE YANKEE [(1)]
Design Failed Fuel, %	11.1.1.1	1	1	1	1	1
Gaseous Waste Processing System	11.1.2.1					
Annual Volume of Gases Discharge, ft^3	11.1.2.1	14,344	[(6)]	4,539	66,240	[(6)]
Annual Activity Discharge, Curies	11.1.2.1	556	14,758	[(6)]	[6)]	[(6)]
Decay Storage Time for Gases, Days	11.1.2.1	60 (Minimum)	45	30 (Minimum)	60	[(6)]
Compressors: Number		2	2 [(7)]	2	2 [(7)]	2

TABLE 1.3-1 Comparison with Other Plants (Continued)

RADIOACTIVE WASTE PROCESSING SYSTEMS

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT (1) UNITS 3 AND 4	PALISADES (1) UNIT 1	CALVERT CLIFFS (1) UNITS 1 AND 2	MAINE YANKEE (1)
Compressors: Capacity, each	11.1.2.2	25 SCFM	40 CFM	2.35 SCFM	4 to 7 SCFM	(6)
Decay Tanks: Number		6	6 (7)	3	3 (7)	3
Decay Tanks: Capacity, (each), ft^3		Specified 582 cuft As Built 627 cuft	525	100	610	200

LIQUID WASTE PROCESSING SYSTEMS

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT (1) UNITS 3 AND 4	PALISADES (1) UNIT 1	CALVERT CLIFFS (1) UNITS 1 AND 2	MAINE YANKEE (1)
Clean Liquid Waste (Reactor Coolant Wastes)	11.1.2.1					
Design Volume Wastes per Year	11.1.2.1	14 Reactor Coolant System Volumes (840,000 Gallons)		(6)	14 Reactor Coolant System	(6)
Expected Volume of Waste Discharge Per Year, Gallons	11.1.2.1	404,234	(Design Incorporates Recycle of Waste to R.C. System Clean Liquid Waste System Not Compared)	724,300	805,542	(6)
Annual Expected Activity Discharged, curies	11.1.2.1	286 (includes H^3)		(6)	(6)	(6)
Percentage of 10 CFR Part 20	11.1.4.1	0.6%		(6)	(6)	(6)
Degasifier: Number	11.1.2.2	1	1	2	2	
Degasifier: Type	11.1.2.2	Packed Column Utilizing Internal Generated Stripping Steam	Vacuum	Packed Tower	Flashing	
Degasifier: Design Flow Rate, gpm	11.1.2.2	132	160	120	100	
Degasifier: Decontamination Factors	11.1.2.2	1,000 (Kr & Xe)	40	10	(6)	
Storage Tanks: Number	11.1.2.2	4		4	4	2
Storage Tanks: Total Capacity	11.1.2.2	3 Reactor Coolant System (180,000 Gallons)		200,000 Gallons	6 Reactor Coolant System Volumes (7)	250,000 Gallons
Storage Tanks: Vent Discharge	11.1.2.2	To Gaseous Waste System for storage and decay		To Exhaust Plenum	Plant Vent	To ventilation System and stack
Demineralizers: Number	11.1.2.2	3		3	4	2

TABLE 1.3-1 Comparison with Other Plants (Continued)

LIQUID WASTE PROCESSING SYSTEMS

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT (1) UNITS 3 AND 4	PALISADES (1) UNIT 1	CALVERT CLIFFS (1) UNITS 1 AND 2	MAINE YANKEE (1)
Demineralizers: Type	11.1.2.2	Mixed Bed Non Regenerative		Mixed bed	Mixed Bed Non Regenerative	Cesium Removal
Demineralizers: Decontamination	11.1.2.2	1,000		10	100	(6)
Demineralizers: Factors	11.1.2.2			(0 for Y, Mo, H^3)		
Evaporator (Boron Recovery): Number	11.1.2.2	1		N/A	2	1
Evaporator (Boron Recovery): Type	11.1.2.2	Vacuum, Submerged U-Tube			Horizontal Spray Film	Forced Calculating, Single Effect
Evaporator (Boron Recovery): Capacity, GPM Distillate	11.1.2.2	25			20	30
Evaporator (Boron Recovery): Decontamination	11.1.2.2				10^5 (Nonvolatiles)	(6)
Evaporator (Boron Recovery): Factors	11.1.2.2	1,000 (Nonvolatiles), 50 (Halogens), 100 (Dissolved Gases)			10^4 (Gases)	

Aerated Liquid Waste Processing System (Miscellaneous Wastes)

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT (1) UNITS 3 AND 4	PALISADES (1) UNIT 1	CALVERT CLIFFS (1) UNITS 1 AND 2	MAINE YANKEE (1)
Design Volume of Waste per year	11.1.2.1	3,639,400 (Gallons)	(6)	(6)	(6)	(6)
Expected Volume of Waste Discharged per year, Gallons	11.1.2.1	313,000	508,620	(6)	1,330,320	(6)
Annual Expected Activity Discharged, Curies	11.1.2.1	1.11 (includes H^3)	0.077	(6)	(6)	(6)
Percentage of 10 CFR Part 20	11.1.4.1	Less than 0.1%				
Storage Tanks: Number	11.1.2.2	1	2	1	2	2
Storage Tanks: Total Capacity	11.1.2.2	5,000 Gallons	2,000 Gallons	5,500 Gallons	8,000 Gallons	24,800 Gallons
Demineralizers: Number	11.1.2.2	1	(6)	N/A	1	N/A
Demineralizers: Type	11.1.2.2	Mixed Bed Non Regenerative	(6)		Mixed Bed Non Regenerative	
Demineralizers: Decontamination Factors	11.1.2.2	500	(6)		100	

TABLE 1.3-1 Comparison with Other Plants (Continued)

Aerated Liquid Waste Processing System (Miscellaneous Wastes)

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT (1) UNITS 3 AND 4	PALISADES (1) UNIT 1	CALVERT CLIFFS (1) UNITS 1 AND 2	MAINE YANKEE (1)
Evaporator:	11.1.2.2	N/A		N/A	N/A	
Evaporator: Number	11.1.2.2		1 (7)			1
Evaporator: Type	11.1.2.2		(6)			(6)
Evaporator: Capacity, Distillate GPM	11.1.2.2		(6)			(6)
Evaporator: Decontamination Factors	11.1.2.2		10^6		(6)	(6)

Solid Waste Processing System

<Parameter>	REFERENCE SECTION	CYCLE 1 MILLSTONE UNIT 2	TURKEY POINT (1) UNITS 3 AND 4	PALISADES (1) UNIT 1	CALVERT CLIFFS (1) UNITS 1 AND 2	MAINE YANKEE (1)
Evaporator Concentrates	11.1.2.1	Solidified in Concrete in 55 Gallon drums	Solidified in Concrete in 55 Gallon drums	N/A	Solidified in Concrete in 55 Gallon drums	55 Gallon drums
Spent Resins Shipping & Volumes	11.1.2.1	Shipping cask after dewatering, 225 ft^3	Dewatered 55 Gallon Drums	(6)	Solidified in Concrete in 55 Gallon Drums	Shipping cask
Contaminated Filter Cartridges & Volumes	11.1.2.1	55 Gallon drums	55 Gallon drums	55 Gallon drums	Solidified in Concrete in 55 Gallon Drums	Cask or 55 Gallon drums
Annual Activity Shipped, curies	11.1.2.1	4,250	(6)	(6)	(6)	(6)

1 The values listed for these plants were taken from public documentation.
2 Based on total heat output of the core rather than heat generated in the fuel alone.
3 Values shown are for beginning of life full power / end of cycle full power.
4 Values shown are for beginning of life zero power/beginning of life cycle full power.
5 Measured value from pre-operational volume verification test and used for integrated leak rate testing. Includes volume of vented pressurizer, safety injection tanks, and other tanks.
6 Not Specifically Available in Public Documents.
7 Shared by Two (2) Units.

1.4 PRINCIPAL ARCHITECTURAL AND ENGINEERING CRITERIA FOR DESIGN

The principal architectural and engineering features used in the design of Unit 2 of the Millstone Nuclear Power Station are summarized in the following material.

1.4.1 PLANT DESIGN

Principal structures and equipment which may serve either to prevent accidents or to mitigate their consequences have been designed, fabricated and erected in accordance with applicable codes so as to withstand the most severe earthquakes, flooding conditions, windstorms, ice conditions, temperature and other deleterious natural phenomena which could be reasonably assumed to occur at the site during the lifetime of this plant. Systems and components designed for Seismic Category I requirements are listed in Table 1.4-1. It should be noted that the terms ‘Category’ and ‘Class’ are used interchangeably throughout the MP2 FSAR in defining seismic design classifications of Structures, Systems and Components. Unit 2 was designed so that the safety of one unit will not be impaired in the unlikely event of an accident in the other unit. Principal structures and equipment were sized for the maximum expected nuclear steam supply system (NSSS) and turbine outputs.

Redundancy is provided in the reactor and safety systems so that the single failure of any active component of either system cannot prevent the action necessary to avoid an unsafe condition. The unit is designed to facilitate inspection and testing of systems and components whose reliabilities are important to the protection of the public and plant personnel.

Provisions have been made to protect against the hazards of such events as fires or explosions.

Systems and components which are significant from the standpoint of nuclear safety are designed, fabricated and erected to quality standards commensurate with the safety function to be performed. Appendix 1.A of this FSAR addresses the implementation of Atomic Energy Commission (AEC) General Design Criteria for Nuclear Power Plants, 10 CFR Part 50, Appendix A. Section 12.8 describes the Quality Assurance Program.

1.4.2 REACTOR

The following criteria (see Chapter 3) apply to the reactor:

a. The reactor is of the pressurized water-type, designed to provide heat to steam generators which, in turn, provide steam to drive a turbine generator. The initial full power core thermal output was 2560 megawatts (the NSSS rating was 2570 megawatts) prior to its uprating to the current 2700 megawatts thermal power level (NSSS rating of 2715 megawatts).

b. The reactor is refueled with slightly enriched uranium dioxide contained in zirconium alloy tubes.

c. Minimum departure from nucleate boiling ratio during normal operation and anticipated transients will not be below that value which could lead to fuel rod failure or damage. The maximum fuel centerline temperature evaluated at the design overpower condition will be below that value which could lead to fuel rod failure. The melting point of the UO_2 will not be reached during routine operation and anticipated transients.

d. Fuel rod clad is designed to maintain cladding integrity throughout fuel life. Fission gas release within the rods and other factors affecting design life will be considered for the maximum expected exposures.

e. The reactor and control systems are designed so that any xenon transients can be adequately damped.

f. The reactor is designed to accommodate the anticipated transients safely and without fuel damage.

g. The reactor coolant system (RCS) is designed and constructed to maintain its integrity throughout the expected plant life. Appropriate means of test and inspection are provided.

h. Power excursions which could result from any credible reactivity addition accident will not cause damage, either by deformation or rupture, to the pressure vessel or impair operation of the engineered safety features (ESF).

i. Control element assemblies (CEA) are capable of holding the core subcritical at hot zero power conditions following a trip, and providing a safety margin even with the most reactive CEA stuck in the full, withdrawn position.

j. The chemical and volume control system (CVCS) can add boric acid to the reactor coolant at a sufficient rate to maintain an adequate shutdown margin when the RCS is cooling down following a reactor trip. This is accomplished at a maximum design rate. This system is independent of the CEA system.

k. The combined response of the fuel temperature coefficient, the moderator temperature coefficient, the moderator void coefficient and the moderator pressure coefficient to an increase in reactor thermal power is a decrease in reactivity. In addition, the reactor power transient remains bounded and damped in response to any expected changes in any operating variable.

1.4.3 REACTOR COOLANT AND AUXILIARY SYSTEMS

1.4.3.1 Reactor Coolant System

The design bases in this section are those used for the integrated design of the RCS or those which apply to all of the system components. The design bases unique to each component are discussed in Section 4.3.

The RCS is designed for the normal operation of transferring 2715 MWt (9.26 x 10 Btu/hr) from the reactor core (2700 MWt) and reactor coolant pumps (15 MWt) to the steam generators. In the steam generator, this heat is transferred to the secondary system forming 5.9 x 10^6 lb/hr of 880 psia saturated steam per generator with a 2.0 percent maximum moisture content.

The RCS is designed to accommodate the normal design transients listed. These transients include conservative estimates of the operational requirements of the systems and are used to make the required component fatigue analyses.

a. 500 heatup and cooldown cycles at a maximum heating and cooling rate of 100°F/hr. The pressurizer is designed for a maximum cooldown rate of 200°F/hr.

b. Pressurizer spray piping is limited to 160 plant heatup and cooldown cycles. Primary manway studs of the replaced steam generators are limited to 200 heatup and cooldown cycles.

c. 15,000 power change cycles in the range between 15 and 100 percent of full load with a ramp load change of 5 percent of full load per minute increasing or decreasing. This will occur without reactor trip.

d. Primary manway studs for the replaced steam generators are limited to 1,000 cycles with a ramp load change of 5% per minute decreasing and 30% per hour increasing (plant loading/unloading).

e. 2,000 step power changes of 10 percent, both increasing and decreasing between 15 and 100 percent of full load. Primary manway studs for the replaced steam generator are limited to 1,500 step power changes.

f. 10 cycles of hydrostatic testing at 3,110 psig and a temperature at least 60°F above the nil ductility transition temperature (NDTT) of the component having the highest NDTT.

g. 200 cycles of leak testing at 2,485 psig and a temperature at least 60°F greater than the NDTT of the component with the highest NDDT.

h. Primary manway studs for the replaced steam generators are limited to 80 cycles of leak testing at 2,485 psig.

i. 10^6 cycles of operating pressure variations of ±100 psi from the normal 2,235 psig operating pressure and ±6°F at operating temperature and pressure.

j. 400 reactor trips when at 100 percent power. Primary manway studs for the replaced steam generator are limited to 200 reactor trips when at 100% power.

In addition to these normal design transients, the following abnormal transients are also considered to arrive at a satisfactory usage factor as defined in Section III, Nuclear Vessels, of the ASME Boiler and Pressure Vessel Code:

a. 40 cycles of loss of turbine load from 100 percent power.

b. 40 cycles of loss of reactor coolant flow when at 100 percent.

c. 5 cycles of loss of main steam system pressure.

Components of the RCS are designed and will be operated so that no deleterious pressure or thermal stress will be imposed on the structural materials. The necessary consideration has been given to the ductile characteristics of the materials at low temperature.

1.4.3.2 Chemical and Volume Control System

The major functions of the CVCS (see Section 9.2) are to:

a. Maintain the required volume of water in the RCS.

b. Maintain the chemistry and purity of the reactor coolant.

c. Maintain the desired boric acid concentration in the reactor coolant.

d. Provide a controlled path to the waste processing system.

The system is designed to accept the discharge when the reactor coolant is heated at the design rate of 100°F/hr and to provide the required makeup when the reactor coolant is cooled at the design rate of 100°F/hr. Discharge is automatically diverted to the waste processing system when the volume control tank is at its highest permissible level. The system will also supply makeup or accept discharge due to power decreases or increases. The design transients are ±10 percent of full power step changes and ramp changes of ±5 percent of full power per minute between 15 to 100 percent power. On power increases, the letdown flow is automatically diverted to the waste processing system when the volume control tank reaches the highest permissible level. On power decreases, sufficient coolant is in the volume control tank to allow a full to zero power decrease without additional makeup, in the event of a makeup system failure or override.

For an assumed 1 percent failed fuel condition, the activity in the reactor coolant does not exceed 411 µCi/cc at 77°F. The system is also designed to maintain the reactor coolant chemistry within the limits specified in Section 4.4.3.

The rate of boron addition is sufficient to counteract the maximum reactivity increase due to cooldown and xenon decay. Any one of the three charging pumps is capable of injecting the required boron (as boric acid). The maximum rate at which the reactor coolant boron concentration can be reduced must be substantially less than the equivalent maximum rate of reactivity insertion by the CEA.

Prior to refueling, the system is capable of increasing the reactor coolant boron concentration from zero to 1720 ppm by feed and bleed when the reactor coolant is at hot standby operating temperature.

Provisions to facilitate the plant hydrostatic testing and to leak test the RCS are included.

1.4.3.3 Shutdown Cooling System

The shutdown cooling system (see Section 9.3) is designed to cool the RCS from approximately 300° to 130°F in 24 hours, assuming that the component cooling water inlet temperature is at its maximum design value of 95°F. The design RCS cooldown rate is 100°F/hr. A temperature of 130°F or less can be achieved 27.5 hours after reactor shutdown, assuming an infinitely exposed core. The maximum allowable pressure for the RCS during shutdown cooling is approximately 285 psig.

1.4.4 CONTAINMENT SYSTEM

The containment (see Sections 5.2 and 14.8), including the associated access openings and penetrations, is designed to contain pressures and temperatures resulting from a postulated main steamline break (MSLB) in which:

a. A range of power level, break sizes, and single failures are considered.

b. Cases with the loss of offsite power and with AC power available are analyzed to determine which scenario maximizes the energy removal into containment.

c. Safety injection is not assumed since it would tend to reduce the energy released into containment.

d. The containment air recirculation cooling system and the containment spray system are credited to mitigate the containment pressure and temperature consequences.

Containment response to a loss-of-coolant (LOCA) accident was also analyzed. It was found that the peak containment pressure and temperature of the MSLB accident bound the LOCA.

The containment is designed to assure integrity against postulated missiles from equipment failures and against postulated missiles from external sources.

Means are provided for pressure and leak rate testing of the containment system. This includes provisions for leak rate testing of individual piping and electrical penetrations that rely on gestated seals, sealing compounds, expansion bellows, and the interior of the containment.

The enclosure building (see Section 5.3) is designed to withstand a wind loading of 115 mph, with gusts of 140 mph, snow load of 60 psf and seismic loads. The Enclosure Building is designed so that is structural framing will withstand tornado loads, but the siding will be blown away (see Section 5.3.3).

