Millstone Power Station Unit 3
Safety Analysis Report

Chapter 1: Introduction and General Description of Plant

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CHAPTER 1 – INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

This Final Safety Analysis Report (FSAR) has been prepared with the guidance of Regulatory Guide 1.70, Revision 3, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants, LWR Edition, dated November 1978. The report is intended to be responsive to the guide, to existing regulations, and to NUREG-75/087, Standard Review Plan for the Review of Safety Analysis Reports for Nuclear Power Plants, LWR Edition.

1.1 INTRODUCTION

This report was submitted in support of an application by the companies listed in the General Information Section of the application (the Applicants) for a Class 103 permit for a facility operating license to operate a nuclear power plant, designated as Millstone Nuclear Power Station - Unit 3 (Millstone 3). This plant is located on a site in the town of Waterford, New London County, Connecticut, on the north shore of Long Island Sound.

Millstone 3 uses a pressurized water type nuclear steam supply system (NSSS) furnished by Westinghouse Electric Corporation (WNES) and a turbine-generator furnished by the General Electric Company (GE). The remainder of the unit, including a subatmospheric reactor containment, was designed and constructed by the Applicants, with the assistance of their representative, Northeast Utilities Service Company (NUSCo.), and their architect-engineer, Stone & Webster Engineering Corporation (SWEC).

The core was originally designed for a warranted power output of 3,411 MWt, which was the original license application rating. This output, combined with the reactor coolant pump heat output of 14 MWt, gave an NSSS warranted output of 3,425 MWt.

The core has been re-analyzed for a power output of 3709 MWt which, when combined with the revised reactor coolant pump output of 16 MWt, gives a 100% NSSS power output of 3725 MWt.

All steam and power conversion equipment, including the turbine-generator, has the capability to generate a maximum calculated gross output of approximately 1305 MWe. When the NSSS is operating at its warranted output of 3725 MWt, the net electrical output is approximately 1254 MWe.

The project schedule was based on a fuel loading date of November 1, 1985, and an anticipated commercial operation date of May 1, 1986. The Low Power License (5 percent) was issued by the NRC November 25, 1985, the Full Power License was issued January 31, 1986, and the Unit became commercially operational April 23, 1986.

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 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.

1.2 GENERAL PLANT DESCRIPTION

This section includes a summary description of the principal characteristics of the site and a concise description of Millstone 3.

1.2.1 GENERAL

Millstone 3 incorporates a four loop closed cycle pressurized water type nuclear steam supply system (NSSS); a turbine generator and electrical systems; engineered safety features; radioactive waste systems; fuel handling systems; structures and other on site facilities; instrumentation and control systems; and the necessary auxiliaries required for a complete and operable nuclear power station. The site plan (Figure 2.1.4) and the plot plan (Figure 1.2–2) show the general arrangement of the unit.

Piping and instrumentation diagrams are included throughout this document with the appropriate system descriptions. Symbols and abbreviations used in the diagrams are illustrated on Figure 1.2–3.

With respect to the numbers, graphs, and drawings included within this report, the normal tolerance permitted by good engineering practice is intended. Where operating parameters are unusually important, such items have been included in the technical specifications.

1.2.2 SITE

The site, approximately 500 acres in area, is on the north shore of Long Island Sound and on the east side of Niantic Bay. It is located in the Town of Waterford, Connecticut, about 3.2 miles west-southwest of New London and about 40 miles southeast of Hartford. The surrounding area is primarily residential with some commercial and industrial uses.

Millstone 1 and 2 are also located on the site. Millstone 1 is a permanently defueled boiling water reactor. Millstone 2 uses a two-loop pressurized water reactor supplied by Combustion Engineering, Inc., with a rated thermal power level of 2,700 MW; the architect-engineer was Bechtel Corporation. Section 2.1 contains a more detailed description of the site and surrounding areas.

1.2.3 STRUCTURES

Millstone 3 major structures are the containment structure, auxiliary building, fuel building (including decontamination facilities), waste disposal building, engineered safety features building, main steam valve building, turbine building, service building, control building, technical support center, emergency generator enclosure, containment enclosure building, warehouse 5 (including the condensate polishing waste treatment facility), auxiliary boiler enclosure, and circulating and service water pumphouse. Section 3.8.4.1 describes the general arrangement of these structures.

The reactor is operated inside a reinforced concrete containment structure maintained at a subatmospheric pressure between 10.6 and 14.0 psia. The containment concept is similar to those of Surry Power Stations 1 and 2, North Anna Power Stations 1 and 2, and Beaver Valley Power Station 1. Following the loss-of-coolant accident (LOCA), described in Section 15.6.5, the containment remains above atmospheric pressure.

The containment structure is housed within the containment enclosure building, which along with structures adjacent to the containment, forms the boundary of the supplementary leak collection and release system (SLCRS). The SLCRS establishes a subatmospheric pressure in the containment enclosure building and contiguous structures. See Section 6.2.3 for a further description.

The seismic criteria used in the design of the structures and equipment for Millstone 3 are described in Section 3.7.

1.2.4 NUCLEAR STEAM SUPPLY SYSTEM

The nuclear steam supply system (NSSS) consists of a Westinghouse pressurized water-type reactor and four closed reactor coolant loops connected in parallel to the reactor vessel. Each loop contains a reactor coolant pump and a steam generator, two loop isolation valves, an isolation bypass valve, and a bypass line. The NSSS also contains an electrically heated pressurizer and auxiliary systems.

High pressure water circulates through the reactor core to remove heat generated by the nuclear chain reaction. The heated water exits from the reactor vessel and passes via the coolant loop piping to the steam generators. Here, it releases heat to the feedwater to generate steam for the turbine generator. The cycle is completed when the water is pumped back to the reactor vessel. The entire coolant system is composed of leaktight components to ensure that all fluids are confined to the system.

The reactor core is of the multi-region type. All fuel reactor assemblies are mechanically identical, although the fuel enrichment is not the same in all assemblies. These assemblies incorporate the rod cluster control concept in canless 17 x 17 fuel rod assemblies using a spring clip grid to provide support for the fuel rods. The reactor moderator has a negative temperature coefficient of reactivity at full power at all times throughout core life.

In the typical initial core loading, three fuel enrichments are used. Fuel assemblies with the highest enrichments are placed in the reactor core outer region, and the two groups of lower enrichment fuel assemblies are arranged in a selected pattern in the central region. In subsequent refuelings, one-third of the fuel is discharged from the central region and fresh fuel is loaded into the outer region of the reactor core. The remaining fuel is arranged in the central two-thirds of the reactor core in such a manner as to achieve optimum power distribution.

Rod cluster control assemblies are used for reactor control and consist of clusters of cylindrical absorber rods. The absorber rods move within guide tubes in certain fuel assemblies. Above the reactor core, each cluster of absorber rods is attached to a spider connector and drive shaft, which

is raised and lowered by a drive mechanism mounted on the reactor vessel head. Downward movement of the rod cluster control after trip is by gravity.

The reactor coolant pumps are vertical, single-stage, centrifugal pumps of the shaft-seal type.

The steam generators are Westinghouse Model F vertical U-tube units which contain Inconel tubes. The Model F steam generator includes features such as improved tube support plate design and high circulation ratio which are designed to minimize most forms of corrosion, sludge buildup, and chemical attack. Integral moisture separation equipment reduces the moisture of steam to one-quarter percent or less.

All of the pressure containing and heat transfer surfaces in contact with reactor water are stainless steel clad or stainless steel except the steam generator tubes and fuel tubes, which are Inconel and Zircaloy, respectively. Reactor core internals, including control rod drive shafts, are primarily stainless steel.

There are two double-disc, motor-operated, loop isolation valves in each loop, one located between the reactor vessel and the steam generator and the other between the reactor vessel and the reactor coolant pump of each of the four loops. The isolation bypass valves, also double-disc, motor-operated valves, are located in a bypass line connecting the two loop isolation valves in each loop.

An electrically heated pressurizer connected to one reactor coolant loop maintains reactor coolant system pressure during normal operation, limits pressure variations during plant load transients, and keeps system pressure within design limits during abnormal conditions. In addition, it provides indication of and maintains reactor coolant system water inventory.

The auxiliary systems, provided as part of the NSSS, charge the reactor coolant system and add makeup water, purify reactor coolant water, provide chemicals for corrosion inhibition and reactivity control, remove decay heat when the reactor is shut down, and provide for emergency safety injection.

Other auxiliary systems supporting the NSSS but not part of the NSSS provide the following:

1. System components cooling
2. Fuel pool cooling
3. Reactor coolant water and other auxiliary system fluid sampling
4. Venting and draining the reactor coolant system and other auxiliary systems
5. Emergency containment depressurization spray and combustible gas control
6. Maintaining the containment atmosphere pressure at sub-atmospheric levels

7. Collecting, processing, and disposing of liquid and gaseous wastes

8. Preparation of solid wastes for disposal

9. Process, store, and supply reactor coolant system boric acid

10. Component ventilation

11. Instrument and valve operator air

1.2.5 INSTRUMENTATION AND CONTROL SYSTEMS

The instrumentation and control for the reactor protection system, engineered safety features actuation system, and other safety related systems meet the requirements of IEEE 279-1971, "Criteria for Protection System for Nuclear Power Generating System." In addition, other applicable criteria are met as described in Sections 3.1 and 7.1.2.

The nonsafety related instrumentation and controls accomplish reliable control and allow monitoring of the plant status without degradation of safety related instrumentation. Section 7.7 describes the design details.

The reactor is controlled by a coordinated combination of chemical shim, mechanical control rods, and temperature coefficients of reactivity. The control system allows the unit to accept step load changes of 10 percent and ramp load changes of 5 percent per minute over the load range of 15 percent to 100 percent power under nominal operating conditions subject to xenon limitations.

Control of the reactor and the turbine generator is accomplished from the control room, which contains all instrumentation and control equipment required for startup, operation, and shutdown, including normal and accident conditions. The turbine generator controls are designed for manual operation; the operator selects the load setpoint and loading rate. The NSSS automatically follows the turbine generator, on decreasing power, from loads of 100 to 15 percent power. The operator takes manual action to match the NSSS to the turbine generator load, for load increases between 15% and 100% power. If, during rapid turbine generator loading (5 percent per minute), the response of the control rods and chemical shim is not adequate to supply the needed reactivity, the reactor coolant temperature automatically drops to supply more reactivity. If a reactor coolant low operating temperature is reached, turbine generator loading is stopped automatically.

1.2.6 RADIOACTIVE WASTE SYSTEMS

Radioactive wastes are collected, processed, and disposed of in a safe manner complying with appropriate regulations, in particular, NRC Regulations 10 CFR 20, 10 CFR 50 Appendix I, 10 CFR 61, 10 CFR 71, 49 CFR 171-178, 10 CFR 100, and General Design Criteria 60 and 64 (Sections 3.1.2.60 and 3.1.2.64). There are three interrelated radioactive waste treatment systems: radioactive liquid waste, radioactive gaseous waste, and radioactive solid waste. Chapter 11 describes these systems.

The radioactive liquid waste system (LWS) collects, classifies, and processes all radioactive waste liquids generated during plant operation and refueling, either for recycle within the plant or for discharge off site. The process operations available to treat the liquid wastes are filtration and demineralization. The process descriptions and flow diagrams illustrate the number and sequence of processing steps to be applied to each type of liquid waste (Section 11.2).

Gaseous wastes, consisting of hydrogen streams and air streams containing various levels of radioactivity, are treated before release to the environment. Through the use of degasification and purification of reactor coolant letdown, the consequences of any reactor coolant leakage are minimized. This degasification and purification process produces hydrogen waste gas streams, which are passed through charcoal decay beds to provide adequate holdup time for the decay of noble gases and the removal of iodines. The decay beds are followed by high efficiency particulate air filters to ensure particulate removal. Aerated waste gas streams, produced by other phases of unit operation, are released to the environment via the Millstone stack. A process flow diagram, illustrating the processing steps for gaseous waste, appears in Section 11.3.

The radioactive solid waste system provides holdup, packaging, and storage facilities for the eventual off site shipment and ultimate disposal of solid radioactive waste material. Available process operations are: solidification of liquid wastes, holdup for decay, sluicing and dewatering of resins, encapsulation of miscellaneous solid wastes, and compacting of compressibles. Provisions for shielding during the processing and shipment of solid wastes are included in the design of the radioactive solid waste system. A process flow diagram illustrates the processing and handling sequences for solid wastes generated by the plant (Section 11.4).