1.4.5 ENGINEERED SAFETY FEATURES SYSTEMS

The design incorporates redundant independent full capacity engineered safety features systems (ESFS). These, in conjunction with the containment, ensure that the release of fission products, following any postulated occurrence, at least the minimum ESF required to terminate that occurrence are operable. The following are required as minimum safety features:

One high pressure safety injection (HPSI) train

One low pressure safety injection (LPSI) train

Four safety injection tanks (water quantity of three is required to reach the core)

One containment spray and two containment air recirculation and cooling subsystems, or equivalent (Section 6.4)

One hydrogen control subsystem

One enclosure building filtration train

One auxiliary feedwater trains

Each of these subsystems is independent of its redundant counterpart with the exception of the safety injection subsystems. The HPSI and LPSI subsystems (Section 6.3) are independent up to the common pipe connections to the four reactor coolant cold legs. Remote manually operated valves provide appropriate cross-connections between redundant subsystems for backup and to allow maintenance. Redundant components are physically separated.

The ESFS are designed to perform their functions for all break sizes in the RCS piping up to and including the double-ended rupture of the largest reactor coolant pipe. The safety injection system limits fuel and cladding damage to an amount which will not interfere with adequate emergency core cooling and holds metal-water reactions to minimal amounts. Two full capacity systems, based on different principles remove heat from the containment to maintain containment integrity, the containment spray system (Section 6.4) and the containment air recirculation and cooling system (Section 6.5). The enclosure building filtration system (EBFS) (Section 6.7) maintains the enclosure building filtration region (EBFR) at a slightly negative pressure and filters the exhaust from this space. The containment postaccident hydrogen control system (Section 6.6) mixes and monitors the accumulation of hydrogen gases within the containment. Purge and recombiners are not credited for any mitigating function.

1.4.6 PROTECTION, CONTROL AND INSTRUMENTATION SYSTEM

A reactor protective system (RPS) (see Section 7.2) is provided which initiates reactor trip if the reactor approaches an unsafe condition.

Interlocks and automatic protective systems are provided along with administrative controls to ensure safe operation of the plant.

Sufficient redundancy is installed to permit periodic testing of the RPS so that failure or removal from service of any one protective system component or portion of the system will not preclude reactor trip or other safety action when required.

The protective system is isolated from the control instrumentation systems so that failure or removal from service of any control instrumentation system component or channel does not inhibit the function of the protective system.

1.4.7 ELECTRICAL SYSTEMS

Normal, reserve and emergency sources of auxiliary electrical power are provided to assure safe and orderly shutdown of the plant and to maintain a safe shutdown condition under all credible circumstances. Onsite electrical power sources and systems are designed to provide dependability, independence, redundancy and testability in accordance with the requirements of 10 CFR Part 50, Appendix A. The load-carrying capability and other electrical and mechanical characteristics of emergency power systems are in accordance with the requirements of Safety Guide Number 9. Two redundant, independent, full capacity emergency power sources and distribution subsystems are provided. Each of these subsystems powers all equipment in the associated safety related subsystems as described in Section 1.4.5.

1.4.8 RADIOACTIVE WASTE PROCESSING SYSTEM

The radioactive waste processing system (see Section 11.1) is designed so that discharges of radioactivity to the environment are minimized and are in accordance with the requirements of Sections 1301 and 1302 and Appendix B of 10 CFR Part 20 and Appendix I of 10 CFR Part 50.

1.4.9 RADIATION PROTECTION

Millstone Unit 2 is provided with a centralized control room which has adequate shielding (see Section 11.2.2.3) and ventilation system features (see Section 9.9.10) to permit occupancy during all postulated accidents involving radiation releases.

The radiation shielding in Millstone Unit 2 and the radiation control procedures ensure that operating personnel do not receive exposures during normal operation and maintenance in excess of the applicable limits of 10 CFR Part 20.

1.4.10 FUEL HANDLING AND STORAGE

Fuel handling and storage facilities (see Section 9.8) are provided for the safe handling and storage of fuel. The design precludes accidental criticality.

TABLE 1.4-1 SEISMIC CLASS I SYSTEMS AND COMPONENTS

System	Components
Safety Injection System	HPSI pumps and motors
	LPSI pumps and motors
	Safety Injection Tanks
	Refueling Water Storage Tank
	Piping and supports
	Valves and valve operators
Containment Spray System	Containment spray pumps and motors
	Shutdown cooling heat exchangers
	Refueling water storage tank
	Piping and supports
	Valves and valve operators
	Containment sump screen
Containment Air Recirculation and Cooling System	Fans and motors
	Cooling Coils
	Housing
Enclosure Building Filtration System and Emergency Spent Fuel Pool Cleanup	Fans and motors
	Filters and housing
	Electric heaters
	Piping, ductwork and supports
	Dampers and damper operators
Hydrogen Control System	Hydrogen recombiners
	PIR fans and motors
	Piping and supports
	Hydrogen purge valves and valve operators
	Hydrogen monitoring system

TABLE 1.4-1 SEISMIC CLASS I SYSTEMS AND COMPONENTS (CONTINUED)

System	Components
Control Room Air Conditioning System (including the control room filtration system)	Fans and motors
	Direct expansion and condenser coils
	Housings
	Compressor
	CRFS Filters
	Ductwork and supports
	Dampers and damper operators
	Refrigeration piping and supports
	Refrigerant valves and valve operators
	Temperature control system
	Control Panels
Engineered Safety Feature Room Air Recirculation System	Fans and motors
	Cooling coils
	Ductwork and supports
	Dampers and damper operators
Diesel Generator Ventilation System	Fans and motors
	Ductwork and supports
	Dampers
Vital Switchgear Ventilation System	Fans and Motors
	Cooling Coils
	Chillers and control panels
	Pumps and motors
	Piping; valves and supports
	Ductwork and supports
	Dampers and Damper Operators
Containment Isolation System	Piping and sleeves
	Valves and valve operators

TABLE 1.4-1 SEISMIC CLASS I SYSTEMS AND COMPONENTS (CONTINUED)

System	Components
Electrical Power Supply System	Station batteries, racks and chargers
	125 VDC Switchgear
	DC/AC Inverters
	Vital AC and DC distribution panels
	4160 Volt Emergency Switchgear
	480 Volt Emergency Load Centers
	480 Volt Emergency Motor Control Centers
Electrical Distribution System	Vital tray system and supports
	Vital underground duct banks
	Penetration assemblies
Reactor Coolant System	Reactor vessel and internals
	Control element assemblies and drives
	Pressurizer
	Reactor coolant pumps and motors
	Reactor coolant piping
	Pressurizer surge line and supports
	Pressurizer safety and relief valves
	Steam generators
	Vent, sampling and drain piping, supports and valves up to and including second isolation valve
	Quench tank *
	Pressurizer safety and relief valves
	piping and supports to quench tank *
	Reactor coolant pump supports

TABLE 1.4-1 SEISMIC CLASS I SYSTEMS AND COMPONENTS (CONTINUED)

System	Components
Chemical and Volume Control System	Boric acid storage tanks
	Boric acid pumps and drivers
	Boric acid piping supports and valves
	Charging pumps and drivers
	Charging line piping, supports, valves and pulsation dampeners
	Letdown line piping, supports and valves up to and including second isolation valve
	Regenerative Heat exchanger
	Letdown heat exchanger *
	Letdown line piping, supports, and valves downstream of reactor coolant system isolation valves *
	Letdown filters *
	Ion exchangers *
	Volume control tank *
Spent Fuel Pool Cooling System	Piping, supports and valves between spent fuel pool and shutdown heat exchangers
	Spent fuel pool cooling pumps
	Spent fuel pool heat exchangers
	Spent fuel pool cooling pump drivers *
	Piping, supports, and valves associated with normal spent fuel cooling (up to and including pipe support beyond isolation valve on branch lines) *
Gaseous Waste Processing System	Waste gas decay tanks *
	Waste gas compressors *
	Waste gas filter *
	High pressure (150 psig) service piping, supports, and valves *

TABLE 1.4-1 SEISMIC CLASS I SYSTEMS AND COMPONENTS (CONTINUED)

System	Components
Fuel and Reactor Component Handling Equipment	Containment polar crane
	Spent fuel cask crane
	Spent fuel platform crane *
	Refueling machine *
	Fuel transfer machine *
	Fuel tilting mechanisms *
	Fuel transfer tube and isolation valve
	New and spent fuel storage racks
	New fuel elevator *
	Spent fuel inspection machine *
RBCCW System	RBCCW Pumps and Motors
	RBCCW Heat Exchangers
	RBCCW Surge Tank
	Piping and Supports
	Expansion Joints
	Valves and Valve Operators
Service Water System	Pumps and Drivers
	Piping and Supports
	Valves and Valve Operators
	Service Water Strainers
Emergency Diesel Generators Diesel Oil System	Air Intake and Exhaust Piping
	Control Panels
	Diesel Oil Supply Tanks
	Piping, Valves and Supports
Lube Oil System	Pumps and motors
	Coolers
	Piping and supports
	Heaters
	Piping and supports

TABLE 1.4-1 SEISMIC CLASS I SYSTEMS AND COMPONENTS (CONTINUED)

System	Components
Jacket Water Cooling System	Pumps and motors
	Coolers
	Piping and supports
	Heaters
	Jacket water expansion tank
	Valves and valve operators
*Designated seismic Class II components but designed for Class I earthquake basis.	
Air Cooling System	Pumps
	Coolers
	Piping and supports
	Valve and valve operators
Starting Air System	AC and DC Motor Driven Compressors
	Starting Air tanks
	Piping and supports upstream of check valves
Auxiliary Feedwater System	Auxiliary. feedwater pumps and drivers
	Condensate storage tank
	Piping and supports
	Valves and valve operators
Main Steam System (Upstream of isolation valves	Main steam safety relief valves
	Atmospheric dump valves
	Main Steam isolation valves
	Piping and supports
	Valves and valve operators
Engineered Safety Actuation System, Status Panel	
Reactor Protection System	
Seismic Measurement Instrumentation	
Main Control Boards	
Main Steam Isolation Panel	

TABLE 1.4-1 SEISMIC CLASS I SYSTEMS AND COMPONENTS (CONTINUED)

System	Components
Hot Shutdown Control Boards	
Boric Acid Heat Tracing Panels	
Radiation Monitoring System	

* Designated seismic Class II components but designed for Class I earthquake basis.

1.5 RESEARCH AND DEVELOPMENT REQUIREMENTS

1.5.1 GENERAL

The design of Millstone Unit 2 is based upon concepts which have been successfully applied in the design of other pressurized water reactor power plants. However, certain programs of theoretical analysis or experimentation (constituting “research and development” as defined in the Atomic Energy Act, as amended, and in Nuclear Regulatory Commission (NRC) Regulations) have been undertaken to aid in plant design and to verify the performance characteristics of the plant components and systems. This section describes the results and status of these analytical and test programs, including experimental production and testing of models, devices, equipment and materials at time of application for an operating license.

Combustion Engineering (CE), Inc., which conducted these programs, had taken into consideration information derived from research and development activities of the NRC and other organizations in the nuclear industry.

All CE research and development programs required to justify the design to Millstone Unit 2 were completed and all test results were factored into design of the plant.

1.5.2 FUEL ASSEMBLY FLOW MIXING TESTS

In 1966, a series of single-phase tests on coolant turbulent mixing was run on a prototype fuel assembly which was geometrically similar to the Palisades assembly. The model enabled determination of flow resistance and vertical subchannel flow rates using pressure instrumentation and the average level of eddy flow using dye-injection and sampling equipment. The tests yielded the value of the inverse Peclet number characteristic of eddy flow (0.00366). During the course of the tests the value was shown to be insensitive to coolant temperature and to vertical coolant mass velocity. The design value of the inverse Peclet Number was established as 0.0035 on the basis of the experimental results.

As part of a CE sponsored research and development program, a new series of single-phase dye injection mixing tests were conducted in 1968. The tests were performed on a model of a portion of control element assembly (CEA) type fuel assembly which was sufficiently instrumented to enable measurement, via a data reduction computer program, of the individual lateral flows across the boundaries of 12 subchannels of the model. Although these tests were not intended for that purpose, some of the test results could be used to determine the average level of turbulent mixing in the reference design assembly. The inverse Peclet Number calculated from the average of 56 individual turbulent missing flows (two for each subchannel boundary) obtained from the applicable data was 0.0034. With respect to general turbulent mixing, therefore, the more recent study on the CEA verifies the constancy of the inverse Peclet number for moderately different fuel assembly geometries and confirms the design value of that characteristic.

1.5.3 CONTROL ELEMENT ASSEMBLY DROP TESTS

A series of tests was completed on both single and dual CEAs in a cold water, low pressure facility to satisfy the following objectives:

a. Determine the mechanical and functional feasibility of the CEA type control rod concept.

b. Experimentally determine the relationship between CEA drop time and CEA drop weight, annular clearance between CEA fingers and guide tubes, and coolant flow rate within the guide tube.

c. Experimentally determine the relationship between flow rate and pressure drop within the guide tube as a function of CEA axial position and of finger-to-guide-tube clearance.

d. Determine the effects on drop time of adding a flow restriction or of plugging the flow holes in the lower portion of a guide tube (as might occur under accident conditions).

e. Determine the effects of misalignment within the CEA guide tube system on drop time.

The results of these tests were used as the basis for selecting the final CEA and guide tube geometrics. The tests have demonstrated that the five-finger CEA concept is mechanically and functionally feasible and that the CEA design has met the criteria established for drop time under the most adverse conditions. The testing has also verified that the analytical model used for predicting the drop times gives uniformly conservative results.

The effects on drop time of all possible combinations of frictional restraining forces in the control element drive mechanism (CEDM), angular and radial misalignment of the CEDM, bowing of the guide tubes, and misalignments of the CEA should have been experimentally investigated and defined. The conditions tested simulated all the effects of tolerance buildup, dynamic loadings, and thermal effects. The tests demonstrated that misalignments and distortions in excess of those expected from tolerance buildup or any other anticipated cause would still result in acceptable drop times.

1.5.4 CONTROL ELEMENT DRIVE ASSEMBLY PERFORMANCE TESTS

An accelerated life test of a magnetic jack coupled to a CEA was completed. This test consisted of continuous operation of the mechanism for a total accumulated travel of 32,500 feet at conditions similar to those it will encounter when installed on the operating reactor. The mechanism was operated at a speed of 40 inches per minute. without malfunction or adjustments. In addition, 200 full height drops were completed with all drop times less than 2.5 seconds for 90 percent insertion. Subsequent testing at various conditions was conducted to determine maintenance cycles.

Tests have shown that the magnetic jack type mechanism will operate in the anticipated containment environment after a Design Basis Accident. Among various other tests documented in Reference 1.5-2, a magnetic jack type CEDM, similar to that installed at Unit 2 was verified to be capable of withstanding a complete loss of air cooling for a 4 hour period with the plant at normal operating temperature and pressure (600°F and 2250 psi) without damage to the CEDM and holding the CEA. In addition, the coils stacks were later subjected to a steam environment for 15 minutes without affecting their electrical capabilities.

The design of the CEDM is such that loss of CEDM cooling will not prevent the CEDM from releasing the CEA. The ability of the CEDM to release the rods is not dependent on the cooling flow provided by the CEDM cooling system. Cooling function is only to ensure reliability of the CEDM coil stack.

1.5.5 FUEL ASSEMBLY FLOW TESTS

Velocity and static pressure measurements were made in an oversized model of a fuel assembly to determine the flow distributions present. Effects of the distributions on thermal behavior and margin are to be evaluated, where necessary, with the use of a CE version of the COBRA thermal and hydraulic code (Reference 1.5-1). Subjects investigated include the following:

a. Assembly inlet flow distribution as affected by the core support plate and bottom header plate flow hole geometry: Flow distribution was measured and results indicate that uniform nominal value is achieved within 10 percent of core height. The normal inlet flow distribution arising from the geometric configuration of the core support plate and lower end fitting of the fuel assembly was shown to have an effect on thermal margin which was small enough so that no allowance had to be made in the context of CE current conservative thermal-hydraulic calculational techniques.

b. Assembly inlet flow distribution as affected by a blocked core support plate flow hole: Flow distribution was measured and indicated that flow was recovered to at least 50 percent of the uniform nominal value at an elevation corresponding to 10 percent of core height. Analysis of several of the flow maldistributions arising from the unlikely blockage of a flow hole in the core support plate or from the blockage of one to nine subchannels indicated that flow recovery is rapid enough downstream of the obstruction so that the complete blockage of a core support-plate flow hole or of a single subchannel during 120 percent of full power operation would not result in a W-3 departure from boiling ratio (DNBR) of less than 1.0. The experimental data also indicated that the upstream influence of a subchannel blockage diminished very rapidly in that direction.

c. Flow distribution within the assembly as affected by complete blockage of one to nine subchannels: The flow distributions were measured and indicated very little upstream effect on such blockage, followed by recovery to normal subchannel flow conditions within 10 to 15 percent of core height, depending upon the number of subchannels blocked.

d. Flow distribution below the top header plate, as affected by the header plate and alignment plate flow hole geometry and by the presence of the CEA shroud: Measurements of the flow distribution near the top of the active core demonstrated that there was a negligible effect of the fuel assembly end fitting, alignment plate, and CEA shroud on that distribution.

1.5.6 REACTOR VESSEL FLOW TESTS

Tests were conducted with one-fifth scale models of CE reactors to determine hydraulic performance. The first tests were performed for the Palisades plant which has a reactor coolant system (RCS) similar to that of Millstone Unit 2. The tests investigated flow distribution, pressure drop and the tracing of flow paths within the vessel for all four pumps operating and various part-loop configurations. Air was used as the test medium. CE has also conducted tests on a one-fourth scale model of the Fort Calhoun reactor using air as the test medium.

Similar one-fifth scale model tests have been performed for Maine Yankee, which has a core similar to that of Millstone Unit 2. These tests were conducted in a cold water loop. All components for the model were geometrically similar to those in the reactor except for the core where 217 cylindrical core tubes were substituted for the fuel bundles. The core tubes contained orifices to provide the proper axial flow resistance.

Flow characteristics for Millstone Unit 2 were determined by taking into consideration similarities between Millstone Unit 2 and other CE reactors in conjunction with the experimental data from the flow model programs.

1.5.7 IN-CORE INSTRUMENTATION TESTS

Tests on in-core thermocouples and flux detectors were performed to ensure that the instrumentation will perform as expected at the temperatures to be encountered and that it does not vibrate excessively and cause excessive wear or fretting. Cold flow testing has been completed on a similar detector cable; no adverse vibrations or wear effects were encountered. Hot flow testing is also complete. After 2,000 hours at 590°F and 2,100 psig in a test loop, no breach of mechanical integrity was observed.

Mechanical tests of the insertion and removal equipment and instrumentation were performed on thimbles of the same approximate configuration as those used on Millstone Unit 2. The top entry in-core instrumentation design provides a means of eliminating the need of handling instrument assemblies separately, thus, minimizing downtime and personnel exposure. A full-scale mockup was built to accommodate three in-core instrumentation thimble assemblies. Major components and subassemblies of the mockup included:

a. An in-core instrumentation test assembly, including the upper guide structure support plate, three thimble guide sleeves, fuel alignment plate, three fuel bundle guide tubes, and the core support plate.

b. A thimble assembly consisting of the instrument plate, three in-core instrumentation thimbles and the lifting sling.

c. An upper guide tube with the guide tube attached to the thimble extension in and the detector cable partially inserted in the guide tube.

Insertion and withdrawal tests were performed to determine the frictional forces of a multi-tube instrument thimble assembly during insertion and withdrawal from a set of fuel bundles. This test simulated the operation that will be performed during the refueling of the reactor. To determine whether jamming of the thimbles would occur during this operation, bending loads were applied to the thimble assembly by tilting the instrument plate in 0.5 degree increments up to a total of five degrees from horizontal. Guide tubes were filled with water. The assembly was raised and lowered approximately five times for each tilt setting. Results showed no discernible difference in the friction forces for the various tilt settings. The tests demonstrated that the repeated insertion and withdrawal of in-core instrumentation thimble assemblies into the fuel bundle guides can be accomplished with reasonable insertion forces.

Life cycle tests were performed to determine if the frictional forces increase as a result of 40 insertions and withdrawals. An automatic timer was installed in the crane electrical circuitry to automatically cycle the thimble assembly between the fully inserted and withdrawn position. The instrument plate was set for five degrees tilt and the assembly was cycled 60 times. The insertion and withdrawal forces were measured during the first and last five cycles. No discernible difference was noticed.

An off-center lift test was performed to determine if the thimble assembly could be withdrawn from the core region while lifting the assembly from an extreme off center position. For a lifting point 11 inches off center, insertion was accomplished without incident. The flexibility of the thimble is such that jamming of the assembly due to off-center lifting does not occur.

Cable insertion tests were performed to determine the forces required to completely insert and withdraw a detector cable from the in-core instrumentation thimble assembly. The guide tube routing included typical bends equal to, or worse than, those found in the reactor. The detector cable was passed through the guide tubing and into a thimble. In all cases, the insertion and withdrawal forces were reasonable for hand insertion.

1.5.8 MATERIALS IRRADIATION SURVEILLANCE

Surveillance specimens of the reactor vessel shell section material are installed on the inside wall of the vessel to monitor the change in fracture toughness properties of the material during the reactor operating lifetime. Details of the program are given in Section 4.6.

1.5.9 REFERENCES

1.5-1 Rowe, D. S., “Cross-Flow Mixing Between Parallel Flow Channels During Boiling.” COBRA Computer Program for Coolant Boiling in Rod Arrays, Part 1, BNWL-371, March 1967.

1.5-2 Combustion Engineering Inc., Test Report Number TR-DT-78, dated 8/21/72, “Magnetic Jack Type Control Element Drive Mechanism Design and Test Report.”

1.6 IDENTIFICATION OF CONTRACTORS

Originally, The Connecticut Light and Power Company (CL&P), the Hartford Electric Light Company (HELCO), and Western Massachusetts Electric Company (WMECO) (the Owners), and Northeast Nuclear Energy Company (NNECO) were the applicants for the operating license for Millstone Unit 2. At that time NNECO acted as the agent for the owners and was responsible for the design, construction and operation of the plant. However, in 2001, the operating license was transferred to Dominion Nuclear Connecticut, Inc., at which time they became the sole owner and operator of Millstone Unit Number 2.

Combustion Engineering (CE), Inc. was engaged to design, manufacture and deliver the Nuclear Steam Supply System (NSSS) and nuclear fuel for the first core and the first two core reload batches to the site. The NSSS includes the reactor coolant system, reactor auxiliary system components, nuclear and certain process instrumentation, and the reactor control and protective system. In addition, CE furnished technical assistance for erection, initial fuel loading, testing and initial startup of the NSSS.

Bechtel Corporation was engaged as the Engineer-Constructor for this project and as such performed engineering and design work for the balance-of-plant equipment, systems and structures not included under the CE scope of supply. Bechtel was engaged to perform onsite construction of the entire plant with technical advice for installation of the reactor components provided by CE.

The reactor vessel closure head was replaced during refueling outage 16 with a new head assembly fabricated from materials that are less susceptible to Primary Water Stress Corrosion Cracking (PWSCC). The new head assembly was manufactured by Mitsubishi Heavy Industries. Westinghouse/CE was engaged in the design, installation and testing of the head.

The pressurizer assembly was replaced in 2006 with a new assembly fabricated from materials that are less susceptible to PWSCC. AREVA was engaged in the design, fabrication, installation and testing of the replacement pressurizer.

1.6.1 REFERENCES

1.6-1 Millstone Unit 3, Final Safety Analysis Report, Section 13.1 - Organizational Structure.

1.7 GENERAL DESIGN CHANGES SINCE ISSUANCE OF PRELIMINARY SAFETY ANALYSIS REPORT

1.7.1 GENERAL

Since the issuing of the Preliminary Safety Analysis Report (PSAR), a number of changes were made in the design of Millstone Unit 2. These changes improved the operating characteristics and enhance plant safety and reliability. The following reflects changes made up to the time of operating license application.

1.7.2 CONTROL ELEMENT DRIVE MECHANISMS

Magnetic jack drive mechanisms are provided for positioning the control element assemblies (CEA) instead of rack and pinion drive mechanisms. The magnetic jack control element drive mechanism (CEDM) is completely sealed by a pressure boundary, eliminating the need for seals. Motion of the control element drive shaft is accomplished by sequencing five solenoid coils located around the pressure boundary.

Combustion Engineering (CE), Inc., supplied identical CEDMs on previous plants, including Maine Yankee (Atomic Energy Commission (AEC) Docket Number 50-309) and Calvert Cliffs Units 1 and 2 (AEC Docket Number 50-317 and 50-318).

1.7.3 RADIOACTIVE WASTE PROCESSING SYSTEM

1.7.3.1 Clean Liquid Waste Processing System

A closed drains system and a 700 gallon equipment drain sump tank were included in the system to collect liquids containing dissolved hydrogen and fission gases from equipment drains, valve stem leakoffs, and relief valve discharges. The liquid wastes are collected in this tank via the closed drains system. This tank was provided to minimize the release of radioactive gases to the atmosphere without prior processing by the gaseous waste system.

The flash tank was replaced by a packed column-type degasifier utilizing internally generated stripping steam. The degasifier has a better decontamination factor for xenon and krypton than would have been possible with the proposed flash tank.

Plant space and the necessary piping and valves were provided for incorporating two additional demineralizers into the system, if required, based on operating experience.

1.7.3.2 Gaseous Waste Processing System

Four additional waste gas decay tanks were added to the system to allow for a minimum of 60 day decay of all hydrogenated waste gases, including cover gases, collected by the system prior to release to the atmosphere through the Millstone stack.

1.7.4 VITAL COMPONENT CLOSED COOLING WATER SYSTEM

The vital components closed cooling water system was deleted and the components cooled as follows:

Component	Cooling System
Service air compressors and instrument air compressors	Turbine building closed cooling water (interconnecting piping provided to reactor building closed cooling water)
Auxiliary feedwater pump turbine oil cooler	Water being pumped
Diesel generator	Service water
Control room air conditioners	Air

1.7.5 ELECTRICAL

1.7.5.1 AC Power

The station service transformers supply power at 6900V and 4160V via their respective station service busses for large motor loads. Further, the 4160V supplies power to the 480V unit substation transformers for smaller loads.

To preserve redundancy and separation, each motor control center is fed from only one 480 volt load center rather than from two.

1.7.5.2 Diesel Generators

For the change in the diesel engine cooling water supply, see Section 1.7.4.

Additional conditions under which the diesel generators will start automatically are noted in Section 8.3.3.1.

1.7.5.3 DC Supply

A third station battery was added to care for the non safety-related 125 volt DC loads associated with the turbine generator.

Each 125 volt DC distribution panel formerly had a feeder from each of the two station batteries, with diodes to prevent tying the battery buses together. To maintain the independence of redundant sources, the diodes were removed and the DC distribution panels fed from redundant battery buses.

1.7.5.4 Instrument Power

Two 120 volt regulated AC instrument buses were provided (instead of one) to assure redundant power sources for vital instrumentation.

1.7.6 AXIAL XENON OSCILLATION PROTECTION

Automatic initiation of an appropriate protection system for axial xenon oscillation was incorporated into the reactor protective system. This addition provided compliance with the AEC's General Design Criterion 20 as published February 20, 1971, in the Federal Register and as interpreted for preceeding reactors of similar design (see Calvert Cliffs Units 1 & 2 Amendment 15, Question 3.14). The basis for this addition was to provide an automatic protective backup to the operator in the unlikely event he should fail to adjust the full length CEA as required late in core life when axial xenon oscillations may become divergent.

1.7.7 NUMBER OF CONTROL ELEMENT ASSEMBLIES AND DRIVE MECHANISMS

The number of CEAs in the Millstone Unit 2 reactor is 73, compared to 85 CEAs shown in the PSAR design. The number of drive mechanisms was changed from 65 in the PSAR to 69 for Cycle 1. Then, removal of 8 part-length CEAs in 1978 reduced the number of drive mechanisms to 61. This resulted in a net increase in the number of single CEAs (37 to 49) and a net reduction in the number of dual CEAs (40 to 24), thereby providing greater flexibility for optimization of CEA programming and fuel management.

1.7.8 BURNABLE POISON SHIMS

Burnable poison shims were added to the fuel assemblies in Cycle 1, replacing some fuel. These shims permitted lowering of the initial boric acid concentration in the coolant. This provided additional assurance that the moderator temperature coefficient, at power at beginning of life, would not be positive.

1.7.9 STRUCTURES

The following changes have been made:

a. The post-tensioning tendons were encased in galvanized rather than ungalvanized semi-rigid sheaths.

b. The bearing plate material was changed from A-36 to VNT steel.

c. The warehouse area and turbine building were designated Class I structures.

d. All concrete reinforcing steel larger than number 11 was mechanically spliced.

e. Dye penetrant and magnetic particle inspection were not used for liner plate weld quality control.

1.7.10 HIGH PRESSURE SAFETY INJECTION PUMPS

High Pressure Safety Injection (HPSI) pump P-41B (Figure 6.1–1) (Sheet 2) was connected to each of the two suction headers but is normally isolated by valving. This HPSI pump served as a spare and was aligned, process wise and electrically, for operation only when either of the other two HPSI pumps is taken out of service. Two operable HPSI pumps satisfy redundancy requirements for core cooling.

1.7.11 CONTAINMENT PURGE VALVE ISOLATION ACTUATION SYSTEM

Containment Purge Valve Actuation System was changed from two-out-of-four to one-out-of-four logic. See Sections 7.3.2.3 and 7.5.6.3 for details.

1.7.12 CONTROL ELEMENT DRIVE SYSTEM

The Control Element Drive System (CEDS) was modified to include a CEA Motion Inhibit feature which acts to help the operator assure that limits on CEA position are not exceeded. The CEDS is described in Section 7.4.2.

1.8 ADVISORY COMMITTEE ON REACTOR SAFEGUARDS SPECIAL INTEREST ITEMS [THIS SECTION PROVIDES HISTORICAL INFORMATION PROVIDED TO THE ACRS AT THE TIME OF INITIAL LICENSING AND WAS NOT INTENDED TO BE UPDATED.]

1.8.1 GENERAL

This section describes the status of programs initiated to investigate the items which were identified by the Advisory Committee on Reactor Safeguards (ACRS) as being of special interest and pertaining to all large water-cooled power reactors up to the time of application for an operating license.

In carrying out these programs, information derived from research and development activities of the Atomic Energy Commission (AEC) and other organizations in the nuclear power industry were considered.

1.8.1.1 Ability of Fuel to Withstand Transients at End of Life and Experimental Verification of Maximum Linear Heat Generation Rate

The fuel cladding was designed to limit the transient stresses to two-thirds of the unirradiated value of the yield stress even during a depressurization transient near the end of life, when the internal gas pressure is highest.

Experimental verification of the maximum linear heat generation rate employed in the Millstone Unit 2 design was discussed in the original FSAR submitted at the time of application for an Operating License. Numerous irradiation tests, which bracket the design of these units, were performed, including those in the Westinghouse test reactor, the Shippingport blanket irradiations, the mixed oxide irradiations in the Saxton reactor, the zirconium clad UO_2 fuel rod evaluations in the Vallecitos boiling water reactor, the large speed blanket reactor rod irradiations, the center melting irradiations in Big Rock, Peach Bottom 2 irradiations, and NRX irradiations (AECL-Canada). In these tests, fuel rods similar to those employed in the design of the Millstone Unit 2 core were successfully irradiated to fuel burnups varying from very short term tests up to 60,000 MWD/MTU and at linear heat rates ranging from 5.6 up to 27.0 kW/ft.

1.8.1.2 Fuel Integrity Following a Loss-of-Coolant Accident

The ACRS had asked that information be developed to show that the “...melting and subsequent disintegration of a portion of fuel assembly...will not lead to unacceptable conditions.” They referred specifically to the “...effects in terms of fission product release, local high pressure production, and the possible initiation of failure in adjacent fuel elements...”.

Inquiry was made as to whether accident conditions that might occur which cause clad temperatures to reach such high temperatures that embrittlement occurs, and whether subsequent quenching operations will cause the embrittled portions to disintegrate and thereby prevent a sufficient flow of emergency core coolant to the remainder of the core.

Fuel damage of the magnitude indicated is prevented by the inherent nuclear and thermal characteristics of the UO_2 core and by the provision of engineered safety features (ESF).

With regard to the nonexcursion mechanisms leading to the conditions described by ACRS, the following two conditions might be conjectured:

A. Fuel bundle inlet flow blockage during full power operation and subsequent overheating of the coolant-starved fuel, or

B. loss of reactor coolant.

Condition A, inlet flow blockage during full-power operation and subsequent overheating and melting of the fuel, is not considered possible because open (nonshrouded) fuel bundles are used, thereby providing cross-flow to the flow-starved channel even if some of the inlet holes were blocked. Details and conclusions of the tests performed at Combustion Engineering (CE), Inc. on the influence of inlet geometry on flow in the entrance region are presented in ASME paper 68-WA/HT-34 delivered at the December 1968 Winter Annual Meeting. Further analysis of these tests showed that if a group of four flow holes in the core support plate at the base of the fuel bundle were blocked, the subchannels above the blocked region would have an inlet velocity about 21 percent of the core average bulk inlet velocity. Because of crossflow from the surrounding nonblocked regions, the net effect of this flow shortage, using conservative calculations, is to increase the enthalpy rise of the blocked region by a maximum of 35 percent. At nominal conditions, the hot channel departure from nucleate boiling ratio (DNBR) would drop from 2.0 to 1.4, assuming that the blockage occurred directly below the design hot channel.

Condition B was covered comprehensively in the Statement of Affirmative Testimony and Evidence of Combustion Engineering in the Matter of Rulemaking Hearing for the Acceptance Criteria for Emergency Core Cooling System for Light-Water-Cooled Nuclear Power Reactors, Docket Number RM-50-1. The emergency core cooling system (ECCS) is designed to remove the decay heat from the core for the necessary period of time following a loss-of-coolant accident (LOCA). Core power distributions and LOCA temperature-time histories indicate that for peak clad temperatures below 2300°F, the total clad oxidation will be significantly less than 1 percent.

1.8.1.3 Primary System Quality Assurance and In-Service Inspectability

A comprehensive quality assurance program has been established to assure that Millstone Unit 2 is designed, fabricated, and constructed in accordance with the requirements of applicable specifications and codes. The program started with the initial plant design and has continued through all phases of equipment procurement, fabrication, erection, construction, and plant operation. The program provides for review of specifications to assure that quality control requirements are included and for surveillance and audits of the manufacturing and construction efforts to assure that the specified requirements are met.

A summary description of the Quality Assurance Program (QAP) is included as Section 12.8. This program fully meets the guidelines established in the former AEC Regulation 10 CFR Part 50, Appendix B entitled “Quality Assurance Criteria for Nuclear Power Plants.” The quality assurance organization is described in the Quality Assurance Program Description Topical Report. That information is incorporated herein by reference.

Baseline inspection and subsequently in-service inspections are performed and are further discussed in Section 4.6.6.

1.8.1.4 Separation of Control and Protective Instrumentation

In addition to any redundancy and separation provided for control or for protective instrumentation, the control and protective instrumentation are independent of each other. Control action and protective action derived from the same process variable are generated by separate instrumentation loops. Malfunction of a single control instrumentation loop cannot impair the operation of the protective instrumentation loop and conversely malfunction of the protective instrumentation loop does not affect operation of the control loop. The instrumentation for a single protective and a single control channel may be located adjacent to one another, and their circuits may be routed in the same cable tray, but each is capable of performing its function independently of the other. Further discussion is provided in Chapters 7 and 8.

1.8.1.5 Instrumentation for Detection of Failed Fuel

Early detection of the gross failure of fuel elements permits early applications of action necessary to limit the consequences.

Based on a study of the expected fission and corrosion product activities in the reactor coolant, it was concluded that the gross gamma plus specific isotope monitor provides a simple and reliable means for early detection fuel failures.

The design bases of the detection system include the following:

a. Trends in fission product activity in the reactor coolant system (RCS) (specifically Rb-88) are used as an indication of fuel element cladding failures.

b. There is a time delay of less than five minutes before the activity, emitted from a fuel element cladding failure, is indicated by the instrumentation. This time delay is a function of the location of the monitor.

c. The information obtained from this system will not be used for automatic protective or control functions or detection of the specific fuel assembly (or assemblies) which has failed.

d. The high activity alarm will be supplemented with radiochemical analysis of the reactor coolant for fission products to provide positive identification of a fuel element failure.

The location and operation of the detector, designated as a process radiation monitor, is described in Sections 7.5.6.3 and 9.2.2.

Note: This section provides historical information provided to ACRs at the time of initial Licensing and was not intended to be updated.