1.2.7 FUEL HANDLING

The reactor is refueled by equipment designed to handle spent fuel under water from the time the spent fuel leaves the reactor vessel until the spent fuel is placed in a cask for shipment from the site. Underwater transfer of spent fuel provides an optically transparent radiation shield as well as a reliable source of coolant for the removal of decay heat produced by the spent fuel. New fuel is transferred to the fuel pool using the new fuel handling crane and is loaded into the reactor using the same equipment that handles the spent fuel. (Section 9.1.4)

1.2.8 TURBINE GENERATOR AND AUXILIARIES

The turbine is a tandem-compound, six-flow, 1,800 rpm unit with 43 inch last stage blades. Two combination moisture separator-reheaters remove moisture and superheat the steam between the high and low pressure turbines. The turbine generator is discussed, in detail, in Section 10.2.

The turbine generator plant and associated steam and power conversion systems are capable of a 40 percent load rejection without producing a reactor trip. Refer to Section 10.4.4.1. This is accomplished by dumping steam into the condenser through the turbine bypass system which reduces the transient to within the NSSS transient response capability.

The turbine control system is an electrohydraulic control (EHC) system capable of remote manual or automatic control of acceleration and loading of the unit at preset rates, and holding speed and

load at a preset level. The EHC provides a normal overspeed protection system and an emergency overspeed protection system to limit turbine overspeed.

A single-pass deaerating surface condenser installed in three sections, two 100 percent design capacity steam jet air ejector units, three 50 percent design capacity condensate pumps, three steam generator feedwater pumps (two turbine-driven and one motor-driven), two 50 percent design capacity motor-driven and one 100 percent design capacity turbine-driven steam generator auxiliary feedwater pumps, three trains of feedwater heaters, each having six stages, and a full flow condensate demineralizer polishing system are provided. Any combination of two steam generator feedwater pumps is adequate to support 100% power operation. Condenser circulating water is provided by six circulating water pumps. The turbine plant component cooling system provides cooling water for lubricating oil coolers, generator hydrogen coolers, and other turbine plant auxiliary heat loads.

1.2.9 ELECTRICAL SYSTEMS

The electric power system includes the electrical equipment and connections required to generate and deliver electric power to the 345 kV system. This system also includes station service electrical equipment to provide electric power to support station auxiliaries during normal operation, startup, shutdown, and accident conditions.

The major component in the system is the turbine-driven main generator. The electric power output from this generator is stepped up in a transformer bank and delivered to the 345 kV switchyard for distribution to the utility grid.

The station service equipment consists of switchgear, load centers, motor control centers, ac vital and nonvital buses, and battery systems. The normal source of station service power is provided by the main generator through normal station service transformers. Startup and shutdown station service power is provided by a preferred off site source from the 345 kV switchyard through the main and normal station service transformers with the generator breaker open, or by the alternative off site source from the 345 kV switchyard through the reserve station service transformers. In the event of an accident with loss of both normal and off site sources, an on site emergency power system, consisting of two redundant diesel engine-driven generators, provides power to the emergency 4,160V buses within 11 seconds after receiving a start signal. These diesel engine-driven generators are sized to provide required power to all safety related equipment.

1.2.10 ENGINEERED SAFETY FEATURES

Engineered safety features (ESF) are provided to mitigate the consequences of postulated accidents including a loss-of-coolant accident (LOCA) resulting from large and small pipe breaks. The ESF systems provided for Millstone 3 have sufficient redundancy and independence of components and power sources so that, under the conditions of the postulated accident, the

systems maintain the integrity of the containment structure and accomplish the following even when operating with a postulated single active failure:

1. Prevent radiation release to the outside environment from exceeding the limit specified in 10 CFR 50.67.

2. Provide core cooling to prevent excessive metal-water reaction, to limit the core thermal transient, and to maintain the core in a coolable geometry.

Millstone 3 is independent of Millstone 1 and 2 with respect to ESF. The systems provided for Millstone 3 are summarized below.

Containment Structure

The steel lined reinforced concrete containment structure provides a barrier against the escape of fission products. It is designed to operate at approximately atmospheric pressure and can withstand the pressures and temperatures resulting from a spectrum of LOCAs and secondary system breaks. The containments response following the accident is similar to other atmospheric PWRs. The structure and all penetrations, including access openings, are of proven design (Section 6.2.1).

Emergency Core Cooling System

The emergency core cooling system (ECCS) provides borated water to cool the reactor core following a major LOCA. This is accomplished by the automatic injection of water from the safety injection accumulators into the reactor coolant loops and by the automatic pumping of a portion of the refueling water storage tank contents into the loops via the charging pumps, the safety injection pumps, and the residual heat removal pumps. After the injection mode of emergency core cooling, long term core cooling is maintained by recirculating the water from the containment structure sump by the containment recirculation pumps, through the containment recirculation coolers, and into the reactor coolant loops directly and via the charging and safety injection pumps (Section 6.3).

Containment Heat Removal System

The containment heat removal system consists of the quench spray and the containment recirculation systems. Following the postulated DBA, the containment pressure is reduced by employing both systems.

The quench spray system sprays borated water from the refueling water storage tank (RWST).

The recirculation spray system draws suction from the containment sump, the content of which consists of the primary or secondary system effluent and the quench spray.

The sump water pH is controlled to be above 7.0 to improve effectiveness of fission products removal (Section 6.5.2), and for materials compatibility (Section 6.1.1).

The pH is controlled by the dissolution of trisodium phosphate crystals (stored in baskets) in the containment sump water.

Supplementary Leak Collection and Release System

Following a postulated accident, particulate and gaseous radioactive material is ducted from the containment enclosure structure and the buildings contiguous to the containment structure to the supplementary leak collection and release system (SLCRS), where it is filtered and discharged to the atmosphere through an elevated stack rather than through a ground-level vent. SLCRS is not credited for a postulated fuel handling accident.

Containment Isolation System

The containment isolation system isolates piping lines which penetrate the containment boundary so that, in the event of a LOCA, radioactivity is not released to the environment through these lines.

The lines are either isolated passively (check valves or locked closed manual valves) or isolated automatically by receipt of a safety injection signal (SIS), a containment isolation (phase A and B) signal, or a steamline isolation (SLI) signal (Section 6.2.4).

Engineered Safety Features Actuation System

The engineered safety features actuation system (ESFAS) monitors selected parameters and determines whether predetermined safety limits have been exceeded. If they have been exceeded, the ESFAS initiates action to mitigate the abnormal occurrence (Chapter 7).

Habitability Systems

The Millstone 3 control room envelope is equipped with an intake isolation system designed to protect the plant operators from the presence of hazardous substances outside the control room envelope.

Control room envelope makeup air is supplied via redundant air filtration trains designed to meet the requirements of Regulatory Guide 1.52 (Section 6.4).

Inservice Inspection

All ASME Section III, Code Class 1, 2, and 3 systems and components which require inservice inspection and testing are designed, fabricated, and erected to meet the inspection requirements of ASME Section XI (except where specific written relief has been granted by the Commission pursuant to 10 CFR 50.55a). The inservice inspection program includes baseline preservice examination and periodic inservice inspection and testing to ensure the operability and integrity of all systems classified Class 1, 2, and 3 pursuant to 10 CFR 50.55a (Sections 5.2.4 and 6.6).

1.2.11 COOLING WATER AND OTHER AUXILIARY SYSTEMS

Cooling water and other auxiliary systems provided in Millstone 3 are outlined below. Their design criteria and details are described in Chapters 9, 10, and 11.

1. The chemical and volume control system performs the following functions:

    a. Fills the reactor coolant system.

    b. Provides a source of high pressure water for pressurizing the reactor coolant system when cold.

    c. Maintains the water level in the pressurizer.

    d. Reduces the concentration of corrosion and fission products in the reactor coolant.

    e. Adjusts the boric acid concentration of the reactor coolant for chemical shim control.

    f. Provides high pressure seal water for the reactor coolant pump seals.

2. The boron recovery system concentrates and stores borated radioactive water from reactor coolant letdown (chemical and volume control system) processed through the gaseous waste system. Processing by an evaporator, ion exchanger, filters, and demineralizers in the boron recovery system is capable of producing primary-grade water and concentrated boric acid solution for station reuse or disposal.

3. Radioactive fluid degasification, liquid concentration, and waste solidification for disposal are provided by the radioactive gaseous waste, radioactive liquid waste, and radioactive solid waste systems.

4. The service water system provides cooling water for heat removal from the reactor plant auxiliary systems during all modes of operation and from the turbine plant auxiliary systems during normal operation.

5. The reactor plant component cooling water system, an intermediate cooling system, transfers heat from systems containing reactor coolant or other radioactive or potentially radioactive liquids to the service water system, and provides a source of safety grade cooling water to systems which have this requirement.

6. The turbine plant component cooling system, also an intermediate cooling system, transfers heat from various turbine plant equipment to the service water system.

7. The fuel pool cooling and purification system removes residual heat from spent fuel stored in the spent fuel pool and purifies the water in the refueling cavity and spent fuel pool.

8. The reactor plant vent and drain systems collect potentially radioactive fluids and gases from various plant systems and transfer these fluids and gases to the boron recovery system or to the appropriate waste disposal system.

9. Individual ventilation systems are provided for the containment and other structures. The containment ventilation system recirculates and cools containment air. The ventilation and air-conditioning system servicing the main control room provides uninterrupted service, even under accident conditions.

10. The circulating water system removes heat resulting from the operation of the turbine generator and main condensers. The service water system provides cooling water from the ultimate heat sink for systems and components which require an ensured supply of cooling water under all conditions.

11. The domestic water system receives water from the Town of Waterford, Conn., public water system and distributes it throughout the unit for power plant system makeup and potable water needs.

12. The compressed air systems consist of the service air system, instrument air system, and containment air system. Dryers are provided in the instrument air system and the containment instrument air system.

13. The fire protection system furnishes the capacity to extinguish any probable fires which might occur at Millstone 3. The system includes a water system, a CO_2 system, a halon system, and portable fire extinguishers.

14. The reactor and turbine plant sampling system have the capability for sampling all normal process systems and principal components listed in Tables 9.3–1 and 9.3–2 for laboratory analysis.

15. The Auxiliary Steam System is designed to supply steam for building heating and freeze protection for outdoor water storage tanks and to carry condensate from various heating and processing equipment associated with both Unit 2 and Unit 3 during normal plant operations. The system is nonnuclear safety (NNS).

1.2.12 REFERENCE FOR SECTION 1.2

FIGURE 1.2–1 NOT USED

Withhold under 10 CFR 2.390 (d) (1)

FIGURE 1.2–2 PLOT PLAN

FIGURE 1.2–3 (SHEETS 1-3) PIPING & INSTRUMENTATION DIAGRAM LEGEND

The figure indicated above represents an engineering controlled drawing that is Incorporated by Reference in the MPS-3 FSAR. Refer to the List of Effective Figures for the related drawing number and the controlled plant drawing for the latest revision.

1.3 COMPARISON TABLES

1.3.1 COMPARISON WITH SIMILAR FACILITY DESIGNS

Principal features of the design of Millstone 3 at the time of application for an operating license were similar to those that were evaluated and approved by the NRC staff for other reactors under construction, operation, or review. Comparison of notable similarities and differences to North Anna 1 and 2, Surry 1 and 2, Comanche Peak 1 and 2, W. B. McGuire 1 and 2, Maine Yankee, and Trojan was provided in a series of comparison tables in this section. This section is retained for historical purposes.

The design of this facility was based on proven technology attained during the development, design, construction, and operation of pressurized water reactors of similar or identical types. The data, performance characteristics, and other information represented a standard design that was engineered for the particular requirements of the utility system and site characteristics.

1.3.1.1 Comparison of Nuclear Steam Supply System

A design comparison of major parameters or features of Millstone 3 with similar plants was presented in Table 1.3-1. The following plants were used in the comparison:

1. Comanche Peak Units 1 and 2, Docket Number 55-445, -446

2. W.B. McGuire Units 1 and 2, Docket Number 50-369, -370

3. Trojan, Docket Number 50-344

1.3.1.2 Comparison of Engineered Safety Features

Table 1.3-2 compared the design data for the Millstone 3 engineered safety features (ESF) systems with similar systems in Trojan (for NSSS scope of supply - ECCS), and North Anna Power Station Units 1 and 2 (for balance of plant). The ESF systems compared were the emergency core cooling, containment depressurization, hydrogen recombiner, and supplementary leak collection and release systems.

These units were chosen for comparison because Millstone 3 systems were similar in design. Trojan and North Anna 1 and 2 obtained operating licenses and are currently in commercial operation.

1.3.1.3 Comparison of Containment Concepts

Table 1.3-3 summarized the design and operating parameters described in Section 6.2.1 for the Millstone 3 containment concept. The table provided a comparison with similar data for subatmospheric containments from the FSAR for North Anna 1 and 2, and the FSAR for Surry 1 and 2. These references were selected because the units used the Stone & Webster subatmospheric containment design. Surry 1 and 2 and North Anna 1 and 2 are in commercial operation.

1.3.1.4 Comparison of Instrumentation Systems

Tables 1.3-4 through 1.3-10 compared Millstone 3 instrumentation with similar instrumentation in North Anna 1 and 2. These units were chosen for the comparison because they have instrumentation similar in design to that of Millstone 3, they have obtained operating licenses, and are currently in commercial operation.

1.3.1.5 Comparison of Electrical Systems

Table 1.3-11 compared the Millstone 3 electrical system parameters with similar systems of Millstone 2 and Maine Yankee Atomic Power Station. The electrical systems compared were the 345 kV transmission, ac power, ac vital bus, 125 volt DC, and emergency power systems. These units were chosen for the comparison because they have electrical systems similar in design to Millstone 3. They have obtained operating licenses, and they are currently in commercial operation.

1.3.1.6 Comparison of Radioactive Waste Systems

Tables 1.3-12, 1.3-13, and 1.3-14 compared the radioactive waste systems features for Millstone 3 with similar systems in North Anna 1 and 2 and Surry 1 and 2. These units were chosen for the comparison, because they had radioactive waste systems similar in design to Millstone 3, have obtained operating licenses, and are currently in commercial operation.

1.3.1.7 Comparison of Other Reactor Plant Systems

Table 1.3-15 summarized the final design and operating parameters for the major reactor plant systems. This table compared the Millstone 3 data with systems data in similar nuclear power plants (North Anna 1 and 2).

1.3.2 COMPARISON OF FINAL AND PRELIMINARY DESIGNS

Table 1.3-16 detailed the significant design changes that were made since the submittal of the PSAR, through the issuance of the operating license.

TABLE 1.3-1 DESIGN COMPARISON

Parameter or Feature	Millstone 3 FSAR Chapter/ Section	Millstone 3	Comanche Peak	W. B. McGuire	Trojan
Reactor Core Heat Output (MWt)	4, 5, 15.0	3,411	3,411	3,411	3,411
Minimum DNBR for Design Transients	4.1, 4.4, 15.0	> 1.30	> 1.30	> 1.30	> 1.30
Total Thermal Flow Rate (10 lb/hr)	4.1, 4.4, 5.1	140.8	140.3	140.3	132.7
Reactor Coolant Temperatures (°F)	4.1, 4.4				
Core Outlet		620.6	620.8	620.8	619.5
Vessel Outlet		617.2	618.2	618.2	616.8
Core Average		590.5	589.4	589.4	585.9
Vessel Average		587.1	588.2	588.2	584.7
Core Inlet		557.0	558.1	558.1	552.5
Vessel Inlet		557.0	558.1	558.1	552.5
Average Linear Power (kW/ft)	4.1, 4.4	5.44	5.44	5.44	5.44
Peak Linear Power for Normal Operation (kW/ft)	4.1, 4.4	12.6	12.6	12.6	13.6
Heat Flux Hot Channel Factor, F_Q	4.1, 4.4, 15.0	2.32	2.32	2.32	2.50
Fuel Assembly Array	4.1, 4.3	17 x 17	17 x 17	17 x 17	17 x 17
Number of Fuel Assemblies	4.1, 4.3	193	193	193	193
Uranium Dioxide Rods Per Assembly	4.1, 4.3	264	264	264	264
Fuel Weight as Uranium Dioxide (lb)	4.1, 4.3	222,739	222,739	222,739	222,739

TABLE 1.3-1 DESIGN COMPARISON (CONTINUED)

Parameter or Feature	Millstone 3 FSAR Chapter/ Section	Millstone 3	Comanche Peak		W. B. McGuire	Trojan
Number of Grids Per Assembly	4.1, 4.3	8-Type R	8-Type R		8-Type R	8-Type R
Rod Cluster Control Assemblies	4.1, 4.3					
Number of Full/Part Length		61/-	53/-		53/8	53/8
Absorber Material		Hf	Hf		Ag-In-Cd/B_4C *	Ag-In-Cd
Clad Material		SS **	SS		SS	SS
Clad Thickness (inches)		0.0185	0.0185		0.0185	0.0185
Equivalent Core Diameter (inches	4.1, 4.3	132.7	132.7		132.7	132.7
Active Fuel Length (inches)	4.1, 4.3	144	143.7		143.7	143.7
Fuel Enrichment (Weight Percent)	4.1, 4.3		Unit 1	Unit 2		
Region 1		2.40	1.60	1.40	2.10	2.10
Region 2		2.90	2.40	2.10	2.60	2.60
Region 3		3.40	3.10	2.90	3.10	3.10
Number of Coolant Loops	5	4	4		4	4
Total Steam Flow (10 lb/hr)	5.1	15.05	15.14		15.14	15.07
Reactor Vessel	5.3					
Inside Diameter (inches)		173	173		173	173
Inlet Nozzle Inside Diameter (inches)		27.5	27.5		27-1/2	27.5
Outlet Nozzle Inside Diameter (inches)		29	29		29	29

TABLE 1.3-1 DESIGN COMPARISON (CONTINUED)

Parameter or Feature	Millstone 3 FSAR Chapter/ Section	Millstone 3	Comanche Peak	W. B. McGuire	Trojan
Number of Reactor Closure Head Studs		54	54	54	54
Reactor Coolant Pumps	5.4.1				
Horsepower		7,000	7,000	7,000	6,000
Capacity (gpm)		100,400	99,000	99,000	88,500
Head (feet)		289	288	288	277
Steam Generators	5.4.2				
Model		F	D	D	51
Heat Transfer Area (ft^2)		55,000	48,300	48,000	51,500
Number of U-Tubes		5,626	4,578	4,674	3,388
Residual Heat Removal	5.4.7				
Initiation Pressure (psig)		425	425	425	400
Initiation/Completion Temperature (°F)		350/120	350/140	350/140	350/ 140
Component Cooling Water Design Temperature (°F)		95	105	95	95
Cooldown Time After Initiation (hr)		20	16	16	16
Heat Exchanger Removal Capacity (10 Btu/hr)		35.27	39.1	34.15	34.2
Pressurizer	5.4.10				

TABLE 1.3-1 DESIGN COMPARISON (CONTINUED)

Parameter or Feature	Millstone 3 FSAR Chapter/ Section	Millstone 3	Comanche Peak	W. B. McGuire	Trojan
Heatup Rate Using Heaters (°F/hr)		55	55	55	55
Internal Volume (ft^3)		1,800	1,800	1,800	1,800
Pressurizer Safety Valves	5.4.13				
Number		3	3	3	3
Maximum Relieving Capacity (lb/hr)		420,000	420,000	420,000	420,000
Accumulators	6.3				
Number		4	4	4	4
Operating Pressure, Minimum (psig)		600	600	600	600
Minimum Operating Water Volume Each (ft^3)		950	950	950	870
Centrifugal Charging Pumps	6.3				
Number		3	2	2	2
Design Flow (gpm)		150	150	150	150
Design Head (feet)		5,800	5,800	5,800	5,800
Safety Injection Pumps	6.3				
Number		2	2	2	2
Design Flow (gpm)		425	425	425	425
Design Head (feet)		2,680	2,680	2,500	2,500

TABLE 1.3-1 DESIGN COMPARISON (CONTINUED)

Parameter or Feature	Millstone 3 FSAR Chapter/ Section	Millstone 3	Comanche Peak	W. B. McGuire	Trojan
Residual Heat Removal Pump	5.4.7, 6.3				
Number		2	2	2	2
Design Flow (gpm)		4,000	3,800	3,000	3,000
Design Head (feet)		350	350	375	375
Instrumentation and Controls	7	***	***	***	***
Chemical and Volume Control	9.3.4				
Total Seal Water Supply Flow Rate, Nominal (gpm)		32	32	32	32
Total Seal Water Return Flow Rate, Nominal (gpm)		12	12	12	12
Letdown Flow, Normal/Maximum (gpm)		75/120	75/120	75/120	75/120
Charging Flow, Normal/Maximum (gpm)		55/100	55/100	55/100	55/100

NOTE:

* The Ag-In-Cd on Unit 1, B_4C on Unit 2

** SS = Stainless steel

*** The instrumentation and control systems discussed in Chapter 7 for Millstone 3 are functionally similar to those systems implemented in Comanche Peak, W. B. McGuire, and Trojan.

TABLE 1.3-2 COMPARISON OF ENGINEERED SAFETY FEATURES

Emergency Core Cooling System (Section 6.3)	Millstone 3	Trojan
Charging Pump (used for high pressure safety injection)		
Number	3	2
Design capacity each (gpm)	150	150
Design total developed head (feet)	5,800	5,800
Safety Injection Pump (used for intermediate pressure safety injection)		
Number	2	2
Design capacity each (gpm)	425	425
Design total developed head (feet)	2,500	2,500
Residual Heat Removal pump (used for low pressure safety injection)		
Number	2	2
Design capacity each (gpm)	4,000	3,000
Design total developed head (feet)	350	375
Safety Injection Accumulator		
Number	4	4
Total volume each (ft^3)	1,350	1,350
Water volume (ft^3, minimum)	950	870
Operating pressure (psig, minimum)	600	600

Containment Depressurization Systems (Section 6.2.2)	Millstone 3	North Anna 1 and 2
Quench Spray Pump		
Number	2	2
Design capacity each (gpm)	4,000	2,000
Design total developed head (feet)	291	265
Containment Recirculation Pump		

TABLE 1.3-2 COMPARISON OF ENGINEERED SAFETY FEATURES (CONTINUED)

Number	4 outside containment	2 out / 2 in [a]
Design capacity each (gpm)	3,950	3,700/3,300
Design total developed head (feet)	342	287/269
	Millstone 3	North Anna 1 and 2
Containment Recirculation Cooler		
Number	4	4
UA per cooler (Btu/hr-°F)	3.865×10^6	3.79×10^6
Recirculation flow (gpm)	3,950	3,500
Service water flow (gpm)	6,500	4,500
Refueling Water Storage Tank		
Volume (gal)	1,206,556	480,000
Temperature (°F, maximum)	75	50
Hydrogen Recombiner System (Section 6.2.5)		
Number	2	2
Flow rate (scfm, each)	50	50
Supplementary Leak Collection and Release System (Section 6.2.3)		
Number of filter trains	2	None
Flow rate (scfm, each)	9,700	None

a. Out - outside containment
In- inside containment

TABLE 1.3-3 COMPARISON OF CONTAINMENT CONCEPTS
(Section 6.2.1)

	Millstone 3	**North Anna 1 and 2**	**Surry 1 and 2**
Type	Subatmospheric	Subatmospheric	Subatmospheric
ID (feet)	140	126	126
Overall height (feet)	200	191	173
Free volume (ft^3)	2.3×10^6	1.825×10^6	1.73×10^6
Maximum design pressure (psig)	45	45	45
Design temperature (°F)	280	280	280
Calculated peak pressure psig)	38.49	44.1 [a]	44.98 [b]
Reactor Coolant System:			
Liquid volume (including pressurizer) (ft^3)	11,695	9,874	9,874
Temperature (mass average) (°F)	583.5	586.8	574.5
Concrete Thickness:			
Vertical wall	4 feet 6 inches	4 feet 6 inches	4 feet 6 inches
Dome	2 feet 6 inches	2 feet 6 inches	2feet 6 inches
Containment structure leak rate (%/day)	0.30 (0-24 hrs)	0.1	0.1
	0.15 (24-720 hrs)	0.0	0.0

a. Based on Tagami condensing heat transfer coefficient

b. Based on Uchida condensing heat transfer coefficient

TABLE 1.3-4 COMPARISON OF CONTAINMENT ATMOSPHERE PRESSURE SENSOR PARAMETERS[A]

Pressure Sensors	Millstone 3	North Anna 1 and 2
Containment Atmosphere High Pressure Transmitter (Millstone 3, High-1)		
Number of channels	3	3
Logic matrix	2/3	2/3
Approximate setpoint (psia)	19.7	15.0
Containment Atmosphere Intermediate High-High Pressure Transmitter (Millstone 3; High-2)		
Number of channels	3	3
Logic matrixÅ	2/3	2/3
Approximate setpoint (psia)	19.7	20
Containment Atmosphere High-High Pressure Transmitter (Millstone 3; High-3)		
Number of channels	4	4
Logic matrix	2/4	2/4
Approximate setpoint (psia)	24.7	25.0

a. The numbers stated for components or system performance do not represent the maximum/minimum acceptable or required values to support system operation. Setpoints are stated in the Technical Specifications.

TABLE 1.3-5 COMPARISON OF REACTOR COOLANT PUMP BUS PROTECTION [A]

	Millstone 3	North Anna 1 and 2
Undervoltage Reactor Coolant Pumps Buses		
Number of channels	N/A	3
Logic matrix	N/A	2/3
Approximate Setting (V)	N/A	70% of 4,160 (2,912)
Underfrequency Reactor Coolant Pumps Buses		
Number of channels	N/A	3
Logic matrix	N/A	2/3
Approximate Setting (Hz)	N/A	54-59
Reactor Coolant Pump Shaft Low-Low Speed		
Number of channels	4	N/A
Logic matrix	2/4	N/A
Approximate Setting (rpm)	(later)	N/A
Reactor Coolant Pump Shaft Low Speed		
Number of channels	N/A	N/A
Logic matrix	N/A	N/A
Approximate Setting (rpm)	N/A	N/A

a. The numbers stated for components or system performance do not represent the maximum/minimum acceptable or required valves to support system operation. Settings are stated in the Technical Specifications.