1.8.1.6 Effects of Blowdown Forces on Core and Primary System Components

The dynamic response of reactor internals resulting from hydrodynamic blowdown forces under a postulated LOCA condition was the subject of a CE topical report which contained a complete description of the theoretical basis for methods of analysis for the various reactor components, as well as documentation of computer programs and the respective analytical structural models.

Reactor vessel internal structures were analyzed to ensure the required structural integrity during abnormal operating conditions, including the effects of blowdown, pressure drop and buckling forces. For the LOCA, the CEFLASH-4 computer program was used to define the flow transient and the WATERHAMMER program determines the corresponding dynamic pressure load distribution. The dynamic response of the reactor vessel internals to the space and time-dependent pressure loads were obtained through the use of a number of structural dynamic analysis codes. Lateral and vertical dynamic response of the internals were considered, as well as the transient response and dynamic buckling of a core support barrel in shell modes. Both the CEFLASH-4 and WATERHAMMER models were evaluated against the LOFT program results.

The loads resulting from the LOCA condition were added to the loads resulting from normal operation and the design basis earthquake (DBE) for each critical component and the component deflections and stresses analyzed to ensure compliance with the criteria specified in Section 4.2.

1.8.1.7 Reactor Vessel Thermal Shock

Sufficient emergency core cooling water is available to flood the core region in the event of a major LOCA. The Millstone Unit 2 design uses a section of each of the RCS cold legs to conduct the water from the safety injection nozzles to the reactor vessel. This water then flows into the downcomer annulus and into the lower plenum of the reactor vessel before flooding the core itself. Analytical investigations were performed to provide assurance that the resultant cooling of the irradiated inner surface of the thick-walled reactor vessel will not induce or propagate cracks sufficient to cause the reactor vessel to fail.

An analytical evaluation of pressurized thermal shock effects in CE’s NSSS was issued by CE in December 1981 (CEN-189). The limiting case is a small break LOCA with the assumption of concurrent loss of all feedwater. For Millstone Unit 2, it was found that crack initiation would not occur during this limiting transient throughout the unit's design life (32 EFPY).

Subsequently, the Pressurized Thermal Shock Rule (10 CFR 50.61, 1986) was used for embrittlement shift prediction. The results confirmed that the reactor vessel was fully able to withstand a postulated pressurized thermal shock imposed by the ECCS through the unit's design life.

1.8.1.8 Effect of Fuel Rod Failure on the Capability of the Safety Injection System

CE conducted experimental and analytical investigations of fuel-rod failures under simulated LOCA conditions. The analytical work provided indications of the actual conditions to be expected in the core during a transient, in terms of potential clad heating rates, internal pressures and transient duration. The experimental work applied these parameters in various combinations to establish the nature of fuel-rod deformation which might occur under accident conditions. This subject was covered comprehensively in the Statement of Affirmative Testimony and Evidence of Combustion Engineering in the Matter of Rulemaking Hearing for the Acceptance Criteria for Emergency Core Cooling Systems for Light-Water-Cooled Nuclear Power Reactors, Docket Number RM-50-1.

1.8.1.9 Preoperational Vibration Monitoring Program

A preoperational vibration monitoring program (PVMP) was completed for the Palisades reactor internals. Results of this program were submitted to the AEC by CE Report CENPD-36. Additional PVMPs were developed for both the Maine Yankee and Fort Calhoun reactor internals. In keeping with the requirements for prototype vibration test programs, predictions of hydraulic forcing functions and structural response were made for the Maine Yankee and Fort Calhoun reactor internals and correlated to test program measurements. Vibration test data from all three reactors was used in demonstrating the adequacy of the Millstone Unit 2 reactor vessel internals to sustain flow-induced vibration effects. The vibration test data available, together with appropriate analyses, permitted the assessment of design or fabrication differences existing among the subject reactors as they related to the vibrational response characteristics of the Millstone Unit 2 reactor internals. A comparison of applicable design parameters for the Palisades, Fort Calhoun, Maine Yankee and Millstone Unit 2 reactors as of the time of application for operating license is presented in Table 1.8-1.

The analytical methods which formed the basis for the CE vibration response predictions were provided in the Maine Yankee and Fort Calhoun vibration monitoring programs submittals. Palisades, Maine Yankee and Fort Calhoun Flow Model Test reports and a description of the methodology utilized to relate these data to in-reactor forcing functions were provided, as well as a description of the structural response computer code.

1.8.1.9.1 Basis of Program

The suitability of using PVMP data from Palisades, Omaha and Maine Yankee as a composite prototype was based on the following:

a. Reactor internals structural response and LOCA hydraulic loadings could be adequately predicted with computer programs available, and the methods and procedures will be provided and justified.

b. The hydraulic forcing function predicting method was provided and justified. The forcing function method was verified by measurements in the prototype(s).

c. Additional instrumentation to measure or derive forcing functions was added to the Fort Calhoun reactor in accordance with Regulatory Guide 1.20 (formerly Safety Guide 20).

The prediction methods and procedures were used to predict the responses (amplitude and frequency) for the Fort Calhoun PVMP.

d. The Maine Yankee and Fort Calhoun PVMP results were satisfactory, satisfying AEC licensing requirements for all CE reactor plants which had either construction or operating permits, providing the configuration and flow modes were similar as specified in Regulatory Guide 1.20 (formerly Safety Guide 20).

e. CE provided predictive methodology and predicted and limiting values of response (acceptance criteria) on the Maine Yankee program. The program results were provided on a timely basis in accordance with the Regulatory Guide 1.20 (formerly Safety Guide 20).

f. CE submitted a report on the LOCA dynamic analysis methods and procedures.

1.8.1.9.2 Millstone Unit 2 Program

The PVMP to be conducted for Millstone Unit 2 reactor internals was consistent with those portions of the former Safety Guide 20 (after replaced by Regulatory Guide 1.20), which addressed nonprototype reactors.

The following was the PVMP plan for Millstone Unit 2. As noted above, this program was contingent upon the results to be obtained from Maine Yankee and Fort Calhoun PVMP.

1. The reactor internals important to safety were be subjected during the preoperation functional testing program to all significant flow modes of normal reactor operation and under the same test conditions conducted on the Palisades, Fort Calhoun, and Maine Yankee designs.

   The test duration was at least as long as that conducted on the Palisades, Fort Calhoun and Maine Yankee designs.

2. Following completion of the preoperational functional tests, the reactor internals were removed from the reactor vessel and visual and nondestructive examination of the reactor internals was conducted. The areas examined included:

   a. All major load bearing elements of the reactor internals relied upon to retain the core structure in place;

   b. The lateral, vertical, and torsional restraints provided within the vessel;

c. Those locking and bolting devices whose failure could adversely affect the structural integrity of the internals;

d. Those other locations on the reactor internal components which were examined on the Palisades, Fort Calhoun, and Maine Yankee designs;

e. The interior of the reactor vessel for evidence of loose parts or foreign material.

A summary of the PVMP inspections described above was submitted after the completion of the inspection and tests in a report.

It should be pointed out that the reactor thermal shield was removed from the lower internals assembly because of the damage suffered due to excessive vibratory movement. An evaluation was performed to assess the effects of thermal shield removal on the vibratory response of the rest of reactor internals. It was concluded that the effect would be minimal and that the conclusions of the PVMP were still valid.

1.8.2 SPECIAL FOR MILLSTONE UNIT 2

1.8.2.1 Release of Radioactivity in Case of Damaged Fuel Assemblies in Spent Fuel Pool

In the event of release or radioactivity resulting from damaged fuel in the spent fuel pool, the auxiliary exhaust system (AES) which is described in Section 9.9.8, diverts the effluent through the enclosure building filtration system (EBFS) charcoal filters prior to release through the Millstone stack. The AES maintains the fuel handling area under a negative pressure to limit uncontrolled release of radioactivity.

1.8.2.2 Hydrogen Control

The independent systems in the hydrogen control systems monitor and mix hydrogen in the containment following a LOCA (see Section 6.6). Each is a full capacity, completely redundant, independent system. Air to operate the hydrogen monitoring system CIV’s is provided by the instrument air system with a backup air bottle system that is designed to meet single failure criteria. Two, full capacity hydrogen purge systems not credited in accident analyses are provided. The hydrogen recombiner system has no mitigating function.

1.8.2.3 Common Mode Failures and Anticipated Transients Without Scram

CE analyzed the response of pressurized water reactors which are typical of Millstone Unit 2 to demonstrate the diversity of the reactor protective system in mitigating common mode failures and the response of the plant to anticipated transients without scram (ATWS). Results of these studies were submitted to the AEC as topical reports.

CE Report CENPD-11, entitled “Reactor Protection System Diversity” was submitted on March 4, 1971. This report evaluated systematic, nonrandom, concurrent failures, (i.e., common mode failures) of redundant devices not considered credible based on quality assurance in design, qualification testing, and periodic testing that common mode failure could disable all instrument channels which measure a given process parameter, the report, nevertheless, addresses this type of failure. Monitoring of the condition by diverse means or principles enables a protection system to withstand common mode failures. The evaluations included the following accidents: control element assembly (CEA) withdrawal, CEA drop, loss of reactor coolant flow, excess load, loss of load and loss of feedwater. The results of the study demonstrated that the diversity of the reactor protective system is such that gross fuel damage or consequential failures in the RCS or in the main steam system will not occur for any of the accidents analyzed.

A draft of the CE report, entitled "Topical Report on Anticipated Transients Without Scram" (Proprietary) was submitted to the AEC on January 10, 1972. Evaluations were performed in this report based upon the assumption that no CEA are inserted into the core during the course of the following transients: CEA withdrawal, CEA drop, idle loop startup, loss of flow, boron dilution, excess load, loss of load, loss of feedwater, sample line break, and pressurizer safety valve failure. The transient resulting from loss of normal onsite and offsite power was also analyzed but with a conservative one percent negative reactivity insertion assumed following reactor trip signal generation, since for this case the failures which initiate the transient would also remove power from the control element drive mechanism (CEDM), allowing the CEAs to insert. The final report, with results and their applicability to Millstone Unit 2, was submitted to the AEC.

1.8.3 REFERENCES

1.8-1 Millstone Unit 3, Final Safety Analysis Report, Section 13.1 - Organizational Structure.

TABLE 1.8-1 COMPARISON OF PREOPERATIONAL VIBRATION MONITORING PROGRAM DESIGN PARAMETERS

<Parameter>	Palisades	Fort Calhoun	Maine Yankee	Millstone Unit 2
R_{mean}, inches	75-7/8	61-5/16	75.25	75.25
Upper CSB: t, inches	2	2	2.5	2.5
Upper CSB: L, inches	109.25	101-3/8	135-5/8	141.75
Upper CSB: $R_{mean,}$ inches	75-5/8	61-1/16	74-7/8	74-7/8
Middle CSB: t, inches	1.5	1.5	1.75	1.75
Middle CSB: L, inches	166.75	166-1/8	144.75	148.75
Middle CSB: $R_{mean,}$ inches	75-3/8	60-11/16	74-5/8	74-5/8
Lower CSB: t, inches	2	2.25	2.25	2.25
Lower CSB: L, inches	38.5	35-5/8	38	38
Lower Cylinder ID, inches	Integral	Integral	141	141
Core Cylinder OD, inches	Integral	Integral	145	145
Support Cylinder L, inches	Integral	Integral	42	42
Structure Supported	Integral	Integral	CSB Flange	CSB Flange
Core Shroud Support	Bolted to CBS	Bolted to CBS	Bolted to CBS	Bolted to CBS
Core Shroud: $R_{mean,}$ inches	73.5	59-1/16	72-5/8	72-5/8
Core Shroud: Cylinder t, inches	2	1.5	2	2
UGS: L, inches	15	24	24	24
UGS: Beams inches	18 by 1.5	24 by 1.5	24 by 1.5	24 by 1.5
UGS: Plate t, inches	3	3.25	4	4
Thermal Shield	No	Yes	Yes	Yes

TABLE 1.8-1 COMPARISON OF PREOPERATIONAL VIBRATION MONITORING PROGRAM DESIGN PARAMETERS

<Parameter>	Palisades	Fort Calhoun	Maine Yankee	Millstone Unit 2
Number of Loops	2	2	3	2
Design Minimum. Flow, 10^6 lbm/hr	125	71.7	122	139
Inlet Design Temperature, F	548	547	546	544
Inlet ID, inches [(a)]	35-1/8	28.75	39	35-3/16
Outlet ID, inches [(a)]	48-5/8	37	40	48-1/8
Inlet Pipe Velocity, ft/sec	37.7	33.7	39.2	41.6
Downcomer Velocity, ft/sec	19.6	25.2	24.9	26.7
Core Inlet Velocity, ft/sec	12.2	12.4	13.0	15.4
Outlet Pipe Velocity, ft/sec	41.4	41.3	42.6	46.5

(a)These IDs are measured at the inside wall of the reactor vessel as shown for the Millstone 2 reactor vessel in Figure 4.3–1.

CSB = Core Support Barrel

UGS = Upper Guide Structure

Velocity = Design Minimum Velocity

1.9 TOPICAL REPORTS

In support of the Final Safety Analysis Report, various "topical reports" prepared by Combustion Engineering, Inc., and Bechtel Corporation were referenced throughout this document. A list of "topical reports" as of the time of application for operating license is given in Table 1.9-1.

TABLE 1.9-1 TOPICAL REPORTS

Combustion Engineering, Inc.

Title	Millstone Unit 2 Original FSAR Section
ASME paper 68-WA/HT-34, December 1968 Winter Annual Meeting	1.8.1.2
Statement of Affirmative Testimony and Evidence of Combustion	1.8.1.2
Engineering in the matter of Rulemaking Hearing for the Acceptance Criteria for Emergency Core Cooling System for Light-Water-Cooled Nuclear Power Reactors, Docket Number RM-50-1	1.8.1.8
Dynamic Analysis of Reactor Vessel Internals Under Loss of Coolant Accident CENPD-42-3 (Submittal to AEC in July 1972)	1.8.1.6
Thermal Shock Analysis of Reactor Vessels Due to Emergency Core Cooling System Operation, A-68-9-1, March 15,1968, submitted as part of Amendment 9 to the Maine Yankee license application	1.8.1.7
Experimental Determination of Limiting Heat Transfer Coefficients During Quenching of Thick Steel Plates in Water, A-68-10-2, December 13, 1968	1.8.1.7
Finite Element Analysis of Structural Integrity of a Reactor Pressure Vessel During Emergency Core Cooling, A-70-19-2, January 1970	1.8.1.7
Palisades Precritical Vibration Monitoring Program, CENPD-36	1.8.1.9
Precritical Vibration Monitoring Program, CENPD-55	1.8.1.9
Reactor Protective System Diversity, CENPD-11, February 1971	1.8.2.3
Topical Report on Anticipated Transients Without Scram, CENPD-41	1.8.2.3
INTHERMIC, A Computer Code for Analysis of Thermal Mixing, CENPD-8	3.5.3
COSMO IV, A Thermal and Hydraulic Steady State Design Code for Water Cooled Reactors, CENPD-9	3.5.3
Seismic Qualification of Category I Electric Equipment for Nuclear Steam Supply Systems, CENPD-61	7.2.6.3

TABLE 1.9-1 TOPICAL REPORTS (CONTINUED)

Bechtel Corporation

Title	Millstone Unit 2 Original FSAR Section
Consumer Power Company Palisades Nuclear Power Plant Containment Building Liner Plate Design Report, B-TOP-1 (submitted to AEC in October, 1969)	5.2.4.5
Full-Scale Buttress Test for Prestressed Nuclear Containment Structures, BC-TOP-7	5.2.3.3.3
Testing Criteria for Integrated Leak Rate Testing of Primary Containment Structures for Nuclear Power Plants, BN-TOP-1	5.2.9.1
Design for Pipe Break Effects, BN-TOP-2 (REV. 1)	Question 4.16

1.10 MATERIAL INCORPORATED BY REFERENCE

The following is a list of material incorporated by reference in the Final Safety Analysis Report [(1)]:

1. Millstone Unit 2 Technical Requirements Manual (TRM).
2. As identified in the List of Figures, the engineering controlled plant drawings that are, coincidentally, MPS-2 FSAR Figures.
3. The Quality Assurance Program Description (QAPD) Topical Report.

(1) Information incorporated by reference into the Final Safety Analysis Report is subject to the update and reporting requirements of 10 CFR 50.71(e) and change controls of 10 CFR 50.59 unless separate NRC change control requirements apply (e.g., 10 CFR 50.54(a)).

1.A AEC GENERAL DESIGN CRITERIA FOR NUCLEAR POWER PLANTS

10 CFR PART 50 APPENDIX A

On February 20, 1971, the Atomic Energy Commission published in the Federal Register the General Design Criteria for Nuclear Power Plants. Prior to this date, proposed General Design Criteria for Nuclear Power Plants as issued on July 11, 1967, in the Federal Register were in effect. Before issuance of the construction permit for Millstone Unit 2, discussions reflecting the design intent in consideration of the 1967 proposed criteria were submitted in the PSAR. Design and construction was thus initiated and has been completed based upon the 1967 proposed criteria.

Since February 20, 1971, the applicants have attempted to comply with the intent of the newer General Design Criteria to the extent possible, recognizing previous design commitments. The extent to which this has been possible is reflected in the discussions of the 1971 General Design Criteria which follow.

CRITERION 1 - QUALITY STANDARDS AND RECORDS

> Structures, systems, and components important to safety are designed, fabricated, erected and tested to quality standards commensurate with the importance of the safety functions performed. Where generally recognized codes and standards are used, they are identified and evaluated to determine their applicability, adequacy, and sufficiency and are supplemented or modified as necessary to assure a quality product in keeping with the required safety function. A quality assurance program has been established and implemented in order to provide adequate assurance that these structures, systems, and components will satisfactorily perform their safety functions. Appropriate records of the design, fabrication, erection and testing of structures, systems, and components important to safety are maintained by or under the control of the nuclear power unit licensee throughout the life of the unit.

Discussion of the quality standards for those structures and components which are essential to the prevention of incidents which would affect the public health and safety or to mitigation of their consequences are presented in appropriate sections of the FSAR. The quality assurance program in effect to assure that these structures, systems, and components will satisfactorily perform their safety functions is discussed in Section 12.8.

For example, components of the safety injection and containment cooling systems are designed and fabricated in accordance with established codes and/or standards as required to assure that their quality is in keeping with the safety function of the component. It is not intended, however, to limit quality standards requirements to this list.