TABLE 1.3-6 COMPARISON OF ENGINEERED SAFETY FEATURE ACTUATION SIGNALS

Signal Actuation	Millstone 3	North Anna 1 and 2
Safety Injection Signal (SIS)		
Low pressurizer pressure coincident with low pressurizer level	No	No
Low pressurizer pressure	Yes	Yes
High main steam line differential pressure	No	Yes
Low steam line pressure	Yes	No
High main steam flow coincident with low main steam pressure or low temperature average	No	Yes
High containment atmosphere pressure	Yes	Yes
Manual initiation	Yes	Yes
Containment Isolation Phase A (CIA) Signal		
Safety injection signal (SIS)	Yes	Yes
Manual initiation	Yes	Yes
Containment Isolation Phase B (CIB) Signal or Containment Depressurization Actuation (CDA) Signal		
High-High containment atmosphere pressure (Millstone High-3)	Yes	Yes
Manual initiation	Yes	Yes
Steam Line Isolation Signal		
High main steam flow coincident with low steam pressure or low main reactor coolant temperature average	No	Yes
High-High containment pressure	No	No
Intermediate High-High containment atmosphere pressure (Millstone; High-2)	Yes	Yes
High steam pressure rate	Yes	No
Low steamline pressure	Yes	No
Manual initiation	Yes	Yes

TABLE 1.3-7 COMPARISON OF EMERGENCY GENERATOR AND STEAM GENERATOR AUXILIARY FEEDWATER PUMP START SIGNALS

Component	Millstone 3	North Anna 1 and 2
Emergency generator auto start signals	Emergency bus under-voltage	Emergency bus under-voltage
	Safety injection signal (SIS)	SIS
Steam generator auxiliary feedwater pump (motor-driven)		All steam generator feedwater pumps tripped
	2/4 - lo-lo level trip any steam generator (1/4 matrix) coincident with reactor coolant loop cold leg stop valve open	2/3 - lo-lo level trip any steam generator (2/3 matrix) coincident with reactor coolant loop hot leg stop valve open or reactor coolant loop cold leg stop valve open
	Sequenced safeguards signal	Undervoltage reserve station service power
	SIS	SIS
Steam generator auxiliary feedwater pump (turbine-driven) auto start signals		2/3 - undervoltage on station service bus
	2/4 - steam generators lo-lo level (2/4 matrix) coincident with reactor coolant loop cold leg stop valve open	2/3 - steam generator lo-lo level trip (2/3 matrix) coincident with reactor coolant loop hot leg stop valve open or reactor coolant loop cold leg stop valve open

NOTE:

The numbers stated for components or system performance do not represent the maximum/ minimum acceptable or required values to support system operation.

TABLE 1.3-8 COMPARISON OF PROCESS AND EFFLUENT RADIATION MONITORING SYSTEMS

Monitor	Number of Locations	
	Millstone 3	North Anna 1 and 2
Aerated vent particulate	0	1
Aerated vent gas	0	1
Ventilation vent particulate	0	1
Ventilation vent gas	1	1
Ventilation vent high range (particulate and gas)	1	0
Hydrogenated vent	1	N/A
Supplementary leak collection	1	0
Condenser air ejector	1	1
Containment recirculation cooler service water outlet	2	4
Component cooling heat exchanger service water discharge	0	1
Liquid waste	1	1
Steam generator blowdown sample	1	3
Auxiliary condensate	1	0
Turbine building floor drains	1	0
Reactor plant component cooling water subsystem	1	1
Reactor Coolant Letdown: Gross activity	0	2
Reactor Coolant Letdown: Specific fission product activity	0	0
Circulating water discharge	0	1
Liquid waste evaporator (removed from service)	0	1
Service water discharge	0	1
Service water reservoir	0	1
Control building inlet	2	0
Regenerant evaporator (removed from service)	1	0
Waste neutralizing sump	1	0
Steam line monitors	5	0

TABLE 1.3-9 COMPARISON OF AREA RADIATION MONITORING SYSTEMS

Monitor	Number of Locations	
	Millstone 3	North Anna 1 and 2
Containment structure low range	4	1
Containment structure high range	4	1
Manipulator crane	1	1
Incore instrumentation transfer area	1	1
Decontamination area	1	1
New fuel storage area	1	1
Spent fuel pool pit bridge and hoist	1	1
Auxiliary building control area	0	1
Sample room	1	1
Control room	1	1
Laboratory (Service building)	1	1
Auxiliary building general area	8	0
Equipment decontamination area (Service Building)	1	0

TABLE 1.3-10 COMPARISON OF AIRBORNE RADIATION MONITORING SYSTEMS

Monitor	Number of Locations	
	Millstone 3	North Anna 1 and 2
Auxiliary building lower levels particulate	2	0
Auxiliary building lower levels gas	2	0
Auxiliary building upper levels particulate	3	0
Auxiliary building upper levels gas	3	0
Charging pumps cubicles particulate	1	0
Charging pumps cubicles gas	1	0
Fuel building particulate	1	0
Fuel building gas	1	0
ESF building particulate	1	0
ESF building gas	1	0
Waste building particulate	1	0
Waste building gas	1	0
Control room particulate	1	0
Control room gas	1	0
Containment structure particulate	1	1
Containment structure gas	1	1
Ventilation vent sample particulate [a]	1	1
Ventilation vent sample gas	1	1
Leak collection area gas	1	0
Leak collection area particulate	1	0

a. The ventilation vent sample particulate and gas monitors have the capability to take a sample from any one of eight different areas, some of which are equivalent to areas being monitored by separate Millstone 3 monitors.

TABLE 1.3-11 COMPARISON OF ELECTRICAL SYSTEM PARAMETERS

Systems and Components	Millstone 3	Millstone 2	Maine Yankee
TRANSMISSION SYSTEM (Section 8.2)			
Transmission Lines to Circuits	4 at 345 kV	3 at 345 kV	2 at 345 kV
	(Total for 3 Units)	(Total for 2 Units)	2 at 115 kV
AC POWER SYSTEMS (Section 8.3.1)			
Main transformer	2 at 630 MVA	1 at 945 MVA	2 at 430 MVA
Normal station service transformer	1 at 50 MVA (6.9 kV)	1 at 45 MVA	1 at 30 MVA
	1 at 40 MVA (4.16 kV)		1 at 20 MVA
Reserve station service transformer	1 at 50 MVA (6.9 kV)	1 at 45 MVA	1 at 30 MVA
	1 at 45 MVA (4.16 kV)		1 at 20 MVA
AC VITAL BUS SYSTEM (Section 8.3.1)			
Distribution Cabinets	6	4	4
Inverters	4 at 25 kVA	4 at 15 kVA	4 at 10 kVA
	1 at 30 kVA		
	1 at 60 kVA		
125-V DC SYSTEM (Section 8.3.2)			
Unit batteries (125V)	2 at 1650 Ah		
	2 at 750 Ah	2 at 2300 Ah	
	2 at 2550 Ah	1 at 1200 Ah	2 at 1800 Ah
	4 at 200 amp	6 at 400 amp	4 at 250 amp

TABLE 1.3-11 COMPARISON OF ELECTRICAL SYSTEM PARAMETERS (CONTINUED)

Systems and Components	Millstone 3	Millstone 2	Maine Yankee
	2 at 50 amp		
	3 at 200 amp (installed spares)		
EMERGENCY POWER SYSTEM (Section 8.3.1)			
Emergency generator (continuous rating)	2 at 4986 kW	2 at 2750 kW	2 at 2500 kW
Emergency 4.16 kV Buses	2 at 2000/3000 amp	2 at 2000 amp	2 at 1200 amp

TABLE 1.3-12 COMPARISON OF RADIOACTIVE LIQUID WASTE SYSTEMS

	Millstone 3	North Anna 1 & 2	Surry 1 and 2
Type of Processing			
Evaporation (evaporation removed from service at Millstone)	Yes	Yes	Yes
Demineralization	Yes	Yes	Yes
Filtration	Yes	Yes	Yes
Treatment of Radioactive Wastes			
High Activity Wastes	Evaporation and/or demineralization and filtration, if required	Same	Same
Low Activity Wastes Including Contaminated Shower Drains	Filtration (evaporation and subsequent operations are optional)	Same	Same
Regenerant Chemical Waste	Filtration if required	None	None
Steam Generator Blowdown	Blowdown piped to flash tank where steam is drawn off to 4th point heater and liquid is drawn to main condenser for treatment by the condensate demineralizers	Blowdown is cooled and sent to clarifier for flocculation, sedimentation, and filtration. The sediment is sent to the solid waste disposal system and the liquid discharged through the liquid waste monitor to the circulating discharge tunnel.	Blowdown is cooled and then either released or demineralized and filtered and recycled to the main condenser
Equipment			
High Level Waste Drain Tanks			
Number	2	2	2

TABLE 1.3-12 COMPARISON OF RADIOACTIVE LIQUID WASTE SYSTEMS (CONTINUED)

	Millstone 3	North Anna 1 & 2	Surry 1 and 2
Capacity (gal/each)	26,000 *	5,000	2,390
Low Level Waste Drain Tanks			
	2	2	2
Capacity (gal/ea)	4,000 *	5,000	2,874
Regenerant Evaporator Feed Tanks (removed from service)		N/A	N/A
Number	2		
Capacity (gal/each)	13,500		
Contaminated Shower Drain Tank			
Number	0	2	2
Capacity (gal/each)		1,400 (including Laundry)	1,230 (including Laundry)
Waste Test Tank			
Number	2	2	2
Capacity (gal/each)	24,000	1,500	548
Regenerant Chemical Evaporator (removed from service)		N/A	N/A
Number	1		
Trays, number	8		
Capacity (gal/each)	35		

TABLE 1.3-12 COMPARISON OF RADIOACTIVE LIQUID WASTE SYSTEMS (CONTINUED)

	Millstone 3	North Anna 1 & 2	Surry 1 and 2
Waste Evaporator (removed from service at Millstone)			
Number	1	1	1
Trays, number	8	0	0
Capacity (gal/each)	35	6	6
Demineralizers			
Number	Two waste evaporators, one regenerant chemical evaporator (removed from service)	One waste evaporator distillate polishing, two clarifier demineralizers	One waste evaporator distillate polishing
Type	Mixed bed	Mixed bed	Mixed bed
Capacity (ft^3)	35	17/45 respectively	17
Effluent Filters			
Number	2	2	1
Capacity (gpm/each)	50	50	75
Filter element type	Wound fiber	Wound cotton fiber	Wound synthetic fiber

NOTE:

* Include condensate demineralizer, liquid waste system tank volumes. The numbers stated for components or systems performance do not represent the maximum/minimum acceptable or required values to support system operation.