<u>High Pressure Injection, Low Pressure Injection, and Containment Spray Pumps</u>

a. Surfaces of pressure retaining materials for the high and low pressure safety injection pumps were examined by liquid penetrant techniques in accordance with the provisions of ANSI-B31.1, Paragraph 136.5.3(d). Surfaces of pressure retaining materials for the containment spray pumps were examined by dye penetrant techniques in accordance with the provisions of Draft ASME Code for Pumps and Valves for Nuclear Power, Class II, 1968. Casings for all three types of pumps have been hydrostatically tested to at least 1.5 times the design pressures.

b. Pressure containing butt welds for the safety injection pumps have been radiographed in accordance with Section VIII of the ASME Code, Paragraph UW-51.

c. The pump supplier submitted certified mill test reports of pressure containing materials.

d. At least one pump of each type has been hydraulic-performance tested for capacity and head, in accordance with the requirements of the Hydraulics Institute.

e. The pump seals have been designed to provide a high degree of assurance of their proper operation, including compatibility of seal materials with water chemistry conditions and minimum dependence on externally supplied cooling water.

f. Pump drive motors conform to NEMA Standards, MG-1.

Safety Injection Tanks

ASME Code, Section III, Class C.

Safety Injection and Containment Spray System Motor-Operated Valves and Control Valves

a. The design criteria for pressure containing parts is in accordance with ANSI B16.5.

b. Radiographic inspection of pressure containing butt welds has been performed in accordance with the requirements of ASME Code, Section VIII.

c. Certified mill test reports of pressure containing materials were provided by the supplier.

d. Pressure containing parts were hydrostatically tested in accordance with MSS-61.

e. Isolation valves are designed, fabricated, and tested in accordance with Draft ASME Code for Pumps and Valves for Nuclear Power, Class II, 1968. Control valves are designed, fabricated, and tested in accordance with ASME Code Section III, Nuclear Power Plant Components, Class II, 1971.

Containment Coolers

a. The cooling coils are similar to a representative section of a coil which was tested under the maximum environmental conditions which would exist following a loss-of-coolant accident (LOCA). The test results demonstrated that the full size coil assembly would be capable of removing the required heat load. These data are filed with the AEC in Topical Report W-CAP-7336-L.

b. The cooling coils are tested in accordance with ASME Code, Section VIII.

c. Air moving equipment, including fan motors, were designed to standards of the Air Moving and Conditioning Association, AMCA-211A.

d. A fan and motor combination were satisfactorily tested to prove their ability to operate under the conditions which would exist within the containment following a LOCA. These data will be presented to the AEC in Topical Report W-CAP-7829. The motor insulation and internal cable splice are filed in Topical Reports W-CAP-7343-L and W-CAP-9003, respectively.

e. Piping from the fan coolers to the containment penetrations was designed in accordance with the provisions of ANSI B31.1.0. The penetrations piping was designed to ANSI B31.7, Class II and the penetration isolation valves to the ASME Pump and Valve Code, Class II.

f. Valves, other than the penetration isolation valves, were designed in accordance with ANSI B31.1.0 and ANSI B16.5. Manually operated butterfly valves were in accordance with AWWA-C504.

Shutdown Heat Exchangers

a. Pressure containing materials were tested and examined per ASME Code, Section III, Class C.

b. Heat transfer design and physical design are in accordance with TEMA standards.

c. Certified mill test reports of pressure containing materials were provided by the supplier.

d. Radiographic inspection of pressure containing butt welds was performed in accordance with the requirements of ASME Code, Section III, Class C.

e. Pressure containing parts were hydrostatically tested in accordance with ASME Code, Section III, Class C.

All tests and inspections are reviewed during material procurement and fabrication of the components to assure conformance with the quality control techniques of the applicable codes and standards.

The appropriate sections in the FSAR discuss the specific codes and standards invoked in fabricating or erecting the structures, systems, and components important to safety.

Appropriate records of the design, fabrication, erection, and testing of structures, systems, and components important to safety shall be maintained for the life of the plant. (See Section 12.8).

CRITERION 2 - DESIGN BASES FOR PROTECTION AGAINST NATURAL PHENOMENA

> Structures, systems, and components important to safety are designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, flood, tsunami, and seiches without loss of capability to perform their safety functions. The design bases for these structures, systems, and components reflect:
>
> (1) Appropriate consideration of the most severe of the natural phenomena that have been historically reported for the site and surrounding area, with sufficient margin for the limited accuracy, quantity, and period of time in which the historical data have been accumulated, (2) appropriate combinations of the effects of the natural phenomena, and (3) the importance of the safety functions to be performed.

All structures, systems, and components important to safety have been designed to withstand, without loss of the capability to protect the public, the additional forces that might be imposed by natural phenomena. The most severe natural phenomena which are considered and discussed in other sections of this FSAR are as follows:

a. Earthquakes / SeismologySection 2.6

a. Wind and Tornadoes / MeteorologySection 2.3

a. Floods / HydrologySection 2.5.4

Appropriate natural phenomena are considered in the designs of structures, systems, and components. Accepted standards for the forces imposed by natural phenomena are used in the design.

A general description of the seismic analysis program is found in Section 5.8. Additional information on major structure design against the effects of natural phenomena is included in the following sections:

Containment Structure Section 5.2
Enclosure Building Section 5.3
Auxiliary Building Section 5.4

Turbine Building Section 5.5
Intake Structure Section 5.6
Reactor Vessel Internals Appendix 3.A
Reactor Coolant System Appendix 4.A

CRITERION 3 - FIRE PROTECTION

Structures, systems, and components important to safety are designed and located to minimize, consistent with other safety requirements, the probability and effect of fires and explosions. Noncombustible and heat resistant materials are used wherever practical throughout the unit, particularly in locations such as the containment and control room. Fire detection and fighting systems of appropriate capacity and capability are provided and designed to minimize the adverse effects of fires on structures, systems, and components important to safety. Fire fighting systems are designed to assure that their rupture or inadvertent operation does not significantly impair the safety capability of these structures, systems, and components.

Millstone Unit number 2 structures, systems, and components important to safety are designed and located to minimize the probability and effects of fires. Fire protection systems (active and passive) have been provided to assure that all possible fires are detected, controlled, and extinguished.

Fire protection and detection systems and components are designed and installed in accordance with applicable requirements of the National Fire Protection Association (NFPA). In areas where combustible material may exist, fixed fire detection and suppression are generally provided (Section 9.10).

Fire detection and fire suppression systems of appropriate types and capacities are designed to minimize the adverse effects of fires on structures, systems, and components important to safety. In some areas, portable extinguishers are used in lieu of water suppression systems. In areas such as the D.C. equipment rooms, a Halon suppression system is used in lieu of fixed water suppression to assure that sensitive electronics are not affected by water spray.

Fire fighting systems are designed to assure that their rupture or inadvertent operation does not significantly impair the capabilities of any structure, system, or component important to safety.

In areas where water may cause damage to safety equipment, such as vital electrical panels or the emergency diesel generators, either shielding is provided or the water suppression system is designed such that its actuation does not affect the safety systems it protects (pre-action sprinkler system, manual activation, shielding, etc.).

CRITERION 4 - ENVIRONMENTAL AND MISSILE DESIGN BASES

Structures, systems, and components important to safety are designed to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation, maintenance, testing, and postulated accidents, including loss-of-coolant accidents. These structures, systems, and components are appropriately 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 unit.

However, dynamic effects associated with postulated pipe ruptures in nuclear power units may be excluded from the design basis when analyses, reviewed and approved by the commission, demonstrate that the probability of fluid system piping rupture is extremely low under conditions consistent with the design basis for the piping.

All structures are designed in accordance with accepted and time proven building codes (as specified in Section 5.1.2) for the loading conditions stated in Sections 5.2.2, 5.3.3, 5.4.3, 5.5.3 and 5.6.3 which insures that they will operate under normal conditions in a safe manner. In addition, those structures and/or components which could affect public safety were designed to function safely during an earthquake as discussed in Section 5.8. Wind and tornado storm protection design criteria are discussed in Sections 5.2.2.1.6, 5.3.3.1.4, 5.4.3.1.6, 5.5.3.3.2, 5.6.3.1.5, and 5.7.3.1.4. Protection against postulated missiles is discussed in Section 5.2.5.1.

The design loads for the containment and major component supports to ensure a safe shutdown after a loss-of-coolant accident are described in Section 5.2.2.1.3.

Systems and components important to safety are designed to operate satisfactorily and to be compatible with environmental conditions associated with normal operation and postulated accident conditions. Those systems and components located in the containment are designed to operate in an environment of 289°F and 54 psig. Systems and components important to safety are designated Seismic Class I and designed in accordance with the criteria given in Section 5.2.4.3. Missile protection and pipe whipping protection criteria for these systems and components are given in Sections 5.2.5.1 and 5.4.3.1.

Leak-before-break (LBB) analyses for the reactor coolant system (RCS) main coolant loops, for the pressurizer surge line, and unisolable RCS portions of the safety injection and shutdown cooling piping, which demonstrated that the probability of fluid system piping rupture was extremely low, were reviewed and approved by the commission. Subsequent to the commission review and approval, weld overlays were applied to dissimilar metal welds (DMWs) at the shutdown cooling, the safety injection and the pressurizer surge nozzles. A revised LBB analysis was performed for these nozzles (see Reference A.30). Accordingly, pursuant to GDC 4, 1998 revision, the dynamic effects associated with pipe ruptures in the above piping segments, including the effects of pipe whipping and discharging fluids have been excluded from the design basis of the following components and systems:

Core barrel snubbers, core barrel stabilizer blocks
Reactor vessel core support ledge
Reactor Cavity Seal, Neutron Shielding
Pressurizer Blockhouse
Protection of Closed Systems
RBCCW piping

Steam Generator Blow Down piping
Steam Generator Blow Down sampling piping

CRITERION 5 - SHARING OF STRUCTURES, SYSTEMS, AND COMPONENTS

> Structures, systems, and components important to safety are not shared among nuclear power units unless it can be shown that such sharing will not significantly impair their ability to perform their safety function, including, in the event of an accident in one unit, an orderly shutdown and cooldown of the remaining units.

Both the auxiliary and the turbine buildings of Millstone Unit 2 are structurally connected to their respective Millstone Unit 1 buildings. The combined buildings are isolated in the lateral direction as discussed in Section 5.4.1 (auxiliary building) and Section 5.5.1 (turbine building). All vertical loads which may interact between Millstone Unit 1 and Millstone Unit 2 portions of the buildings were investigated to ensure that they will function safely under all design conditions.

The Millstone Unit 2 Condensate Polishing Facility is located in Warehouse Number 5, which is situated North of the Millstone Unit 2 Turbine Building and South of the Millstone Unit 3 Condensate Polishing Facility and Auxiliary Boiler Building.

A list of shared facilities appears in Section 1.2.13.

The safe shutdown of any unit will not be impaired by the failure of the facilities and systems which are shared.

CRITERION 10 - REACTOR DESIGN

> The reactor core and associated coolant, control and protection systems are designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during any condition of normal operation, including the effects of anticipated operational occurrences.

Plant conditions have been categorized in accordance with their anticipated frequency of occurrence and risk to the public, and design requirements are given for each of the four categories. These categories covered by this criterion are Condition I - Normal Operation and Condition II - Faults of Moderate Frequency.

The design requirement for Condition I is that margin shall be provided between any plant parameter and the value of that parameter which would require either automatic or manual protective action; it is met by providing an adequate control system. The design requirement for Condition II is that such faults shall be accommodated with, at most, a shutdown of the reactor, with the plant capable of returning to operation after corrective action; it is met by providing an adequate protective system. The following design limits apply:

a. The value of the departure from nucleate boiling ratio (DNBR) will not be less than its design limit to ensure that fuel failure does not occur.

b. The peak temperature in the fuel will be less than the melting point of irradiated UO_2 (considering effects of irradiation on melting point).

c. The maximum primary stresses in the zircaloy fuel clad shall not exceed two-thirds of the minimum yield strength of the material at the operating temperature.

d. Net unrecoverable circumferential strain shall not exceed 1 percent as predicted by computations considering clad creep and fuel-clad interaction effects.

e. Cumulative strain cycling usage, defined as the sum of the ratios of the number of cycles at a given effective strain range (E) to the permitted number (N) at that range shall not exceed 1.0.

f. The fuel rod will be designed to prevent gross clad deformation under the combined effects of external pressure and long term creep.

The thermal margins during normal operation ensure that the minimum thermal margins during anticipated operational occurrences do not exceed the design basis. The DNBR limit ensures a low probability of occurrence of DNB.

The occurrence of DNB does not necessarily signify cladding damage; it represents a local increase in temperature which may or may not cause thermal damage, depending upon severity and duration.

The design is adequate to satisfy the design bases in the event of a reactor coolant system depressurization transient at the end of a fuel cycle.

Limitation of fuel burnup will be determined by material rather than nuclear considerations. See references in Chapter 3. Sufficient margin is provided in this core design to allow for the ratio of peak-to-average burnup.

CRITERION 11 - REACTOR INHERENT PROTECTION

> The reactor core and associated coolant systems are designed so that in the power operating range the net effect of the prompt inherent nuclear feedback characteristics tends to compensate for a rapid increase in reactivity.

The combined response of the fuel temperature coefficient, the moderator temperature coefficient, the moderator void coefficient, and the moderator pressure coefficient to an increase in reactor power in the power operating range will be a decrease in reactivity; i.e., the inherent nuclear feedback characteristics will not be positive.

The reactivity coefficients for this reactor are listed in Table 3.4-2 and are discussed in detail in Section 3.4.3.

CRITERION 12 - SUPPRESSION OF REACTOR POWER OSCILLATIONS

> The reactor core and associated coolant, control, and protection systems are designed to assure that power oscillations which can result in conditions exceeding specified acceptable fuel design limits are not possible or can be reliably and readily detected and suppressed.

The reactor core is designed not to have sustained power oscillations. If any power oscillations occur, the control system is sufficient to suppress such oscillations.

The basic stability of a pressurized water reactor with UO_2 fuel is due to the fast acting negative contribution to the power coefficient provided by the Doppler effect.

Any trend toward xenon oscillations which may occur in the core are controlled and suppressed by movement of the control element assemblies (CEAs) so that the thermal design bases are not exceeded. Xenon oscillations are characterized by long periods and slow changes in power distribution. The nuclear instrumentation will provide the information necessary to detect these changes.

Xenon stability analysis for Millstone Unit 2 is discussed in Section 3.4.5. The reactor protective system is discussed in Section 7.2.

The reactor protective system automatically trips the reactor if axial xenon oscillations are permitted to approach unsafe limits (Sections 7.2.3.3.10 and 1.7.6).

CRITERION 13 - INSTRUMENTATION AND CONTROL

> Instrumentation are provided to monitor variables and systems over their anticipated ranges for normal operation, for anticipated operational occurrences, and for accident conditions as appropriate to assure adequate safety, including those 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. Appropriate controls are provided to maintain these variables and systems within prescribed operating ranges.

Instrumentation is provided, as required, to monitor and maintain significant process variables which can affect the fission process, the integrity of the reactor core, the reactor coolant pressure boundary, and the containment and its associated systems. Controls are provided for the purpose of maintaining these variables within the limits prescribed for safe operation.

The principal variables and systems to be monitored include neutron level (reactor power); reactor coolant temperature, flow, and pressure; pressurizer liquid level; steam generator level and pressure; and containment pressure and temperature. In addition, instrumentation is provided for continuous automatic monitoring of process radiation level and boron concentration in the reactor coolant system.

The following is provided to monitor and maintain control over the fission process during both transient and steady state periods over the lifetime of the core:

a. Ten independent channels of nuclear instrumentation, which constitute the primary monitor of the fission process. Of these channels, the four wide range channels are used to monitor the reactor from startup through full power; four will monitor the reactor in the power range and are used to initiate a reactor shutdown in the event of overpower; two Reactor Regulating channels will monitor the reactor in the power range.

b. Two independent CEA Position Indicating Systems.

c. Manual control of reactor power by means of CEA's.

d. Manual regulation of coolant boron concentrations.

In-core instrumentation is provided to supplement information on core power distribution and to provide for calibration of out-of-core flux detectors.

Instrumentation measures temperatures, pressures, flows, and levels in the main Steam System and Auxiliary Systems and is used to maintain these variables within prescribed limits.

The reactor protective system is designed to monitor the reactor operating conditions and to effect reliable and rapid reactor trip if any one or a combination of conditions deviate from a preselected operating range.

The containment pressure and temperature instrumentation is designed to monitor these parameters during normal operation and the full range of postulated accidents.

The instrumentation and control systems are described in detail in Chapter 7.

CRITERION 14 - REACTOR COOLANT PRESSURE BOUNDARY

> The reactor coolant pressure boundary is designed, fabricated, erected and tested so as to have an extremely low probability of abnormal leakage, of rapidly propagating failure and of gross rupture.

Reactor coolant system components are designed in accordance with the ASME Code, Section III, Pump and Valve Code (reactor coolant system pumps), and ANSI B31.7 (see Section 4 for codes and effective dates). Quality control, inspection, and testing as required by these standards and allowable reactor pressure-temperature operations ensure the integrity of the reactor coolant system.

The reactor coolant system components are considered Class I for seismic design.

CRITERION 15 - REACTOR COOLANT SYSTEM DESIGN

> The reactor coolant system and associated auxiliary, control, and protection system is designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during any condition of normal operation, including anticipated operational occurrences.

The design criteria and bases for the reactor coolant pressure boundary are described in the response to Criterion 14.

The operating conditions established for the normal operation of the plant are discussed in the FSAR and the control systems are designed to maintain the controlled plant variables within these operating limits, thereby ensuring that a satisfactory margin is maintained between the plant operating conditions and the design limits.

The reactor protective system functions to minimize the deviation from normal operating limits in the event of certain anticipated operational occurrences. The results of analyses show that the design limits of the reactor coolant pressure boundary are not exceeded in the event of such occurrences.

CRITERION 16 - CONTAINMENT DESIGN

> Reactor containment and associated systems are provided to establish an essentially leak-tight barrier against the uncontrolled release of radioactivity to the environment and to assure that the containment design conditions important to safety are not exceeded for as long as postulated accident conditions require.