TABLE 1.3-13 COMPARISON OF RADIOACTIVE GASEOUS WASTE SYSTEMS

	Millstone 3	North Anna 1 and 2	Surry 1 and 2
Type of Treatment			
Degasification	Yes	Occurs in boron recovery system	Occurs in boron recovery system
Decay of noble gases in high activity gas stream	Yes	Yes	Yes
Filtration of low activity gas streams recombiners	Gas streams	Yes	Yes
	No		
Recombiners	Streams	Yes	Yes
	No		
Treatment of Streams			
Continuous degasification of reactor letdown	Yes	Yes	Yes
Degasification of letdown to boron recovery system	Yes	Yes	Yes
Decay method for gases stripped in degasifier	Adsorption on charcoal for minimum 60-day xenon decay before recycle or release to the environment	Recombination of hydrogen stream for reduction storage in gas decay tank, then charcoal filtration before release to atmosphere	Recombination of hydrogen stream for reduction storage in gas decay tanks before release to the environment
Low activity air streams (nonventilation streams)	Release through Millstone stack	Filtration (charcoal and HEPA filters) and release thru vent on top of Unit 1 containment	Filtration (charcoal and particulate filters) and release to environment

TABLE 1.3-13 COMPARISON OF RADIOACTIVE GASEOUS WASTE SYSTEMS (CONTINUED)

	Millstone 3	North Anna 1 and 2	Surry 1 and 2
Equipment			
Degasifier			
Number	1	2	1 (common to both units)
Capacity (gpm/ea)	150		240
Process Gas Compressor			
Number	2	2	2
Capacity (scfm/ea)	3	1.5	2.5
(Currently abandoned in place)			
Process Gas Charcoal Absorbers			
Number	2	0	0
Capacity - Charcoal (lb/ea)	13,500	-	-
Gas Decay Tanks			
Number	0	2	2
Capacity (ft^3/ea)	-	462	434
Pressure (psig)	-	115 - the inside tank 4 - the outside tank	115
Gas Surge Tank (Process Gas Receiver)			
Number	1	1	1
Capacity (ft^3)	10	15	15.7
Waste Gas Recombiners			

TABLE 1.3-13 COMPARISON OF RADIOACTIVE GASEOUS WASTE SYSTEMS (CONTINUED)

	Millstone 3	North Anna 1 and 2	Surry 1 and 2
Number	0	1	1
Capacity (scfm)		1.5	1.31
Pressure (psia)		14	22
Waste Gas Compressors			
Number	0	2	2
Capacity (scfm/ea)	0	1.5	1.5
Pressure (psig)		150	120
Filter for low activity aerated activity aerated gaseous waste effluents	None	Activated charcoal with HEPA after-filter	Activated charcoal with HEPA after-filter

TABLE 1.3-14 COMPARISON OF RADIOACTIVE SOLID WASTE SYSTEMS

	Millstone 3	North Anna 1 and 2	Surry 1 and 2
Type of Treatment			
Solidification/Dewatering	Yes	Yes	Yes
Drumming	Yes	Yes	Yes
Solidification agent	Cement	Urea formaldehyde	Cement
Inputs (Type) Treated			
Boron evaporator bottoms	Yes	Yes	Yes
Waste evaporator bottoms	Waste Evaporator removed from service at Millstone	Yes	Yes
Regenerant evaporator bottoms	Evaporator removed from service	N/A	N/A
Filters	Yes	Yes	Yes
Resins	Yes	Yes	Yes
Miscellaneous wastes (contaminated clothing, tools, paper products, etc.)	Yes	Yes	Yes
Equipment			
Spent resin hold tank:			
Number	1	1	1
Capacity (gal)	3,000	1,800	2,019
Spent resin dewatering tank:			
Number	1	1	1
Capacity (gal)	500	500	619

TABLE 1.3-14 COMPARISON OF RADIOACTIVE SOLID WASTE SYSTEMS (CONTINUED)

	Millstone 3	North Anna 1 and 2	Surry 1 and 2
Spent resin transfer pump filter:			
Number	1	1	1
Capacity (gpm)	150	100	150
Type	Wound fiber	Wound fiber	Wound fiber
Evaporator bottoms tank: (removed from service at Millstone)			
Number	1	1	N/A
Capacity (gal)	3,150	1,100	
Shipping container (with cask):			
Capacity	50 ft^3	50 ft^3	55 gallon drum
Type	All solid waste is 50 ft^3 or greater shipping containers	All solid waste is shipped in 50 ft^3 containers except compressible waste which is in 55 gallon drums	All waste in 55 gallon drums

TABLE 1.3-15 COMPARISON OF OTHER REACTOR PLANT SYSTEMS

Systems with Components	Operating Parameters	
	Millstone 3	**North Anna 1 and 2**
Fuel Pool Cooling and Purification System (Section 9.1.3)		
Fuel Pool Cooling Pumps:		
Number	2	2
Design capacity (gpm)	3,500	2,750
Design total head (ft)	92	80
Fuel Pool Coolers:		
Number	2	2
Duty per heat exchanger (Btu/hr)	27,700,000	56,800,000
Fuel pool cooling flow (gpm)	3,500	2,750
Component cooling flow (gpm)	1,800	3,350
Number of cores cooled	15-1/2 (3048 Fuel Assemblies)	1-1/3
Fuel pool temperature, normal (°F)	150	140
Reactor Plant Component Cooling System (Section 9.2.2.1)		
Reactor Plant Component Cooling Pumps:		
Number	3	4
Design capacity (gpm)	8,100	8,000
Design total head (ft)	284	190

TABLE 1.3-15 COMPARISON OF OTHER REACTOR PLANT SYSTEMS (CONTINUED)

Systems with Components	Operating Parameters	
	Millstone 3	**North Anna 1 and 2**
Reactor Plant Component Cooling Heat Exchangers:		
Number	3	2
Duty per unit exchanger (Btu/hr)	76,000,000	52,000,000
Reactor plant component cooling water flow (gpm)	8,100	9,000
Service water flow (gpm)	8,000	10,500
System Design Basis (safety related)	Reactor cooldown to 120°F in 24 hr	Reactor cooldown to 140°F in 16 hr
Duty per unit exchanger (Btu/hr)	76,000,000	52,000,000
Reactor plant component cooling water flow (gpm)	8,100	9,000
Service water flow (gpm)	8,000	0,500
System Design Basis (safety related)	Reactor cooldown to 120°F in 24 hr	Reactor cooldown to 140°F in 16 hr
Service Water System (Section 9.2.1)		
Service Water Pumps:		
Number	4	6
Design capacity (gpm)	15,000	11,500
Design total head (ft)	120	127
Design temperature, maximum (°F)	80	95
Boron Recovery System (Section 9.3.2)		
Type of Treatment:		

TABLE 1.3-15 COMPARISON OF OTHER REACTOR PLANT SYSTEMS (CONTINUED)

Systems with Components	Operating Parameters	
	Millstone 3	**North Anna 1 and 2**
Degasification of liquid entering liquid entering system	Occurs in radioactive gaseous waste system	Yes
Storage of liquid prior to processing	Yes	Yes
Evaporation	Yes, in boron evaporator	Yes, in boron evaporator
Demineralization of boron evaporator distillate	Yes, if required	Yes, if required
System Effluents:		
Boron evaporator distillate	Primarily recycled to primary grade water storage tank; remainder to radioactive liquid waste system for discharge	Primarily recycled to primary grade water storage tank; remainder to radioactive liquid waste system for discharge
Boron evaporator bottoms	Recycled to boric acid tanks or drummed in radioactive solid waste system	Recycled to boric acid tanks or drummed in radioactive solid waste system
Cesium Removal Ion Exchanger:		
Number	2	2
Resin Capacity (ft^3)	35	45
Boron recovery tanks:		
Number	2	3
Capacity (gal)	150,000	120,000
Boron Evaporator Subsystem:		

TABLE 1.3-15 COMPARISON OF OTHER REACTOR PLANT SYSTEMS (CONTINUED)

Systems with Components	Operating Parameters	
	Millstone 3	**North Anna 1 and 2**
Evaporator type	Forced circulation	Forced circulation
Number	1	2 (for 2 units)
Reboiler	External	External
Trays	8	5
Capacity (gpm)	25	20
Boron Test Tanks:		
Number	2	2
Capacity (gal)	12,000	20,000
Boron Demineralizers:		
Number	2	2
Resin capacity (ft^3)	35	45

TABLE 1.3-16 COMPARISON OF FINAL AND PRELIMINARY INFORMATION

FSAR Section		Significant Changes Since PSAR	PSAR Reference
2.5.3		Thirteen faults have been uncovered in the rock excavation mapping since November 1979.	2.5.3
3.1.1.3		Application of the single failure criterion has been upgraded to be consistent with regulatory requirements and industry standards and practices.	3.1.1.2
3.3.2		Postulated tornado missiles and their characteristics have been revised to conform to Regulatory Guide 1.76 dated April 1974.	3.3.2
3.5.1.4		Conformance to Regulatory Guide 1.117, Rev. 1, dated April 1978. Have revised the spectrum of tornado generated missiles used for plant design.	3.5.1.4
3.7B.4.2		The number of seismic instrumentation packages has been increased and those locations revised.	3.7.4
3.8.1		Structural ring added around the containment structure to maintain isolation from the surrounding rock.	3.8.2.1
3.8.4	1.	Revised plan and design of waste disposal building to conform to Regulatory Guide 1.143, Rev. 1, dated October 1979.	3.8.2
	2.	Description of railroad canopy adjacent to fuel building added.	3.8.1
3.11		Class IE electrical equipment has been qualified to IEEE 323-1974 and IEEE 344-1975 requirements.	3.11
5.2.4		Inservice methods have been changed from closed circuit television monitors to ultrasonic scanners in the reactor vessel.	5.2.8
5.2.5		Air cooler outlet temperature and reactor coolant system make up rates are no longer monitored.	5.2.7
5.4.2		Model D steam generators have been replaced by Model F steam generators.	5.4.2

TABLE 1.3-16 COMPARISON OF FINAL AND PRELIMINARY INFORMATION (CONTINUED)

FSAR Section		Significant Changes Since PSAR	PSAR Reference
5.4.7 6.3		Contents have been revised to include cold shutdown requirements of Regulatory Guide 1.139, Rev. 0, date May 1978.	5.5.7, 6.3
6.2.1.7		Temperature monitoring for containment sump is not Q.A. Category I as implied in PSAR Section 7.5.3.2.	7.5.3.2
6.2.6.2		Added description of electrical penetrations leakage rate tests.	6.2.6.4
6.3		Boron injection tank (BIT) has been eliminated.	6.3
6.4	1.	Pressurization system split into two banks of 100% capacity each.	6.4
	2.	Deletion of vinyl chloride detection.	
	3.	Air-conditioning units which serve areas outside the control room pressure envelope have been relocated from this control room area to outside the control room pressure envelope.	
6.4.6		Only one smoke detector is provided on the control building ventilation inlet. The response to PSAR Question 9.57 stated redundant detectors would be provided.	Response to PSAR Question 9.57
7.3.1, 1.3.1.4	1.	Change in pressurizer pressure required for actuation of safety injection signal (SIS).	7.3.4.3, 1.3.1.4
	2.	Change in steamline and containment pressure required for actuation of steamline isolation signal.	
7.4 3.1.2.19		Millstone 3 design now has the capability for a safety grade cold shutdown from the auxiliary shutdown panel.	7.4, 3.1.2.19
7.8		Addition of ATWS Mitigation System Actuation Circuitry, conformance to 10 CFR 50.62.	None
8.3.1	1.	Millstone 2 electrical power tie has been deleted.	8.3.1

TABLE 1.3-16 COMPARISON OF FINAL AND PRELIMINARY INFORMATION (CONTINUED)

FSAR Section		Significant Changes Since PSAR	PSAR Reference
	2.	Improvements in cable separation to comply with Regulatory Guide 1.75, Rev. 2, dated Sept. 1978.	
	3.	The capacity of the emergency generators has been increased from 4,300 kW to 4,986 kW.	
	4.	The emergency generator enclosure has been relocated closer to the control building.	
	5.	The second off site source of power has been changed from having a capacity for one emergency system to having a capacity for both emergency systems and all nonsafety systems. The unit now has two independent off site sources each with the capacity for all emergency and normal loads during any circumstances, and each source is now immediately (0.1 second or less) available.	
	6.	Physical and electrical separation of nonsafety from safety circuits within safety related 4,160V switchgear, 480V load centers and motor control centers is no longer required. Instead these nonsafety circuits are considered as safety circuits within this equipment and subsequently changed to nonsafety status after going through qualified isolation devices.	
8.3.2	1.	Addition of a second 125V battery and its associated equipment to the normal (nonsafety related) dc power system.	8.3.2
	2.	Compliance to Regulatory Guide 1.75, Rev. 2, dated Sept. 1978.	
9.1.2		Large increase in amount of spent fuel that is able to be stored.	9.1.2
9.2.1		Addition of motor control center (MCC) and rod drive area air-conditioning (a/c) units as loads to the service water system (SWP). This addition requires two booster pumps to meet flow requirements to the a/c units.	9.2.1
9.2.2.1	1.	The component cooling system no longer supplies cooling water to the reactor coolant pump (RCP) motor air cooler and the control rod drive mechanism (CRDM) shroud coolers. Loads transferred to chilled water system.	9.2.2.1

TABLE 1.3-16 COMPARISON OF FINAL AND PRELIMINARY INFORMATION (CONTINUED)

FSAR Section		Significant Changes Since PSAR	PSAR Reference
	2.	Containment air recirculation coolers are supplied with chilled water except during a CIA or LOP when component cooling water is automatically supplied. They were previously supplied with component cooling water during winter conditions when there were no thermal bypass lines around the reactor plant components cooling heat exchangers.	
	3.	One component cooling pump (CCP) can supply all four reactor coolant pump coolers, if only one CCP pump is available.	
9.2.2.2	1.	Urea formaldehyde air-conditioning unit load deleted.	9.2.2.2
	2.	RCS pump oil cooler, thermal barrier and bearing loads have been transferred to the reactor component cooling system.	
	3.	Added process vent cooler load.	
	4.	Reactor plant component cooling water system (CCP) - chilled water system (CDS) interaction.	
9.2.3		Ultrafiltration system (UF modules, cleaning solution skid, pH solution skid, cartridge filter, permeate tank, and pumps) have been added to the water treating system. UF membranes will remove all colloidal material and large organic molecules from the raw water. The ultrafiltered water is then forwarded to the demineralizer system.	9.2.7
9.2.4		Hydropneumatic pumps deleted. Electric hot water in lieu of steam. Domestic water service throughout plant.	9.2.6
9.2.8		Addition of 200 gpm deaerator system to the primary water system to lower oxygen level in the primary water to comply with Westinghouse requirements.	9.2.4
9.3.1.1		Addition of two shutdown air compressors to improve plant operability following a loss of power.	9.3.3.1
9.4.0		Air-conditioning units which serve areas outside the control room pressure envelope have been relocated from this control room area to outside the control room pressure envelope.	9.4.1

TABLE 1.3-16 COMPARISON OF FINAL AND PRELIMINARY INFORMATION (CONTINUED)

FSAR Section		Significant Changes Since PSAR	PSAR Reference
9.4.1.5	1.	The control building isolation signal (CBI) has been changed; it is no longer initiated by all safety injection signals.	9.4.1.5
	2.	Both trains of control building air-conditioning do not start automatically on receipt of a safety injection signal (SIS). Both trains receive a CBI signal; however, one train is normally running and the standby system will not start unless there is a fault in the running system. The CBI signal serves to block a manual stop of the running system. Since Trains A and B air-conditioning units do not operate simultaneously, the units discharge into common ductwork designed to accommodate airflow from one train only.	
	3.	The chilled water expansion tank level is alarmed on low level only, not high-low. The chilled water system was originally designed as a pressurized system and a high level alarm was necessary. However, the system is now open to the atmosphere. The expansion tank is manually filled and has an overflow connection. Therefore, a high level alarm is not required.	
9.4.1	1.	Manual diversion of exhaust air systems to filtration units occurs on receipt of high radioactivity or CIA signal.	9.4.5
	2.	Addition of safety related dampers in supply air ductwork.	
9.4.5	1.	The emergency generator enclosure ventilation system uses supply fans, not exhaust fans.	9.4.10
	2.	Fans are single speed instead of two-speed.	
	3.	Air in the emergency generator enclosure can now be recirculated.	
9.4.8	1.	The auxiliary building filter system will filter the waste disposal building exhaust air instead of main filter bank of the SLCRS.	9.4.3
	2.	The air supply equipment for the waste disposal building has been relocated from the auxiliary building to the roof of the waste disposal building.	

TABLE 1.3-16 COMPARISON OF FINAL AND PRELIMINARY INFORMATION (CONTINUED)

FSAR Section		Significant Changes Since PSAR	PSAR Reference
	3.	The waste disposal building ventilation system has been redesigned to reflect the new building layout. The system now utilizes three 50-percent units rather than two 100-percent units.	
9.4.10		Addition of hydrogen recombiner heating, ventilation and air-conditioning system.	None
9.5.1 and Fire Protection Evaluation Report (June 1977)		Change in scope attributed to increased level of fire protection design and analysis as documented in the Millstone 3 Fire Protection Evaluation, June 1977 as amended.	9.5.1
9.5.4.5	1.	Fuel oil storage tank: local fuel oil level indication deleted. (Remote indication on emergency generator panel).	9.5.4.5
	2.	Manual control changed from main control board to emergency generator panel.	
	3.	Fuel oil transfer pump discharge: Local pressure indication deleted (local flow indication provided).	
10.3.2	1.	MOVs have been replaced with AOVs in the steam supply system to the auxiliary feedwater pump turbine.	10.3
	2.	Addition of bypass valves around the main steam pressure relieving valves.	
10.4.3		The PSAR stated that the exhaust from the gland seal condenser exhaust would be passed through a charcoal filter and released from the auxiliary boiler blowdown vent stack. Now there is no charcoal filter and the release is from the turbine building roof.	11.3, Figure 11.3-1B

TABLE 1.3-16 COMPARISON OF FINAL AND PRELIMINARY INFORMATION (CONTINUED)

FSAR Section		Significant Changes Since PSAR	PSAR Reference
10.4.5		A circulating water pump is not automatically tripped in the event of low flow in the associated condenser discharge line coincident with high level in the condenser discharge strainer pit. Manual shutdown was selected as the alternative to automatic controls as a more dependable method to prevent flooding in the turbine building in the event of a condenser expansion joint failure. Circulating water flows are not measured. High level in the condenser discharge pit will be alarmed in the control room to allow the operator to manually trip the circulating water pumps in the event of the postulated rupture.	Response to PSAR Questions 10.2, 10.8
10.4.8	1.	System redesigned with a single blowdown tank and no longer contains a blowdown tank condenser.	10.4.6
	2.	Blowdown liquid drains to the steam generator blowdown tank. The steam generator blowdown tank drains to the condenser (closed cycle) or the circulating water discharge tunnel (open cycle).	
	3.	Blowdown lines are no longer isolated on a containment isolation Phase A (CIA) signal.	
	4.	Blowdown rate changed.	
10.4.9	1.	Deletion of smart valves and addition of cavitating venturies to limit flow to faulted steam generator.	10.4.3
	2.	Addition of service water system supply to pump suctions.	
	3.	Auxiliary feedwater lines join main feedwater lines inside containment.	
11.2.3		Conformance to Regulatory Guide 1.112 dated April 19, 1976.	11.2.5
11.4		The radwaste solidification system has been changed from an urea formaldehyde system to a Dow polymer solidification system.	11.5.6
11.5		Addition of radiation monitors for post accident monitoring.	11.4
14.2		Entirely rewritten to conform to Regulatory Guide 1.70, Rev. 3.	14.2

TABLE 1.3-16 COMPARISON OF FINAL AND PRELIMINARY INFORMATION (CONTINUED)

FSAR Section	Significant Changes Since PSAR	PSAR Reference
17.1.2	The Stone & Webster QA Program was revised on March 31, 1975 to conform to NRC approved Stone & Webster Topical Report SWSQAP 1-74A, Revision N/A, dated December 31, 1974.	17.1.2
17.1.3	The original QA Program implemented by WNES for Millstone 3 was described in Chapter 17 of the Millstone 3 PSAR. During the design and initial procurement activities for Millstone 3, the upgrading of the WNES QA Program reflected changes in regulatory requirements and industry standards. These changes have culminated in the present WNES QA Program as presented in WCAP-8370, Revision 7A. This revision of the WNES QA Program applies to activities within the WNES scope performed for Millstone 3 which were initiated after January 1, 1975. Section 3.1.3 and WCAP 8370, Revision 7A, include WNES positions on regulatory guides for Millstone 3.	17.1.3

1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

1.4.1 LICENSEE'S SUBSIDIARIES

Dominion Nuclear Connecticut, Inc. (DNC) is responsible for the operation, maintenance, and testing of Millstone 3. DNC is an indirect wholly owned subsidiary of Dominion Energy, which is in turn wholly owned by Dominion Resources, Inc.

1.4.2 ARCHITECT-ENGINEER

Stone & Webster Engineering Corporation (SWEC) in Boston, Massachusetts, provided engineering design and construction management services for Millstone 3. SWEC is an engineering and construction firm serving the electric utility industry in the design and construction of all types of power stations.

1.4.3 NUCLEAR STEAM SUPPLY SYSTEM MANUFACTURER

Westinghouse Electric Corporation (Westinghouse) was responsible for supplying the NSSS and first fuel load for Millstone 3.

Westinghouse has designed, developed, and manufactured nuclear facilities since the 1950s, beginning with the world's first large central station nuclear power plant (Shippingport), which has produced power since 1957. Completed or presently contracted commercial nuclear capacity totals in excess of 97,000 MW. Westinghouse pioneered new nuclear design concepts, such as chemical shim control of reactivity and the rod cluster control concept, throughout the last two decades. Among the company's own related manufacturing facilities are the Columbia Plant, Nuclear Fuel Division, the largest commercial nuclear fuel fabrication facility in the world, the Pensacola Plant which fabricates reactor internals and steam generators, and the Cheswick Plant which produces control rod drive mechanisms and reactor coolant pumps.

1.4.4 TURBINE GENERATOR MANUFACTURER

The turbine generator was manufactured by General Electric Company (GE). Design of the turbine generator was under the direction of the Steam Turbine-Generator Products Division located in Schenectady, New York.

GE has extensive experience manufacturing turbine generators for nuclear and nonnuclear applications and has supplied them for 36 operating nuclear power plants. These include 11 pressurized water reactors (PWR), 24 boiling water reactors (BWR), and 1 high temperature gas reactor (HTGR). GE is also providing turbine generators for 36 nuclear power plants in various stages of construction. These include 19 PWRs and 17 BWRs.

1.5 REQUIREMENTS FOR FURTHER TECHNICAL INFORMATION

This section has been deleted in its entirety.

1.6 GENERAL REFERENCES (HISTORICAL)

Table 1.6-1 lists topical reports which provided information that was filed separately with the Nuclear Regulatory Commission (NRC) in support of this and similar applications.

The information contained in Section 1.6 is retained for historical purposes. The review status codes previously included in this section were presented for information only and were accurate at the time of license application. Since the review status codes are not required by Regulatory Guide 1.70, are no longer applicable, and could lead to misinterpretation, they were removed in 1997 as part of a general FSAR upgrade and clarification of information.

TABLE 1.6-1 TOPICAL REPORTS AS GENERAL REFERENCES (HISTORICAL)

Report Number	Title	NRC Submittal	Reference Section(s)
WCAP-2048	The Doppler Effect for a Non-Uniform Temperature Distribution in Reactor Fuel Elements	July 1962	4.3
WCAP-2850 (Proprietary)	Single Phase Local Boiling and Bulk Boiling Pressure Drop Correlations	April 1966	4.4
WCAP-7916 (Non-proprietary)		June 1972	
WCAP-2923	In-Pile Measurement of $U0_2$ Thermal Conductivity	1966	4.4
WCAP-3269-8	Hydraulic Tests of the San Onofre Reactor Model	June 1964	4.4
WCAP-3269-26	LEOPARD - A Spectrum Dependent Non-Spatial Depletion Code for the IBM - 7094	Sept 1963	4.3, 15.0, 15.4
WCAP-3385-56	Saxton Core II Fuel Performance Evaluation, of Mass Spectrometric and Radio-Chemical Analyses of Irradiated Saxton Plutonium Fuel	July 1970	4.3, 4.4
WCAP-3680-20	Xenon-Induced Spatial Instabilities in Large PWRs (EURAEC-1974)	March 1968	4.3
WCAP-3680-21	Control Procedures for Xenon-Induced X-Y Instabilities in Large PWRs (EURAEC-2111)	Feb 1969	4.3
WCAP-3680-22	Xenon-Induced Spatial Instabilities in Three-Dimensions (EURAEC-2116)	Sept 1969	4.3
WCAP-3696-8	Pressurized Water Reactor pH - Reactivity Effect Final Report (EURAEC-2074)	Oct 1968	4.3
WCAP-3726-1	$PU0_2$ - $U0_2$ Fueled Critical Experiments	July 1967	4.3
WCAP-6065	Melting Point of Irradiated $U0_2$	Feb 1965	4.4

TABLE 1.6-1 TOPICAL REPORTS AS GENERAL REFERENCES (HISTORICAL) (CONTINUED)

Report Number	Title	NRC Submittal	Reference Section(s)
WCAP-6069	Burnup Physics of Heterogenous Reactor Lattices	June 1965	4.4
WCAP-6073	LASER - A Depletion Program for Lattice Calculations Based on MUFT and THERMOS	April 1966	4.3
WCAP-6086	Supplementary Report on Evaluation of Mass Spectrometric and Radiochemical Analyses of Yankee Core I Spent Fuel, Including Isotopes of Elements Thorium Through Curium	Aug 1969	4.3
WCAP-7015 Revision 1	Subchannel Thermal Analysis of Rod Bundle Cores	Jan 1969	4.4
WCAP-7048-PA (Proprietary)	The PANDA Code	Jan 1975	4.3
WCAP-7757-A (Non-proprietary)			
WCAP-7198-L (Proprietary)	Evaluation of Protective Coatings for Use in Reactor Containment	April 1969	6.1N
WCAP-7825 (Non-proprietary)		Dec 1971	
WCAP-7213-P-A (Proprietary)	The TURTLE 24.0 Diffusion Depletion Code	Feb 1975	4.3
WCAP-7758-A (Non-proprietary)			15.0, 15.4
WCAP-7308-L (Proprietary)	Evaluation of Nuclear Hot Channel Factor Uncertainties	Dec 1971	4.3

TABLE 1.6-1 TOPICAL REPORTS AS GENERAL REFERENCES (HISTORICAL) (CONTINUED)