The reactor containment structure, described in Section 5.2, consists of a prestressed concrete cylinder and dome with a reinforced concrete base. A one-quarter inch thick welded steel liner plate is attached to the inside face of the concrete to provide a high degree of leak tightness. Designed as a pressure vessel, the containment structure is capable of withstanding all design postulated accident conditions including a loss-of-coolant accident (LOCA). All containment penetrations are sealed as described in Section 5.2.6. Isolation valves are provided for all piping systems which penetrate the containment, as described in Section 5.2.7.

As an extra measure of safety, an enclosure building completely surrounds the containment. In the event of an accident, the enclosure building filtration region (EBFR), described in Section 6.7.2, is maintained at a slightly negative pressure to preclude leakage to the environment. Potential leakage from the containment is channeled into the enclosure building filtration system as described in Section 6.7. Throughline leakage that can bypass the EBFR is discussed in Section 5.3.4.

CRITERION 17 - ELECTRIC POWER SYSTEMS

> An on site electric power system and an off site electric power system are provided to permit functioning of structures, systems, and components important to safety. The safety

function for each system (assuming the other system is not functioning) is to provide sufficient capacity and capability to assure that (1) specified acceptable fuel design limits and design conditions of the reactor coolant pressure boundary (RCPB) are not exceeded as a result of anticipated operational occurrences; and (2) the core is cooled and containment integrity and other vital functions are maintained in the event of postulated accidents.

The on site electric power supplies, including the batteries, and the on site electric distribution system, have sufficient independence, redundancy, and testability to perform their safety functions assuming a single failure.

Electric power from the transmission network to the on site electric distribution system is supplied by two physically independent circuits (not necessarily on separate rights-of-way), designed and located so as to minimize to the extent practical, the likelihood of their simultaneous failure under operating and postulated accident and environmental conditions. A switchyard common to both circuits is acceptable. Each of these circuits is designed to be available in sufficient time following a loss of all on site AC power supplies and the other off site electric power circuit, to assure that specified acceptable fuel design limits and design conditions of the RCPB are not exceeded. One of these circuits is designed so it is available within a few seconds after a loss-of-coolant accident (LOCA) to assure that core cooling, containment integrity, and other vital safety functions are maintained.

Provisions are included 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, the transmission network, or from the on site electric power supplies.

The off site power supplies system is described in Sections 8.1 and 8.2. The preferred source of auxiliary power for unit shutdown is from or through the reserve station service transformers. System interconnection is provided by four 345 kV circuits. These transmission lines are on a single right-of-way with each line installed on an independent set of structures. A description of the structure routing configuration is described in Section 8.1.2.1.

The combination breaker-and-a-half and double breaker-double bus switching arrangement in the 345 kV substation includes two full capacity main buses. Primary and backup relaying are provided for each circuit along with circuit breaker failure protection. These provisions permit the following:

a. Any circuit can be switched under normal or fault conditions without affecting another circuit.

b. Any single circuit breaker can be isolated for maintenance without interrupting the power or protection to any circuit.

c. Short circuits on any section of bus are isolated without interrupting service to any element other than those connected to the faulty bus section.

d. The failure of any circuit breaker to trip within a set time initiates the automatic tripping of the adjacent breakers and thus may result in the loss of a line or generator for this contingency condition; however, power can be restored to the good element in less than eight hours by manually isolating the fault with appropriate disconnect switches.

Overhead lines from the switchyard to the reserve station service transformers are separated at the switchyard structure and are carried on separate towers. These transformers are located near each Unit, and are physically isolated from the normal station service transformers and from the main transformers.

In the event of loss of power from the normal station service transformer, there is an immediate automatic transfer of auxiliary loads to the Unit 2 reserve station service transformer. In the unlikely event that power is not available from this source, and from the On site Emergency Diesel mentioned below, the operator can manually connect emergency bus A-5 (24E) to Unit 3 bus 34A or 34B. By means of interlocked circuit breakers, the Unit 2 post accident loads can be fed from this source.

The on site power supply system is described in Sections 8.3 and 8.5. Two full capacity, separate and redundant batteries are provided for all DC loads and for 120 volt AC vital instrument loads.

In the event that off site power is not available when needed, a “start” signal is given to both emergency diesel generators (DG). These generators and their auxiliaries are entirely separate and redundant, and each generator feeds one 4,160 volt emergency bus. A generator is automatically connected to its bus only if there is no bus voltage and only if the dead bus did not result from protective relay action.

The electric power distribution system is described in Section 8.7. The redundancy of the power sources is enhanced by separate and redundant auxiliary power and control distribution systems. A single failure and any possible related failures in that channel cannot adversely affect equipment and components on the other redundant channel.

Due to the redundancy and separation of power supplies, distribution and control required for vital functions, all components can be readily inspected and tested. Similarly, most subsystems can be tested in their entirety.

CRITERION 18 - INSPECTION AND TESTING OF ELECTRIC POWER SYSTEMS

Electric power systems important to safety are designed to permit appropriate periodic inspection and testing of important areas and features, such as wiring, insulation, connections, and switchboards, to assess the continuity of the systems and the condition of their components. The systems shall be designed with a capability to test periodically (1) the operability and functional performance of the components of the systems, such as on site power sources, relays, switches, and buses, and (2) the operability of the systems as a whole and, under conditions as close to design as practical, the full operation sequence that brings the systems into operation, including operation of applicable portions of the protection system, and the transfer of power among the nuclear power unit, the off site power system, and the on site power system.

The operability and functional performance of the components of these systems are verified by periodic inspections and tests as described in Chapter 8.

To verify that the emergency power system will properly respond within the required time limit when required, the following tests are performed:

a. Manually initiated demonstration of the ability of the diesel-generators to start, synchronize and deliver power up to 2750 kW continuous, when operating in parallel with other power sources. Normal unit operation will not be affected.

b. Demonstration of the readiness of the on site generator system and the control systems of vital equipment to automatically start, or restore to operation, the vital equipment by initiating an actual loss of all normal AC station service power. This test will be conducted during each refueling interval.

   Demonstration of the automatic sequencing equipment during normal unit operation. This test exercises the control and indication devices, and may be performed any time, as the sequencing equipment is redundant to normal operations. If there is a safety injection actuation signal while the test is underway, it takes precedence and immediately cancels the test. The equipment then responds to the safety injection actuation signal in the manner described in Section 8.3.

Since operation of the protective system will be infrequent, each system is periodically and routinely tested to verify its operability. Each channel of the protective systems, including the sensors up to the final protection element, is capable of being checked during reactor operation. The output circuit breakers are provided to permit individual testing during plant operation. See Chapters 7 and 8 for further details.

CRITERION 19 - CONTROL ROOM

A control room is provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents (LOCA). Adequate radiation protection is provided to permit access and occupancy of the control room under accident conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident.

Equipment at appropriate locations outside the control room is provided (1) with a design capability for prompt hot shutdown of the reactor, including necessary instrumentation and controls to maintain the unit in a safe condition during hot shutdown, and (2) with a potential capability for subsequent cold shutdown of the reactor through the use of suitable procedures.

The control room is provided with two separate air conditioning systems and two particulate, absolute, charcoal filter unit assemblies, an airborne radioactivity detector in the fresh air supply line and dampers which act to shunt the intake air through the filters in the event of a high airborne radioactivity level. The dampers are automatically actuated from the control room monitors. Acting on a high radiation level indication, the fresh air dampers close and recirculation dampers open to provide a complete closed cycle ventilation mode with a portion of the air stream being drawn through the HEPA-charcoal filter assembly. In addition, an area radiation monitor is provided to indicate and alarm on high radiation level.

In the event the operator is forced to abandon the control room, a hot shutdown panel (C21) provide the instrumentation and control necessary to maintain the plant in the hot shutdown condition (see Section 7.6.4). The potential capability for bringing the plant to a shutdown is also provided outside the control room.

Fire Shutdown System Panels located outside the control room contain the instruments and controls necessary to achieve a hot shutdown condition should the control room become uninhabitable due to fire (see Section 7.6.5). The Fire Shutdown Panel can be utilized for any emergency event which requires control room evacuation.

Not all indicators and controls provided on the Fire Shutdown Panel are available for all fires. Alternate methods of compliance are documented in the Millstone Unit 2 10 CFR 50 Appendix R Compliance Report.

CRITERION 20 - PROTECTION SYSTEM FUNCTIONS

> The protection system is designed (1) to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences and (2) to sense accident conditions and to initiate the operation of systems and components important to safety.

The reactor is protected by the Reactor Protective System from reaching a condition that could result in exceeding acceptable fuel design limits as a result of anticipated operational occurrences (ANS-N18.2, Condition II). The Protective System is designed to monitor the reactor operating conditions and initiate a reactor trip if any of the following measured variables exceeds the operating limits:

a. High power level (variable, highest of thermal or neutron flux).

b. High pressurizer pressure.

c. Thermal margin (variable low pressure).

d. Turbine trip (equipment protection only).

e. Low reactor coolant flow.

f. Low steam generator level.

g. Low steam generator pressure.

h. Local power density.

i. High containment pressure.

The Engineered Safeguards Actuation System detects accident conditions and initiates the Safety Features Systems which are designed to localize, control, mitigate, and terminate such accidents. The Engineered Safeguards Actuation System protects the general public from the release of radioactivity by actuating components that cool the reactor core, depressurize the containment, isolate the containment, and filter any containment leakage (see Section 7.3). The following parameters are continuously monitored;

a. Low pressurizer pressure.

b. High/high-high containment pressure.

c. Containment gaseous and particulate radiation.

d. Low steam generator pressure.

e. High fuel handling area radiation.

f. Low refueling water storage tank level.

g. Emergency bus undervoltage.

The Auxiliary Feedwater Automatic Initiation System (AFAIS) provides a dedicated source of feedwater of sufficient capacity to remove decay heat and sensible heat following casualty situations. Automatic initiation of auxiliary feedwater occurs in response to a low Steam Generator level in a two out of four (2 of 4) channel auctioneered matrix (see Section 7.3.2.2.h).

CRITERION 21 - PROTECTION SYSTEM RELIABILITY AND TESTABILITY

The protection system is designed for high functional reliability and inservice testability commensurate with the safety functions to be performed. Redundancy and independence designed into the protection system is sufficient to assure that (1) no single failure results in loss of the protection function, and (2) removal from service of any component or channel does not result in loss of the required minimum redundancy unless the acceptable reliability of operation of the protection system can be otherwise demonstrated. The protection system is designed to permit periodic testing of its functioning when the reactor is in operation, including a capability to test channels independently to determine failures and losses of redundancy that may have occurred.

The protective system is designed to provide a high functional reliability and inservice testability. No single failure will result in the loss of the protective function. The protective channels are independent, e.g., with respect to piping, wire routing, mounting and supply of power. This independence permits testing and the removal from service of any component or channel without loss of the protection function.

Each channel of the protective system, including the sensors up to the final protective element, is capable of being checked during reactor operation. Measurement sensors of each channel used in protective systems are checked by observing outputs of similar channels which are presented on indicators and recorders on the control board. Trip units and logic are tested by inserting a signal into the measurement channel ahead of the trip units and, upon application of a trip level input, observing that a signal is passed through the trip units and the logic to the logic output relays. The logic output relays are tested individually for initiation of trip action. See Chapter 7.

CRITERION 22 - PROTECTION SYSTEM INDEPENDENCE

> The protection system is designed to assure that the effects of natural phenomena, and of normal operating, maintenance, testing and postulated accident conditions on redundant channels do not result in loss of the protection function, or is demonstrated to be acceptable on some other defined basis. Design techniques, such as functional diversity or diversity in component design and principles of operation, is used to the extent practical to prevent loss of the protection function.

The reactor protective systems conform to the provisions of the Institute of Electrical and Electronic Engineers (IEEE) Criteria for Nuclear Power Plant Protection Systems, IEEE-279, 1971. Two to four independent measurement channels, complete with sensors, sensor power supplies, signal conditioning units and bistable trip units, are provided for each protective parameter monitored by the protective systems. The measurement channels are provided with a high degree of independence by separate connection of the channel sensors to the process systems. Power to the channels is provided by independent vital power supply buses. See Section 7.2.

Combustion Engineering Topical Report CENPD-11 (“Reactor Protection System Diversity,” W. C. Coppersmith, C. I. Kling, A. T. Shesler, and B. M. Tashjian CENPD, February 1971) demonstrates that functional diversity has been incorporated in the protective system design.

CRITERION 23 - PROTECTION SYSTEM FAILURE MODES

> The protection system is designed to fail into a safe state or into a state demonstrated to be acceptable on some other defined basis if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air) or postulated adverse environments (e.g., extreme heat or cold, fire, pressure, steam, water, and radiation) are experienced.

Protective system instrumentation has been designed to fail into a safe state or into a state established as acceptable in the event of loss of power supply or disconnection of the system, Redundancy, channel independence, and separation are incorporated in the protective system design to minimize the possibility of the loss of a protection function under adverse environmental conditions. See Sections 7.2 and 7.3.

CRITERION 24 - SEPARATION OF PROTECTION AND CONTROL SYSTEMS

> The protection system is separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems is limited so as to assure that safety is not significantly impaired.

The reactor protective systems are separated from the control instrumentation systems so that failure or removal from service of any control instrumentation system component or channel does not inhibit the function of the protective system. See Section 7.2.

CRITERION 25 - PROTECTION SYSTEM REQUIREMENTS FOR REACTIVITY CONTROL MALFUNCTIONS

> The protection system is designed to assure that specified acceptable fuel design limits are not exceeded for any single malfunction of the reactivity control systems, such as accidental withdrawal (not ejection or dropout) of control rods.

Reactor shutdown with CEA's is accomplished completely independent of the control functions since the trip breakers interrupt power to the full length CEA drive mechanisms regardless of existing control signals. The design is such that the system can withstand accidental withdrawal of controlling groups without exceeding acceptable fuel design limits. An analysis of these accidents is given in Section 14.4. The reactor protection system will prevent specified acceptable fuel design limits from being exceeded for any anticipated transients.

CRITERION 26 - REACTIVITY CONTROL SYSTEM REDUNDANCY AND CAPABILITY

> Two independent reactivity control systems of different design principles is provided. One of the systems uses control rods, preferably including a positive means for inserting the rods, and is capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded. The second reactivity control system is capable of reliably controlling the rate of reactivity changes resulting from planned, normal power changes (including xenon burnout) to assure acceptable fuel design limits are not exceeded. One of the systems is capable of holding the reactor core subcritical under cold conditions.

Two independent systems are provided for controlling reactivity changes. The Control Element Drive System (CEDS) controls reactivity change required for power changes and power distribution shaping, and is also used for reactor protection. The boric acid shim control compensates for long term reactivity changes such as those associated with fuel burnup, variation in the xenon and samarium concentrations, and plant cooldown and heatup. See Sections 7.4.2 and 9.2.2.1.

Either system acting independently is capable of making the core subcritical from a hot operating condition and holding it subcritical in the hot standby condition at 532°F.

Either system is able to insert negative reactivity at a sufficiently fast rate to prevent exceeding acceptable fuel design limits as the result of a power change (i.e., the positive reactivity added by burnup of xenon).

The boron addition system is capable of holding the reactor core subcritical under cold conditions.

CRITERION 27 - COMBINED REACTIVITY CONTROL SYSTEMS CAPABILITY

> The reactivity control system is designed to have a combined capability, in conjunction with poison addition by the emergency core cooling system, of reliably controlling reactivity changes to assure that under postulated accident conditions and with appropriate margin for stuck rods the capability to cool the core is maintained.

The combined capability of the reactor control systems in conjunction with dissolved boron addition by the safety injection system is such that under postulated accident conditions, even with the CEA of highest worth stuck out of the core, the core would be maintained in a geometry which assures adequate short and long term cooling. See Criteria 26 and 28.

CRITERION 28 - REACTIVITY LIMITS

> The reactivity control systems are designed with appropriate limits on the potential amount of rate of reactivity increase to assure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core. These postulated reactivity accidents include consideration of ejection (unless prevented by positive means) rod dropout, steam line rupture, changes in reactor coolant temperature and pressure, and cold water addition.

The basis for selecting the number of control element assemblies in the core includes assuring that the reactivity worth of any one assembly is within a preselected maximum value. The control element assemblies have been separated into sets: a shutdown set and a regulating set further subdivided into groups as necessary. Administrative procedures and interlocks are used to permit only one shutdown group to be withdrawn at a time, and to permit withdrawal of the regulating groups only after the shutdown groups are fully withdrawn. The regulating groups are programmed to move in sequence and within limits that prevent the rates of reactivity change and the worth of individual assemblies from exceeding limiting values. See Sections 7.4.2, 14.4.1, 14.4.2, and 14.4.3.

The reactor coolant pressure boundary and reactor vessel internals are designed to be capable of accommodating without rupture, and with limited plastic deformation, the static and dynamic loads associated with an inadvertent and sudden release of energy to the coolant such as that resulting from CEA ejection, CEA drop, steam line rupture or cold water addition. See Sections 14.4.8, 14.4.9, and 14.1.5.

The boric acid system rate of reactivity addition is too slow to cause rupture of the reactor coolant pressure boundary or disturb the reactor pressure vessel internals.

CRITERION 29 - PROTECTION AGAINST ANTICIPATED OPERATIONAL OCCURRENCES

> The protection and reactivity control systems are designed to assure an extremely high probability of accomplishing their safety functions in the event of anticipated operational occurrences.

Anticipated operational occurrences have been considered in the design of the protection and reactivity control systems. As is demonstrated in the safety analysis in Chapter 14 and the Combustion Engineering Report CENPD-11 (“Reactor Protection System Diversity”, W. C. Coppersmith, Cl. L. Kling, A. T. Shesler, and B. M. Tashjian, CENPD-11, February 1971), the design is adequate to minimize the consequences of such occurrences and assures that the health and safety of the public is protected from the consequences of such occurrences.

The adherence to a detailed program for quality assurance, careful attention to design, component selection and system installation, coupled with the design features of redundancy, independence, and testability will assure that a high probability exists that the protection and reactivity control systems will accomplish their functions. See Criteria 21 through 26.

CRITERION 30 - QUALITY OF REACTOR COOLANT PRESSURE BOUNDARY

> Components which are part of the reactor coolant pressure boundary are designed, fabricated, erected and tested to the highest quality standards practical. Means are provided for detecting and, to the extent practical, identifying the location of the source of reactor coolant leakage.