Report Number	Title	NRC Submittal	Reference Section(s)
WCAP-7810 (Non-proprietary)			
WCAP-7359-L (Proprietary)	Application of THINC Program to PWR Design	Aug 1969	4.4
WCAP-7838 (Non-proprietary)		Jan 1972	
WCAP-7477-L (Proprietary)	Sensitized Stainless Steel in Westinghouse PWR Nuclear Steam Supply Systems	March 1970	5.2
WCAP-7735 (Non-proprietary)		Aug 1971	
WCAP-7488-L (Proprietary)	Solid State Logic Protection System Description	March 1971	7.2
WCAP-7672 (Non-proprietary)		June 1971	7.3
WCAP-7518-L (Proprietary)	Radiological Consequences of a Fuel Handling Accident	Dec 1971	15.7
WCAP-7828 (Non-proprietary)			
WCAP-7536-L (Proprietary)	Seismic Testing of Electrical and Control Equipment (High Seismic Plants)	Dec 1971	3.10N

TABLE 1.6-1 TOPICAL REPORTS AS GENERAL REFERENCES (HISTORICAL) (CONTINUED)

Report Number	Title	NRC Submittal	Reference Section(s)
WCAP-7821 (Non-proprietary) Supplements 1-6			
WCAP-7558	Seismic Vibration Testing with Sine Beats	Oct 1971	3.10N
WCAP-7588 Revision 1-A	An Evaluation of the Rod Ejection Accident in Westinghouse Pressurized Water Reactors Using Spatial Kinetics Methods	Jan 1975	15.4
WCAP-7623	Dynamic Fracture Toughness Properties of Heavy Section A533 Grade B Class 1 Steel Plate	Dec 1970	5.4
WCAP-7667-P-A (Proprietary)	Interchannel Thermal Mixing With Mixing Vane Grids	Jan 1975	4.4
WCAP-7755-A (Non-proprietary)			
WCAP-7695-P-A (Proprietary)	DNB Tests Results for New Mixing Vane Grids (R)	Jan 1975	4.4
WCAP-7958-A (Non-proprietary)			
WCAP-7695 Addendum 1-P-A (Proprietary)	DNB Test Results for R Grid Thimble Cold Wall Cells	Jan 1975	4.4
WCAP-7958 Addendum 1-A (Non-proprietary)			

TABLE 1.6-1 TOPICAL REPORTS AS GENERAL REFERENCES (HISTORICAL) (CONTINUED)

Report Number	Title	NRC Submittal	Reference Section(s)
WCAP-7706-L (Proprietary)	An Evaluation of Solid State Logic Reactor Protection in Anticipated Transients	Feb 1971	4.6
WCAP-7706 (Non-proprietary)			7.1
			7.2
WCAP-7769	Overpressure Protection for Westinghouse Pressurized Water Reactors	Oct 1971	15.2
WCAP-7798-L (Proprietary)	Behavior of Austenitic Stainless Steel in Post Hypothetical Loss of Coolant Accident Environment	Jan 1972	6.1N
WCAP-7803 (Non-proprietary)			
WCAP-7800 Revision 5	Nuclear Fuel Division Quality Assurance Program Plan	Nov 1984	4.2, 17.1.3
WCAP-7806	Nuclear Design of Westinghouse Pressurized Water Reactors with Burnable Poison Rods	Dec 1971	4.3
WCAP-7811	Power Distribution Control of Westinghouse Pressurized Water Reactors	Dec 1971	4.3
WCAP-7832	Evaluation of Steam Generator Tube, Tubesheet and Divider Plate Under Combined LOCA Plus SSE Conditions	Dec 1973	5.4
WCAP-7836	Inlet Orificing of Open PWR Cores	Jan 1972	4.4
WCAP-7870	Neutron Shielding Pads	May 1972	3.9N

TABLE 1.6-1 TOPICAL REPORTS AS GENERAL REFERENCES (HISTORICAL) (CONTINUED)

Report Number	Title	NRC Submittal	Reference Section(s)
WCAP-7907	LOFTRAN Code Description	June 1972	6.3, 15.0, 15.1, 15.2, 15.3, 15.4, 15.5, 15.6
WCAP-7908	FACTRAN - A FORTRAN-IV Code for Thermal Transients in a UO_2 Fuel Rod	June 1972	15.0, 15.2, 15.3, 15.4
WCAP-7912-P-A (Proprietary)	Power Peaking Factors	Jan 1975	4.3
WCAP-7912-A (Non-proprietary)			4.4
WCAP-7913	Process Instrumentation for Westinghouse Nuclear Steam Supply System (4-Loop Plant Using WCID 7300 Series Process Instrumentation)	March 1973	7.3, 7.2
WCAP-7921-AR	Damping Values of Nuclear Power Plant Components	May 1974	3.7N
WCAP-7941-P-A (Proprietary)	Effect of Axial Spacing on Interchannel Thermal Mixing with the R Mixing Vane Grid	Jan 1975	4.4
WCAP-7959-A (Non-proprietary)			
WCAP-7950	Fuel Assembly Safety Analysis for Combined Seismic and Loss of Coolant Accident	July 1972	3.7N
WCAP-7956	THINC-IV An Improved Program for Thermal-Hydraulic Analysis of Rod Bundle Cores	June 1973	4.4
WCAP-7964	Axial Xenon Transient Tests at the Rochester Gas and Electric Reactor	June 1971	4.3

TABLE 1.6-1 TOPICAL REPORTS AS GENERAL REFERENCES (HISTORICAL) (CONTINUED)

Report Number	Title	NRC Submittal	Reference Section(s)
WCAP-7979-P-A (Proprietary)	TWINKLE - A Multi-Dimensional Neutron Kinetics Computer Code	Jan 1975	15.0
WCAP-8028-A (Non-proprietary)			15.4
WCAP-7988 (Proprietary)	Application of Modified Spacer Factor to L. Grid Typical and Cold Wall Cell DNB	Oct 1972	4.4
WCAP-8030-A (Non-proprietary)			
WCAP-8054 (Proprietary)	Application of the THINC-IV Program to PWR Design	Oct 1973	4.4
WCAP-8195 (Non-proprietary)			
WCAP-8163	Reactor Coolant Pump Integrity in LOCA	Sept 1973	5.4
WCAP-8170 (Proprietary)	Calculational Model for Core Reflooding After a Loss of Coolant Accident (WREFLOOD Code)	June 1974	6.2
WCAP-8171 (Non-proprietary)			15.6
WCAP-8174 (Proprietary)	Effect of Local Heat WCAP-8202 Flux Spikes on DNB in Non-Uniform Heated Rod Bundles	Aug 1973	4.4
WCAP-8202 (Non-proprietary)			

TABLE 1.6-1 TOPICAL REPORTS AS GENERAL REFERENCES (HISTORICAL) (CONTINUED)

Report Number	Title	NRC Submittal	Reference Section(s)
WCAP-8183 Revision 6	Operational Experience With Westinghouse Cores	June 1977	4.2
WCAP-8200 Revision 2 (Proprietary)	WFLASH, A FORTRAN-IV Computer Program for Simulation of Transients in a Multi-Loop PWR	July 1974	15.6
WCAP-8261 Revision 1 (Non-proprietary)			
WCAP-8218-P-A (Proprietary)	Fuel Densification Experimental Results and Model for Reactor Application	March 1975	4.1
WCAP-8219-A (Non-proprietary)			4.2, 4.3, 4.4
WCAP-8236 (Proprietary))	Safety Analysis of the 17x17 Fuel Assembly for Combined Seismic and Loss of Coolant Accident	Jan 1974	3.7N
WCAP-8288 (Non-proprietary		Dec 1973	4.2
WCAP-8236 Addendum 1 (Proprietary)	Safety Analysis of the 8-Grid 17x17 Fuel Assembly for Combined Seismic and Loss of Coolant Accident	March 1974	3.7N
WCAP-8268 Addendum 1 (Non-proprietary)		April 1974	
WCAP-8252 Revision 1	Documentation of Selected Westinghouse Structural Analysis Computer Code	July 1977	3.9N
WCAP-8253 Amendment 1	Source Term Data for Westinghouse Pressurized Water Reactors	July 1975	11.1

TABLE 1.6-1 TOPICAL REPORTS AS GENERAL REFERENCES (HISTORICAL) (CONTINUED)

Report Number	Title	NRC Submittal	Reference Section(s)
WCAP-8255	Nuclear Instrumentation System	Jan 1974	7.2, 7.7
WCAP-8264-P-A (Proprietary)	Westinghouse Mass and Energy Release Data for Containment Design	June 1975	6.2.1
WCAP-8312-A (Non-proprietary) Revision 2			
WCAP-8278 (Proprietary)	Hydraulic Flow Test of the 17x17 Fuel Assembly	Feb 1974	4.2
WCAP-8279 (Non-proprietary)			4.4
WCAP-8296-P-A (Proprietary)	Effect of 17x17 Fuel Assembly Geometry on DNB	Feb 1975	4.4
WCAP-8297 (Non-proprietary)			
WCAP-8298-P-A (Proprietary)	The Effect of 17x17 Fuel Assembly Geometry on Interchannel Thermal Mixing	Jan 1975	4.4
WCAP-8299-A (Non-proprietary)			
WCAP-8301 (Proprietary)	LOCTA-IV Program: Loss of Coolant Transient Analysis	June 1974	15.0
WCAP-8305 (Non-proprietary)			15.6

TABLE 1.6-1 TOPICAL REPORTS AS GENERAL REFERENCES (HISTORICAL) (CONTINUED)

Report Number	Title	NRC Submittal	Reference Section(s)
WCAP-8302 (Proprietary)	SATAN-VI Program: Comprehensive Space-Time Dependent Analysis of Loss of Coolant	June 1974	6.2.1
WCAP-8306 (Non-proprietary)			15.0, 15.6
WCAP-8303-P-A (Proprietary)	Prediction of the Flow-Induced Vibration of Reactor Internals by Scale Model Tests	July 1975	3.9N
WCAP-8317-A (Non-proprietary)			
WCAP-8324-A	Control of Delta Ferrite in Austenitic Stainless Steel Weldments	June 1975	5.2
WCAP-8327 (Proprietary)	Containment Pressure Analysis Code (COCO)	June 1974	15.6
WCAP-8326 (Non-proprietary)			
WCAP-8330	Westinghouse Anticipated Transients Without Trip Analysis	Aug 1974	4.3, 4.6, 15.1, 15.2, 15.4, 15.8
WCAP-8339	Westinghouse ECCS Evaluation Model -Summary	July 1974	6.2.1, 15.6
WCAP-8340 (Proprietary)	Westinghouse ECCS - Plant Sensitivity Studies	July 1974	15.6AE
WCAP-8356 (Non-proprietary)			

TABLE 1.6-1 TOPICAL REPORTS AS GENERAL REFERENCES (HISTORICAL) (CONTINUED)

Report Number	Title	NRC Submittal	Reference Section(s)
WCAP-8341 (Proprietary)	Westinghouse ECCS Evaluation Model Sensitivity Studies	July 1974	15.6
WCAP-8342 (Non-proprietary)			
WCAP-8359	Effects of Fuel Densification Power Spikes on Clad Thermal Transients	July 1974	4.3
WCAP-8370/7800 Revision 10A/6A	Westinghouse Water Reactor Divisions Quality Assurance Plan	Nov 1984	17.1.3
WCAP-8377 (Proprietary)	Revised Clad Flattening Model	July 1974	4.2
WCAP-8381 (Non-proprietary)			
WCAP-8385 (Proprietary)	Power Distribution Control and Load Follow Procedures	Sept 1974	4.3
WCAP-8403 (Non-proprietary)			4.4
WCAP-8424 Revision 1	An Evaluation of Loss of Flow Accidents Caused by Power System Frequency Transients in Westinghouse PWRs	June 1975	15.3
WCAP-8446 (Proprietary)	17x17 Drive Line Components Tests -Phase IB, II, III, D-Loop Drop and Deflection	Dec 1974	3.9N
WCAP-8449 (Non-proprietary)			

TABLE 1.6-1 TOPICAL REPORTS AS GENERAL REFERENCES (HISTORICAL) (CONTINUED)

Report Number	Title	NRC Submittal	Reference Section(s)
WCAP-8453-A	Analysis of Data from the Zion (Unit 1) THINC Verification Test	May 1976	4.4
WCAP-8471 (Proprietary)	Westinghouse ECCS Evaluation Model -Supplement Information	April 1975	15.6
WCAP-8472 (Non-proprietary)			
WCAP-8472 (Non-proprietary)			
WCAP-8498	Incore Power Distribution Determination in Westinghouse Pressurized Water Reactors	July 1975	4.3
WCAP-8516-P (Proprietary)	UHI Plant Internals Vibration Measurement Program and Pre and Post Host Functional Examinations	April 1975	3.9N
WCAP-8517 (Non-proprietary)			
WCAP-8536 (Proprietary)	Critical Heat Flux Testing of 17x17 Fuel Assembly Geometry With 22-Inch Grid Spacing	May 1975	4.4
WCAP-8537 (Non-proprietary)			
WCAP-8565-P-A (Proprietary)	Westinghouse ECCS-Four Loop Plant (17x17) Sensitivity Studies	July 1975	15.6
WCAP-8566-A (Non-proprietary)			