The reactor coolant pressure boundary components have been designed, fabricated, erected and tested in accordance with the ASME Code Section III, 1971 through summer 1971 Addenda and ANSI B31.7, 1969 as specified in Criterion 14. Replacement steam generator subassemblies were fabricated in accordance with ASME Code Section III 1983 through summer 1984 Addenda.

The replacement reactor vessel closure head including all nozzles (CEDM, HJTC, ICI and the vent) is constructed in accordance with ASME Boiler and Pressure Vessel Code, Section III, Subsection NB, 1998 Edition through 2000 Addenda.

Containment sump instrumentation is used to detect reactor coolant system leakage by providing information on rate of rise of sump levels and frequency of sump pump operation. Flow instrumentation indicates and records makeup flow rate and volumes from the primary water system. This instrumentation allows detection of suddenly occurring leaks or those which are gradually increasing. The containment air monitoring system (see Section 7.5.6) provides an additional means of reactor coolant system leakage detection.

CRITERION 31 - FRACTURE PREVENTION OF REACTOR COOLANT PRESSURE BOUNDARY

> The reactor coolant pressure boundary is designed with sufficient margin to assure that when stressed under operating, maintenance, testing, and postulated accident conditions (1) the boundary behaves in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design reflects consideration of service temperatures and other conditions of the boundary material under operating, maintenance, testing and postulated accident conditions and the uncertainties in determining (1) material properties, (2) the effects of irradiation on material properties, (3) residual, steady state and transient stresses, and (4) size of flaws.

Carbon and low alloy steel materials which form part of the pressure boundary meet the requirements of the ASME Code, Section III, paragraph N-330 at a temperature of +40°F. (Ref. Section 4.2.2). The actual nilductility transition temperature (NDTT) of the materials has been determined by drop weight tests in accordance with ASTM-E-208. For the reactor vessel base metals, Charpy tests were also performed and the results used to plot a Charpy transition curve. To address changes in regulations, the original design requirements of N-330 were supplemented and the materials' initial nil-ductility reference temperatures (RT_{NDT}) were conservatively established based upon available or supplemental material toughness testing. In the case of the replacement steam generators, the materials were required to satisfy NB-2331 and RT_{NDT} values were established to satisfy current requirements.

Carbon and low alloy steel materials including weld filler metal which form part of the reactor pressure boundary for replacement reactor vessel closure head satisfy ASME Section III, NB 2000. Actual NDTT was established by drop weight test in accordance with ASTM-E-208 at - 40°F. RT_{NDT} of the replacement head based materials was established by Charpy V-notch test at - 40°F. Charpy transition curves were plotted using test data for the base material of the replacement reactor vessel head.

All the reactor coolant pressure boundary components are constructed in accordance with the applicable codes and comply with the test and inspection requirements of these codes. These test inspection requirements assure that flaw sizes are limited so that the probability of failure by rapid propagation is extremely remote. Particular emphasis is placed on the quality control applied to the reactor vessel, on which tests and inspections exceeding code requirements are performed. The tests and inspections performed on the reactor vessel are summarized in Section 4.6.5.

The reactor vessel beltline materials receive sufficient neutron irradiation to cause embrittlement (an increase in RT_{NDT}). To provide conservative margins against nonductile or rapidly propagating failure, several techniques are employed. Operating limits which account for the

RT_{NDT} of all pressure boundary materials, both unirradiated and irradiated, are established in accordance with the requirements of 10 CFR 50 Appendix G (Additional details are provided in Section 4.5.1). In addition, compliance with 10 CFR 50.61 assures that the shift in the transition temperature of the reactor vessel beltline materials provides adequate margins of safety against severe pressurized thermal shock events.

To assure that the reactor vessel beltline materials are behaving in the predicted manner, a reactor vessel material surveillance program is conducted (See Criterion 32 and Section 4.6.2). Toughness testing of unirradiated reactor vessel materials was performed to establish the baseline, and the irradiated surveillance materials are periodically tested as surveillance capsules are removed during the plant's design life, in accordance with the requirements of 10 CFR 50, Appendix H.

The activation of the safety injection systems introduces highly borated water into the reactor coolant system at pressures significantly below operating pressures and will not cause adverse pressure or reactivity effects.

The thermal stresses induced by the injection of cold water into the vessel have been examined. Analysis shows the there is no gross yielding across the vessel wall using the minimum specified yield strength in the ASME Boiler and Pressure Vessel Code, Section III. (Ref. Section 4.5.4).

Adverse effects that could be caused by exposure of equipment or instrumentation to containment spray water is avoided by designing the equipment or instrumentation to withstand direct spray or by locating it or protecting it to avoid direct spray.

CRITERION 32 - INSPECTION OF REACTOR COOLANT PRESSURE BOUNDARY

> Components which are part of the reactor coolant pressure boundary are designed to permit (1) periodic inspection and testing of important areas and features to assess their structural and leak-tight integrity, and (2) an appropriate materials surveillance program for the reactor pressure vessel.

Provisions are made for inspection, testing, and surveillance of the Reactor Coolant System boundary as required by ASME Boiler and Pressure Vessel Code, Section XI.

The Reactor vessel surveillance program was designed in accordance with ASTM E185. It complies with ASTM E185-73 and 10 CFR 50, Appendix H. Section 4.6.3 presents the details of the reactor surveillance program. Sample pieces taken from the same shell plate material used in fabrication of the reactor vessel are installed between the core and the vessel inside wall. These samples will be removed and tested at intervals during vessel inside wall. These samples will be removed and tested at intervals during vessel life to provide an indication of the extent of the neutron embrittlement of the vessel wall. Charpy tests will be performed on the samples to develop a Charpy transition curve. By comparison of this curve with the Charpy curve and drop weight tests for specimens taken at the beginning of the vessel life, the change of NDTT will be determined and operating instructions adjusted as required.

CRITERION 33 - REACTOR COOLANT MAKEUP

> A system to supply reactor coolant makeup for protection against small breaks in the reactor coolant pressure boundary is provided. The system safety function shall be to assure that specified acceptable fuel design limits are not exceeded as a result of reactor coolant loss due to leakage from the reactor coolant pressure boundary and rupture of small piping or other small components which are part of the boundary. The system is designed to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished using the piping, pumps and valves used to maintain coolant inventory during normal reactor operation.

Reactor Coolant System (RCS) makeup during normal operation is provided by the Chemical and Volume Control System (CVCS) which includes three positive displacement charging pumps rated at 44 gpm each. Two operating CVCS pumps are capable of making up the flow loss for leaks in the reactor coolant boundary of up to 0.250 inches equivalent diameter. Two CVCS pumps are sufficient to makeup for a 0.250 inch equivalent diameter RCS break assuming either: 1) minimum letdown with no RCS leakage or 2) letdown isolated with maximum Technical Specification allowed leakage. This CVCS design results in a substantial RCS steady state pressure that is well above the shutoff head of the high pressure safety injection pumps. The above described CVCS capability fulfills the intent of Criterion 33. Information on CVCS is contained in Section 9.2.

CRITERION 34 - RESIDUAL HEAT REMOVAL

> A system to remove residual heat is provided. The system safety function is to transfer fission product decay heat and other residual heat from the reactor core at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary are not exceeded.
>
> Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities are provided to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

Residual heat removal capability is provided by the shutdown cooling system for reactor coolant temperature less than 300°F (see Section 9.3). For temperatures greater than 300°F, this function is provided by the steam generators (see Section 10.3). Sufficient redundancy, interconnections, leak detection, and isolation capabilities exist with these systems to assure that the residual heat removal function can be accomplished, assuming failure of a single active component. Within appropriate design limits, either system will remove fission product decay heat at a rate such that specified acceptable fuel design limits and the design conditions of the reactor coolant pressure boundary will not be exceeded.

CRITERION 35 - EMERGENCY CORE COOLING

> A system to provide abundant emergency core cooling is provided. The system safety function is to transfer heat from the reactor core following any loss of reactor coolant at a rate such that (1) fuel and clad damage that could interfere with continued effective core cooling is prevented, and (2) clad metal-water reaction is limited to negligible amounts.
>
> Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities is provided to assure that for onsite electrical power system operation (assuming offsite power is not available) and for offsite electrical power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

The emergency core cooling system is discussed in detail in Chapter 6. It consists of the high pressure safety injection subsystem, the low pressure safety injection subsystem, and the safety injection tanks (see Section 6.3).

This system is designed to meet the criterion stated above with respect to the prevention of fuel and clad damage that would interfere with the emergency core cooling function, for the full spectrum of break sizes, and to the limitation of metal-water reaction. Each of the subsystems is fully redundant, and the subsystems do not share active components other than the valves controlling the suction headers of the high and low pressure safety injection pumps. Minimum safety injection is assured even though one of these valves fails to function. These valves are in no way associated with the function of the safety injection tanks.

The ECCS design satisfies the criteria specified in 10 CFR 50.46(b).

CRITERION 36 - INSPECTION OF EMERGENCY CORE COOLING SYSTEM

> The emergency core cooling system is designed to permit appropriate periodic inspection of important components, such as spray rings in the reactor pressure vessel, water injection nozzles, and piping to assure the integrity and capability of the system.

Chapter 6 describes the arrangement and location of the components in the emergency core cooling system. All pumps, the shutdown cooling heat exchangers, and valves and piping external to the containment structure are accessible for physical inspection at any time. All safety injection valves and piping inside the containment structure, and the safety injection tanks, may be inspected during refueling.

The accessibility for inspection of the reactor vessel internals, reactor coolant piping and items such as the water injection nozzles is described in Sections 4.6.3 through 4.6.6.

CRITERION 37 - TESTING OF EMERGENCY CORE COOLING SYSTEM

> The emergency core cooling system is designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components,

(2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

The Emergency Core Cooling System (Safety Injection System) is provided with testing facilities to demonstrate system component operability. Testing can be conducted during normal plant operation with the test facilities arranged not to interfere with the performance of the systems or with the initiation of control circuits, as described in Section 6.3.4.2.

The safety injection system is designed to permit periodic testing of the delivery capability up to a location as close to the core as practical. Periodic pressure testing of the Safety Injection System is possible using the cross connection to the charging pumps in the Chemical and Volume Control System.

The low pressure safety injection pumps are used as shutdown cooling pumps during normal plant cooldown. The pumps discharge into the safety injection header via the shutdown cooling heat exchangers and the low pressure injection lines.

With the plant at operating pressure, operation of safety injection pumps may be verified by recirculation back to the refueling water storage tank. This will permit verification of flow path continuity in the high pressure injection lines and suction lines from the refueling water storage tank.

Borated water from the safety injection tanks may be bled through the recirculation test line to verify flow path continuity from each tank to its associated main safety injection header.

The operational sequence that brings the Safety Injection System into action, including transfer to alternate power sources, can be tested in parts as described in Chapters 6, 7, and 8.

CRITERION 38 - CONTAINMENT HEAT REMOVAL

A system to remove heat from the reactor containment is provided. The system safety function is to reduce rapidly, consistent with the functioning of other associated systems, the containment pressure and temperature following any loss-of-coolant accident and maintain them at acceptably low levels.

Suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities are provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

The containment spray system (Section 6.4) and the containment air recirculation and cooling system (Section 6.5) are provided as redundant, independent systems, each fully capable of reducing the containment pressure and temperature following any loss-of-coolant accident (LOCA) and maintaining them at acceptably low levels.

Sufficient heat removal capability is provided by any of the following combinations of equipment:

a. Two containment spray pumps with associated heat exchangers.

b. Three of the four containment air recirculation and cooling units.

c. One containment spray pump with associated heat exchanger in combination with two containment air recirculation and cooling units.

The containment heat removal systems are provided with suitable interconnections such that each combination of two containment air recirculation and cooling units and one containment spray pump, aligned with the associated shutdown cooling heat exchanger, are provided with cooling water from the same RBCCW header and powered by the same emergency bus. All associated components, such as valves, are likewise powered from the same emergency bus. Each combination of these components is capable of removing heat at a rate greater than required to limit the postaccident containment pressure and temperature. A single failure of any active component does not render the redundant group inoperable.

The containment spray system is provided with containment isolation capabilities in accordance with Criterion 56. The above containment penetration is provided with leak detection capabilities in accordance with Criterion 54.

CRITERION 39 - INSPECTION OF CONTAINMENT HEAT REMOVAL SYSTEM

> The containment heat removal system is designed to permit appropriate periodic inspection of important components, such as the torus, sumps, spray nozzles, piping to assure the integrity and capability of the system.

Major components of the containment spray system are located to permit access for periodic maintenance and inspection. Components of the containment air and recirculation system are located within the containment and are therefore accessible for maintenance and inspection during shutdown.

The containment sump is located in the lowest elevation of the containment at Elevation (-)22-6 and is accessible during reactor shutdown for periodic visual inspections (see Section 6.2).

The containment spray nozzles are accessible for periodic inspection during reactor shutdown.

CRITERION 40 - TESTING OF CONTAINMENT HEAT REMOVAL SYSTEM

> The containment heat removal system is designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the system, and (3) the operability of the system as a whole, and, under conditions as close to the design and practical, the performance of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of the associated cooling water system.

The spray system and the air recirculation and cooling systems in the containment have provisions for online testing to assure system operation, performance and structural and leaktight integrity of the associated components. Testing procedures are described in Sections 6.4.4.2 and 6.5.4.2, respectively.

The containment heat removal systems undergo preoperational testing prior to plant startup. The test procedure is described in Chapter 13.

CRITERION 41 - CONTAINMENT ATMOSPHERE CLEAN UP

> Systems to control fission products, hydrogen, oxygen, and other substances which may be released into the reactor containment are provided as necessary to reduce, consistent with the functioning of other associated systems, the concentration and quality of fission products released to the environment following postulated accidents, and to control the concentration of hydrogen or oxygen and other substances in the containment atmosphere following postulated accidents to assure that containment integrity is maintained.
>
> Each system has suitable redundancy in components and features, and suitable interconnections, leak detection, isolation, and containment capabilities to assure that for offsite electric power system operation (assuming onsite power is not available) its safety function can be accomplished, assuming a single failure.

The containment is not provided with an atmosphere cleanup system. However, a second barrier, the enclosure building, is provided around the containment to collect potential leakage from the containment under postaccident conditions.

The enclosure building filtration system (EBFS) is provided to collect and process potential leakage from the containment during postaccident operation. Potential containment leakage is into the enclosure building filtration region (EBFR) which forms the outer barrier in the double containment boundary. The EBFS is described in Section 6.7. Throughline leakage that can bypass the EBFR is discussed in Section 5.3.4.

The hydrogen control system is provided to mix and monitor the concentration of hydrogen in the containment atmosphere following postulated accidents to assure the containment integrity is maintained. This is discussed in Section 6.6. Reduction of hydrogen concentration is not credited for design basis accidents.

Each of these cleanup systems consist of completely redundant, independent safety function. These are provided with suitable interconnections and separations such that a single failure in any subsystem does not render the redundant subsystem inoperable.

The hydrogen control system is incorporated with containment isolation capabilities for each piping subsystem which penetrates the primary containment. Containment isolation is in accordance with Criterion 56. Provision for leak detection is incorporated in accordance with Criterion 54.

CRITERION 42 - INSPECTION OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEMS

> The containment atmosphere cleanup systems are designed to permit appropriate periodic inspection of important components, such as filter frames, fans, hydrogen recombiners, analyzers, valves, ducts, and piping to assure the integrity and capability of the systems.

The enclosure building filtration system (EBFS) is located to permit access for periodic inspection and maintenance. The components of the hydrogen control system located outside the containment are accessible for periodic inspection and maintenance. The components located inside containment are accessible for inspection and maintenance during shutdown.

The hydrogen control system and EBFS are incorporated with provisions for online testing to demonstrate system operation, performance and integrity. These tests procedures are described in Sections 6.6.4.2 and 6.7.4.2, respectively.

CRITERION 43 - TESTING OF CONTAINMENT ATMOSPHERE CLEANUP SYSTEM

> The containment atmosphere cleanup systems are designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and performance of the active components of the systems such as fans, filters, dampers, pumps, and valves and (3) the operability of the systems as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the systems into operation, including operation of applicable portions of the protection system, the transfer between normal and emergency power sources, and the operation of associated systems.

The enclosure building filtration system (EBFS) and hydrogen control system are incorporated with provisions for online testing. The test procedures are described in Sections 6.7.4.2 and 6.6.4.2, respectively.

The containment atmosphere cleanup systems undergo preoperational tests prior to plant startup. Test procedures are described in Chapter 13.

CRITERION 44 - COOLING WATER

> A system to transfer heat from structures, systems, and components important to safety, to an ultimate heat sink is provided. The system safety function is to transfer the combined heat load of these structures, systems, and components under normal operating and accident conditions.
>
> Suitable redundancy in components and features, and suitable interconnections, leak detection, and isolation capabilities are provided to assure that for onsite electric power system operation (assuming offsite power is not available) and for offsite electric power system operation (assuming onsite power is not available) the system safety function can be accomplished, assuming a single failure.

The RBCCW system, described in Section 9.4, and the service water system, described in Section 9.7.2, are provided to transfer heat from structures, systems, and components important to safety to an ultimate heat sink. The systems are designed to transfer the combined heat load of these structures, systems, and components under normal and accident conditions.

The RBCCW supplies cooling water to components important to safety through two independent headers. One header provides adequate heat removal capability to safely shutdown the plant under accident conditions, but at a lesser rate. Service water is supplied to the RBCCW heat exchangers by two independent headers to assure heat removal capability. Two service water pumps are in continuous operation with a spare pump provided. One pump supplies sufficient heat removal capability for the RBCCW heat exchangers to safely shut down the plant and for accident mitigation.

The RBCCW and service water systems are provided with suitable redundancy in components and suitable interconnections to assure heat removal capability. The systems are designed to enable isolation of system components or headers and to detect system maloperation.

The RBCCW and service water systems are designed to operate with onsite power (assuming offsite power is not available) and with offsite power (assuming onsite power is not available).

The systems are designed such that a single failure in either system will not adversely affect safe operation, accident mitigation, or safe shutdown of the plant.

CRITERION 45 - INSPECTION OF COOLING WATER SYSTEM

> The cooling water system is designed to permit appropriate periodic inspection of important components, such as heat exchangers and piping, to assure the integrity and capability of the system.

The RBCCW system and service water system, excluding underground piping, are designed to permit periodic inspection of important components, such as pumps, heat exchangers, valves and piping to assure the integrity and heat removal capability of the system. The components of the RBCCW system located outside the containment are located in a low radiation area, which permits access for periodic inspection and maintenance during operation. Components of the RBCCW system located inside the containment are accessible for inspection and maintenance during plant shutdown. Inspection of RBCCW system components is described in Section 9.4.4.2. Major service water system components, such as pumps and strainers, are accessible for periodic inspection during normal operation. Inspection of the service water system is described in Section 9.7.2.5.

CRITERION 46- TESTING OF COOLING WATER SYSTEM

> The cooling water system is designed to permit appropriate periodic pressure and functional testing to assure (1) the structural and leaktight integrity of its components, (2) the operability and the performance of the active components of the system, and (3) the operability of the system as a whole and, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for loss-of-coolant accidents (LOCA), including operation of applicable portions of the protection system and the transfer between normal and emergency power sources.

Online testing provisions are incorporated in the RBCCW and service water systems to demonstrate the operability, performance, structural and leaktight integrity of the systems. The RBCCW and service water systems are designed so that under conditions as close to design as practical, the performance shall be demonstrated of the full operational sequence that brings the system into operation, including operation of applicable portions of the protection system, and the transfer between normal and emergency power sources. Testing of the RBCCW and service water systems are described in Sections 9.4.4.2 and 9.7.2.5, respectively.

CRITERION 50 - CONTAINMENT DESIGN BASIS

> The reactor containment structure, including access openings, penetrations, and the containment heat removal system are designed so that the containment structure and its internal compartments can accommodate, without exceeding the design leakage rate and, with sufficient margin, the calculated pressure and temperature conditions resulting from any loss-of-coolant accident. This margin reflects consideration of (1) the effects of potential energy sources which have not been included in the determination of the peak conditions, such as energy in steam generators and energy from metal-water and other chemical reactions that may result from degraded emergency core cooling functioning, (2) the limited experience and experimental data available for defining accident phenomena and containment responses, and (3) the conservatism of the calculational model and input parameters.

The containment structure, including the access openings, penetrations and the containment heat removal system, is designed to withstand a pressure of 54 psig and a temperature of 289°F following a loss-of-coolant accident (LOCA) or a main steam line break accident (see Section 14.8.2). Details of the methods used to analyze the containment structure are described in Section 5.2.2. To obtain an adequate margin of safety, a factored load was selected for a design which allows a 25 percent increase over the calculated postulated accident load.

A high degree of leak tightness is provided by a one-quarter inch thick steel liner plate which completely encloses the interior surface of the containment structure. Components of the liner plate, such as penetration sleeves, personnel locks, and equipment hatch, are designed to meet the requirements of the ASME Boiler and Pressure Vessel Code, Section III (Nuclear Vessels) 1968 Edition through the summer 1969 addenda Paragraph N-1211. Further description of the liner plate is contained in Section 5.2.3.

As a further check on the design a structural integrity test, composing a test pressure load of 115 percent of the design accident pressure load, is conducted prior to operation. In addition to this, a leak rate test will be conducted prior to operation and at certain intervals during operation. Details of the leak rate test are provided in Section 5.2.8.1.

CRITERION 51 - FRACTURE PREVENTION OF CONTAINMENT PRESSURE BOUNDARY

> The reactor containment boundary is designed with sufficient margin to assure that under operating, maintenance, testing, and postulated accident conditions (1) its ferritic materials behave in a nonbrittle manner and (2) the probability of rapidly propagating fracture is minimized. The design reflects consideration of service temperatures and other conditions of the containment boundary material during operation, maintenance, testing, and postulated accident conditions, and the uncertainties in determining (1) material properties, (2) residual, steady state, and transient stresses, and (3) size of flaws.

The containment consists of a prestressed reinforced concrete cylinder and dome connected to and supported by a massive reinforced concrete slab. A one-quarter inch thick steel liner plate is attached to the inside surface of the concrete containment and its penetrations. Consideration has been given to both design and construction techniques to assure the containment pressure boundary behaves in a ductile manner and the probability of a rapidly propagating fracture is minimized.

The liner plate is designed to carry no load, and serves only as a leaktight barrier. Analytical calculations of the strains under an extreme and most improbably set of load conditions indicate the strains are well within the ductile limits of the material. The analytical approach to liner design is presented in the Bechtel Corporation Proprietary Report B-TOP-1.

At all penetrations the liner plate is thickened using the 1968 ASME Code, Section III for Class B Vessels as a guide to limit stress concentrations.

Provisions, as described in Section 5.2.5.1.1, are made to prevent a potential internally generated missile from rupturing the liner plate.

Materials for the penetrations require satisfactory Charpy V-notch impact test results. All penetrations are stress relieved. The construction materials selected for the liner plate and penetrations are given in Section 5.2.1.

Additional details concerning the construction techniques and inspection provisions are outlined in Section 5.9.3.5.

CRITERION 52 - CAPABILITY FOR CONTAINMENT LEAKAGE RATE TESTING

> The reactor containment and other equipment which may be subjected to containment test conditions are designed so that periodic integrated leakage rate testing can be conducted at containment design pressure.

The reactor containment and other equipment which is subjected to containment test conditions are designed so that periodic integrated leakage rate testing can be conducted at containment design pressure. The test procedure is described in Section 5.2.8.

CRITERION 53 - PROVISIONS FOR CONTAINMENT TESTING AND INSPECTION

> The reactor containment is designed to permit (1) appropriate periodic inspection of all important areas, such as penetrations, (2) an appropriate surveillance program, and (3) periodic testing at containment design pressure of the leak tightness of penetrations which have resilient seals and expansion bellows.

The reactor containment is designed to permit appropriate periodic testing of all important areas. Details of the containment testing and inspection are discussed in Section 5.2.8.

CRITERION 54 - PIPING SYSTEMS PENETRATING CONTAINMENT

> Piping systems penetrating primary reactor containment are provided with leak detection, isolation, and containment capabilities having redundancy, reliability, and performance capabilities which reflect the importance to safety of isolating these piping systems. Such piping systems are designed with a capability to test periodically the operability of the isolation valves and associated apparatus and to determine if valve leakage is within acceptable limits.

Piping systems penetrating containment are provided with suitable redundancy to assure the systems function adequately during postulated accidents such that failure of a portion of a system will not create a hazard to safe unit operation. Piping systems are provided with containment isolation valves in accordance with the requirements of Criterion 55, 56, and 57. Containment isolation valves have been selected and tested to provide adequate operation at maximum flow conditions. Provisions are incorporated for leak detection and performance testing of those piping systems penetrating the containment (Section 5.2.7.4.2).

CRITERION 55 - REACTOR COOLANT PRESSURE BOUNDARY PENETRATING CONTAINMENT

> Each line that is part of the reactor coolant pressure boundary and that penetrates primary reactor containment is provided with containment isolation valves as follows, unless it can

be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment are located as close to containment as practical and upon loss of actuating power, automatic isolation valves are designed to take the position that provides greater safety.

Other appropriate requirements to minimize the probability or consequences of an accidental rupture of these lines or of lines connected to them are provided as necessary to assure adequate safety. Determination of the appropriateness of these requirements, such as higher quality in design, fabrication, and testing, additional provisions for inservice inspection, protection against more severe natural phenomena, and additional isolation valves and containment, include consideration of the population density, use characteristics, and physical characteristics of the site environs.

For those piping systems penetrating the containment and connected directly to the reactor coolant pressure boundary, isolation provisions have been incorporated. Section 5.2.7 indicates applicable valve arrangements, a complete description of penetrations and valve position on air/power failure.

Provisions are made for leak testing as described in Section 5.2.7.4.2.

CRITERION 56 - PRIMARY CONTAINMENT ISOLATION

Each line that connects directly to the containment atmosphere and penetrates primary reactor containment is provided with containment isolation valves as follows, unless it can be demonstrated that the containment isolation provisions for a specific class of lines, such as instrument lines, are acceptable on some other defined basis:

(1) One locked closed isolation valve inside and one locked closed isolation valve outside containment; or

(2) One automatic isolation valve inside and one locked closed isolation valve outside containment; or

(3) One locked closed isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment; or

(4) One automatic isolation valve inside and one automatic isolation valve outside containment. A simple check valve may not be used as the automatic isolation valve outside containment.

Isolation valves outside containment are located as close to the containment as practical and upon loss of actuating power, automatic isolation valves are designed to take the position that provides greater safety.

For those piping system penetrating the containment and connected directly to the containment atmosphere, isolation provisions have been incorporated. Section 5.2.7 indicates the applicable valve arrangements, a complete description of penetrations and valve position on air/power failure.

CRITERION 57 - CLOSED SYSTEM ISOLATION VALVES

Each line that penetrates primary reactor containment and is neither part of the reactor coolant pressure boundary not connected directly to the containment atmosphere has at least one containment isolation valve which shall be either automatic, or locked closed, or capable of remote manual operation. This valve is outside containment and located as close to the containment as practical. A simple check valve may not be used as the automatic isolation valve.

For those piping systems penetrating the containment which are neither part of the reactor coolant pressure boundary nor connected directly with the containment atmosphere, isolation provisions have been incorporated.

Section 5.2.7 indicates applicable valve arrangements, a complete description of penetrations and valve position on air/power failure.

CRITERION 60 - CONTROL OF RELEASES OF RADIOACTIVE MATERIALS TO THE ENVIRONMENT

The nuclear power unit design includes means to control suitably the release of radioactive materials in gaseous and liquid effluents and to handle radioactive solid waste produced during normal reactor operation, including anticipated operational occurrences. Sufficient holdup capacity is provided for retention of gaseous and liquid effluents containing radioactive materials, particularly where unfavorable site environmental conditions can be expected to impose unusual operational limitations upon the release of such effluents to the environment.

The radioactive waste processing system (RWS), as described in Section 11.1, is designed to provide controlled handling and disposal of liquid, gaseous, and solid wastes from Millstone Unit 2. The RWS is designed to ensure that the general public and plant personnel are protected against exposure to radioactive material in accordance with 10 CFR Part 20, Sections 1301 and 1302, and Appendix B and 10 CFR Part 50, Appendix I.

All liquid and gaseous radioactive releases from the RWS are designed to be accomplished on a batch basis. All radioactive materials are sampled prior to release to ensure compliance with 10 CFR Part 20, Sections 1301 and 1302, and Appendix B and 10 CFR Part 50, Appendix I and to determine release rates. Radioactive materials which do not meet release requirements will not be discharged to the environment. The RWS is designed with sufficient holdup capacity and flexibility for reprocessing of wastes to ensure release limitations are met.

The RWS is designed to preclude the inadvertent release of radioactive material.

All storage tanks in the clean liquid waste and gaseous waste systems are provided with valve interlocks which prevent the addition of waste to a tank which is being discharged to the environment. Each discharge path from the RWS is provided with a radiation monitor which alerts unit personnel and initiates automatic closure of redundant isolation valves to prevent further releases in the event of noncompliance to 10 CFR Part 20, Sections 1301 and 1302, and Appendix B.

Section 11.1.5 describes the plant design for the handling of solid wastes.

CRITERION 61 - FUEL STORAGE AND HANDLING AND RADIOACTIVITY CONTROL

> The fuel storage and handling, radioactive waste and other systems which may contain radioactivity are designed to assure adequate safety under normal and postulated accident conditions. These systems are designed (1) with a capability to permit appropriate periodic inspection and testing of components important to safety, (2) with suitable shielding for radiation protection, (3) with appropriate containment, confinement, and filtering systems, (4) with a residual heat removal capability having reliability and testability that reflects the importance to safety of decay heat and other residual heat removal, and (5) to prevent significant reduction in fuel storage coolant inventory under accident conditions.

Systems for fuel storage and handling, and all systems containing radioactivity are designed to ensure adequate safety under normal and postulated accident conditions. Design of these systems are described in the sections listed below:

System	Section
Reactor Coolant System	4.0
Engineering Safety Features Systems	6.0

System	Section
Auxiliary Systems	9.0
Radioactive Waste Processing System	11.0

All components important to the safety of these systems are located to permit periodic inspection as required. Suitable shielding, as described in Section 11.2, is provided for these components to protect plant personnel and to allow inspection and testing.

To ensure the containment and confinement of radioactivity, all components are designed and tested in accordance with accepted Codes and Standards. All system components are visually inspected and adjusted, if required, to ensure correct installation and arrangement. The completely installed systems were subject to acceptance tests or preoperation tests as described in Chapter 13 to ensure the integrity of the systems.

The spent fuel pool cooling system described in Section 9.5, is designed to ensure adequate decay heat removal from stored fuel. Sections 5.4.3 and 9.5 describe how the spent fuel pool is designed to prevent significant reduction in fuel storage coolant inventory.

CRITERION 62 - PREVENTION OF CRITICALITY IN FUEL STORAGE AND HANDLING

> Criticality in the fuel storage and handling system is prevented by physical systems or processes, preferably by use of geometrically safe configurations.

New fuel assemblies are stored in dry racks in parallel rows at elevation 38 feet 6 inches of the auxiliary building. The base of the new fuel racks at elevation 38 feet 6 inches minimizes the possibility of flooding the fuel assemblies. Nevertheless, the new fuel racks maintain a center to center distance of 20.5 inches, large enough to prevent criticality in the unlikely event of flooding with unborated water. Additional details of new fuel storage are given in Sections 9.8.2.1.1and 9.8.4.1.1.

Spent fuel assemblies are stored in parallel rows at the bottom of the spent fuel pool. The racks are separated into 4 regions, designated Regions 1, 2, 3, and 4.

Fuel assemblies used at Millstone Unit 2 may include reduced enrichment fuel rods adjacent to guide thimbles and reduced enrichment axial blanket regions. The criticality analyses are performed using a single enrichment in all fuel rods that is the highest initial planar average U-235 enrichment of the axial regions in the fuel assembly. This averaged enrichment is designated as the initial planar average enrichment.

Region 1 can store, in a 2 out of 4 storage pattern, any fuel assembly with a maximum initial planar average enrichment up to 4.85 weight percent U-235. The other two locations in the 2 out of 4 storage pattern are designated as Restricted Locations (shown in Figure 9.8–7). Fuel storage rack locations designated as Restricted Locations in Figure 9.8–7 shall remain empty. No fuel assembly, no Non-standard Fuel Configuration, no non-fuel component, nor any hardware/material of any kind may be stored in a Restricted Location.(1)

Regions 2 and 4 use fuel burnup credit and store fuel assemblies in a 3 out of 4 storage pattern, in which the fourth location in a 2 x 2 storage array is designated as a Restricted Location per Figure 9.8–7.

Regions 1 and 2 contain Boraflex panels which are no longer credited as neutron absorbers.

Region 3 uses fuel burnup credit and has all storage locations available. In addition, fuel assemblies stored in Region 3 must contain either three Borated Stainless Steel Poison Rodlets (installed in the assembly's center guide tube and in two diagonally opposite guide tubes) or a full length, full strength Control Element Assembly (CEA).

There are also Non-standard Fuel Configurations in the spent fuel pool (SFP). A Non-standard Fuel Configuration is an object containing fuel that does not conform to the standard fuel configuration. The standard fuel configuration is a 14 x 14 array of fuel rods (or fuel rods replaced by un-enriched fuel rods or stainless steel rods) with five (5) guide tubes that occupy four lattice pitch locations each. Fuel in any other array is a “Non-standard Fuel Configuration.” Reconstituted fuel in which one or more fuel rods have been replaced by either un-enriched fuel rods or stainless steel rods is considered to be a standard fuel configuration.

Note that each of the Non-standard Fuel Configurations must have a separate criticality analysis which may allow storage in one or multiple Regions, and which may or may not require Borated Stainless Steel Poison Rodlets or a CEA if stored in Region 3.

GDC 62 states that the “Criticality in the fuel storage and handling system shall be prevented by physical systems or processes, preferably by use of geometrically safe configurations.” As detailed above, the Region 1, 2, 3, and 4 storage racks, require more than just fuel geometry alone for reactivity control. All four regions credit soluble boron in the spent fuel pool water. Regions 1, 2, and 4 credit Restricted Locations per Figure 9.8–7. Regions 2, 3, and 4 use fuel burnup credit. Region 3 requires that fuel assemblies contain either three Borated Stainless Steel Poison Rodlets or a full length, full strength CEA (note that the criticality analysis of a given Non-standard Fuel Configuration may qualify it for Region 3 storage without these inserts). Administrative controls are used to ensure proper placements of Borated Stainless Steel Poison Rodlets and CEAs, use of soluble boron and fuel burnup credit, and control of Restricted Locations. Further, for accident conditions, soluble boron is credited in the spent fuel pool water. The NRC has concurred that the credit for these neutron poisons, soluble boron, fuel burnup credit, Restricted Locations, and associated administrative controls are acceptable in meeting the requirements of GDC 62.

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(1) Note that Region 1 and 2 SFP rack storage locations contain removable Boraflex panel boxes which house the Boraflex panels. The Boraflex panel boxes were manufactured as an integral part the original SFP racks and as such are NOT stored components in SFP rack storage locations. Criticality analysis has shown that the Restricted Locations are acceptable with or without the Boraflex panel boxes.

Both the spent fuel and new fuel storage racks are designed to preclude any deformation of the racks during earthquake loads that would reduce the center to center spacing to a point where the fuel would approach criticality.

Fuel handling equipment is designed to ensure safe handling of fuel assemblies and to prevent criticality. Section 9.8.4 describes the safety features of the fuel handling equipment.

CRITERION 63 - MONITORING FUEL AND WASTE STORAGE

> Appropriate systems are provided in fuel storage and radioactive waste systems and associated handling areas (1) to detect conditions that may result in loss of residual heat removal capability and excessive radiation levels and (2) to initiate appropriate safety actions.

Section 9.5.2.1 describes the monitoring and alarm instrumentation provided for the spent fuel storage system to detect conditions that may result in loss of decay heat removal capability and excessive radiation levels. Section 7.5.6 describes the monitoring provisions for radioactive waste handling and storage areas.

CRITERION 64 - MONITORING RADIOACTIVITY RELEASES

> Means are provided for monitoring the reactor containment atmosphere, spaces containing components for recirculation of loss-of-coolant accident fluids, effluent discharge paths, and the plant environs for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

Containment radiation is monitored by gaseous and particulate monitors as described in Sections 7.5.1.2 and 7.5.6.3.

Radiation in effluent discharge paths and the plant environs are monitored as described in Sections 7.5.6.2 and 7.5.6.3.