TABLE 1.6-1 TOPICAL REPORTS AS GENERAL REFERENCES (HISTORICAL) (CONTINUED)

Report Number	Title	NRC Submittal	Reference Section(s)
WCAP-8584 (Proprietary)	Failure Mode and Effects Analysis (FMEA) of the Engineered Safeguard Features Actuation System	Feb 1976	4.6
WCAP-8760 (Non-proprietary)			
WCAP-8760 (Non-proprietary)			
WCAP-8587 Revision 2	Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety Related Electrical Equipment	Feb 1979	3.10N, 3.11N
WCAP-8587 Supplement 1	Equipment Qualification Data Packages	Nov 1978	3.10N, 3.11N
WCAP-8622 (Proprietary)	Westinghouse ECCS Evaluation Model -October 1975 Version	Nov 1975	15.6
WCAP-8623 (Non-proprietary)			
WCAP-8624 (Proprietary)	General Method of Developing Multi-frequency Biaxial Test Inputs for Bistables	Sept 1975	
WCAP-8695 (Non-proprietary)		Aug 1975	
WCAP-8682 (Proprietary)	Experimental Verification of Wet Fuel Storage Criticality Analyses	Dec 1975	4.3
WCAP-8683 (Non-proprietary)			

TABLE 1.6-1 TOPICAL REPORTS AS GENERAL REFERENCES (HISTORICAL) (CONTINUED)

Report Number	Title	NRC Submittal	Reference Section(s)
WCAP-8691 (Proprietary)	Fuel Rod Bowing	Dec 1975	4.2
WCAP-8692 (Non-proprietary)			
WCAP-8693	Delta Ferrite in Production Austenitic Stainless Steel Weldments	Jan 1976	5.2
WCAP-8708 (Proprietary)	MULTIFLEX - A FORTRAN-IV Computer Program for Analyzing Thermal-Hydraulic-Structure System Dynamics	Feb 1976	3.9N
WCAP-8709 (Non-proprietary)			
WCAP-8720 (Proprietary)	Improved Analytical Models Used in Westinghouse Fuel Rod Design Computations	Oct 1976	4.2
WCAP-8785 (Non-proprietary)			
WCAP-8768 Revision 2	Safety-related Research and Development for Westinghouse Pressurized Water Reactors, Program Summaries-Winter 1977-Summer 1978	Oct 1978	1.5, 4.2, 4.3
WCAP-8780	Verification of Neutron Pad and 17x17 Guide Tube Designs by Preoperational Tests on Trojan 1 Power Plant	May 1976	3.9N
WCAP-8846-A	Hybrid B_4C Absorber Control Rod Evaluation Report	Sept 1976	15.3
WCAP-8892-A (Non-proprietary)	7300 Series Process Control System Noise Tests	June 1977	7.1

TABLE 1.6-1 TOPICAL REPORTS AS GENERAL REFERENCES (HISTORICAL) (CONTINUED)

Report Number	Title	NRC Submittal	Reference Section(s)
WCAP-8929	Benchwork Problem Solutions Employed for Verification WECAN Computer Code	June 1977	3.9N
WCAP-8963 (Proprietary)	Safety Analysis for the Revised Fuel Rod Internal Pressure Design Basis	Nov 1976	4.2
WCAP-8964 (Non-proprietary)		Aug 1977	
WCAP-8970 (Proprietary)	Westinghouse ECCS Small Break October 1975 Model	April 1977	15.6
WCAP-8971 (Non-proprietary)			
WCAP-8976	Failure Mode and Effects Analysis (FMEA) of the Solid State Full Length Rod Control System	Sept 1977	4.6
WCAP-9168 (Proprietary)	Westinghouse Emergency Core Cooling System Evaluation Model - Modified October 1975 Version	Sept 1977	15.6
WCAP-9169 (Non-proprietary)			
WCAP-9179 Revision 1 (Proprietary)	Properties of Fuel and Core Component Materials	Sept 1977	4.2
WCAP-9224 (Non-proprietary)		July 1978	
WCAP-9220-P-A (Proprietary)	Westinghouse ECCS Evaluation Model, 1981 Version	Dec 1981	15.6

TABLE 1.6-1 TOPICAL REPORTS AS GENERAL REFERENCES (HISTORICAL) (CONTINUED)

Report Number	Title	NRC Submittal	Reference Section(s)
WCAP-9221-P-A (Non-proprietary) Revision 1			
WCAP-9227	Reactor Core Response to Excessive Secondary Steam Release	Jan 1978	15.1
WCAP-9485 (Proprietary)	PALADON - Westinghouse Nodal Computer Code	Dec 1978	4.3
WCAP-9486 (Non-proprietary)			
WCAP-9600	Report on Small Break Accidents for Westinghouse NSSS System	June 1979	5.4.15
WCAP-10858P-A Rev. 1	AMSAC Generic Design Package	July 1987	7.8
RP-8A (SWEC)	Radiation Shielding Design and Analysis Approach for Light Water Reactor Power Plants	May 1975	12.2, 12.3
SWSQAP-1-74A (SWEC)	Standard Nuclear Quality Assurance Program	Dec 1974	17.1.2
QA-1 Revision 3A (NUSCO)	Quality Assurance Program	March 1979	17.1.1

1.7 DRAWINGS AND OTHER DETAILED INFORMATION

1.7.1 ELECTRICAL, INSTRUMENTATION, AND CONTROL DRAWINGS

Table 1.7-1 identifies the safety related electrical, instrumentation, and control drawings used on Millstone 3.

1.7.2 PIPING AND INSTRUMENTATION DIAGRAMS

Table 1.7-2 identifies the piping and instrumentation diagrams (P&ID) used on Millstone 3. These diagrams are included throughout the FSAR in conjunction with specific system descriptions. Symbols and abbreviations used in the diagrams are illustrated on Figure 1.2–3. A more complete listing of Drawing & Figure Numbers for the P&IDs is contained in the Generation Records Information System (GRITS), and for those drawings used as Figures in the FSAR, refer to the Summary Table of Contents, and Effective Pages List.

1.7.3 LOOP AND SYSTEMS DIAGRAMS

This table has been deleted.

1.7.4 OTHER DETAILED INFORMATION (SPECIAL REPORTS AND PROGRAMS)

Table 1.7-4 identifies special reports and programs referenced in the FSAR and submitted separately from the FSAR.

TABLE 1.7-1 ELECTRICAL, INSTRUMENTATION, AND CONTROL REFERENCE DOCUMENTATION

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-39001	Westinghouse System or Title	Westinghouse Drawing Number
15001	Process Control Block Diagram	7244D80 Sh. 1
15002	Process Control Block Diagram	7244D80 Sh. 2
15003	Process Control Block Diagram	7244D80 Sh. 3
15004	Process Control Block Diagram	7244D80 Sh. 4
15005	Process Control Block Diagram	7244D80 Sh. 5
15006	Process Control Block Diagram	7244D80 Sh. 6
15007	Process Control Block Diagram	7244D80 Sh. 7
15008	Process Control Block Diagram	7244D80 Sh. 8
15009	Process Control Block Diagram	7244D80 Sh. 9
15010	Process Control Block Diagram	7244D80 Sh. 10
15011	Process Control Block Diagram	7244D80 Sh. 11
15012	Process Control Block Diagram	7244D80 Sh. 12
15013	Process Control Block Diagram	7244D80 Sh. 13
15014	Process Control Block Diagram	7244D80 Sh. 14
15015	Process Control Block Diagram	7244D80 Sh. 15
15016	Process Control Block Diagram	7244D80 Sh. 16
15017	Process Control Block Diagram	7244D80 Sh. 17
15018	Process Control Block Diagram	7244D80 Sh. 18
15019	Process Control Block Diagram	7244D80 Sh. 19
15020	Process Control Block Diagram	7244D80 Sh. 20
15021	Process Control Block Diagram	7244D80 Sh. 21
15022	Process Control Block Diagram	7244D80 Sh. 22
15023	Process Control Block Diagram	7244D80 Sh. 23
15024	Process Control Block Diagram	7244D80 Sh. 24
15025	Process Control Block Diagram	7244D80 Sh. 25
15026	Process Control Block Diagram	7244D80 Sh. 26

TABLE 1.7-1 ELECTRICAL, INSTRUMENTATION, AND CONTROL REFERENCE DOCUMENTATION (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-39001	Westinghouse System or Title	Westinghouse Drawing Number
15027	Process Control Block Diagram	7244D80 Sh. 27
15028	Process Control Block Diagram	7244D80 Sh. 28
15029	Process Control Block Diagram	7244D80 Sh. 29
15030	Process Control Block Diagram	7244D80 Sh. 30
15031	Process Control Block Diagram	7244D80 Sh. 31
15032	Process Control Block Diagram	7244D80 Sh. 32
15033	Process Control Block Diagram	7244D80 Sh. 33
15034	Process Control Block Diagram	7244D80 Sh. 34
15035	Process Control Block Diagram	7244D80 Sh. 35
15036	Process Control Block Diagram	7244D80 Sh. 36
15037	Process Control Block Diagram	7244D80 Sh. 37
15038	Process Control Block Diagram	7244D80 Sh. 38
15039	Process Control Block Diagram	7244D80 Sh. 40
15040	Process Control Block Diagram	7244D80 Sh. 39
15041	Process Control Block Diagram	7244D80 Sh. 41
15042	Process Control Block Diagram	7244D80 Sh. 42
15043	Process Control Block Diagram	7244D80 Sh. 43
15044	Process Control Block Diagram	7244D80 Sh. 44
15045	Process Control Block Diagram	7244D80 Sh. 45
15046	Process Control Block Diagram	7244D80 Sh. 46
15047	Process Control Block Diagram	7244D80 Sh. 47
15048	Process Control Block Diagram	7244D80 Sh. 48
04021	NIS Source Range Functional Block Diagram	5655D49
04022	NIS Source Range Functional Block Diagram	5655D50
04023	NIS Source Range Functional Block Diagram	5655D51
04023	NIS Source Range Functional Block Diagram	5655D52
07011	Safeguard Test Cabinets	8758D57 Sh. 1

TABLE 1.7-1 ELECTRICAL, INSTRUMENTATION, AND CONTROL REFERENCE DOCUMENTATION (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-39001	Westinghouse System or Title	Westinghouse Drawing Number
07012	Safeguard Test Cabinets	8758D57 Sh. 2
07013	Safeguard Test Cabinets	8758D57 Sh. 3
07014	Safeguard Test Cabinets	8758D57 Sh. 4
07015	Safeguard Test Cabinets	8758D57 Sh. 5
07016	Safeguard Test Cabinets	8758D57 Sh. 6
07017	Safeguard Test Cabinets	8758D57 Sh. 7
07018	Safeguard Test Cabinets	8758D57 Sh. 8
07019	Safeguard Test Cabinets	8758D57 Sh. 9
07020	Safeguard Test Cabinets	8758D57 Sh. 10
07021	Safeguard Test Cabinets	8758D57 Sh. 11
07022	Safeguard Test Cabinets	8758D57 Sh. 12
07023	Safeguard Test Cabinets	8758D57 Sh. 13
07025	Safeguard Test Cabinets	8758D57 Sh. 15
07026	Safeguard Test Cabinets	8758D57 Sh. 16
07027	Safeguard Test Cabinets	8758D57 Sh. 17
07028	Safeguard Test Cabinets	8758D57 Sh. 18
07029	Safeguard Test Cabinets	8758D57 Sh. 19
07030	Safeguard Test Cabinets	8758D57 Sh. 20
04002	Logic Diagram	108D684 Sh. 1
04003	Logic Diagram	108D684 Sh. 2
04004	Logic Diagram	108D684 Sh. 3
04005	Logic Diagram	108D684 Sh. 4
04006	Logic Diagram	108D684 Sh. 5
04007	Logic Diagram	108D684 Sh. 6
04008	Logic Diagram	108D684 Sh. 7
04009	Logic Diagram	108D684 Sh. 8
04010	Logic Diagram	108D684 Sh. 9

TABLE 1.7-1 ELECTRICAL, INSTRUMENTATION, AND CONTROL REFERENCE DOCUMENTATION (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-39001	Westinghouse System or Title	Westinghouse Drawing Number
04011	Logic Diagram	108D684 Sh. 10
04012	Logic Diagram	108D684 Sh. 11
04013	Logic Diagram	108D684 Sh. 12
04014	Logic Diagram	108D684 Sh. 13
04015	Logic Diagram	108D684 Sh. 14
04016	Logic Diagram	108D684 Sh. 15
04017	Logic Diagram	108D684 Sh. 16
04018	Logic Diagram	108D684 Sh. 17
04019	Logic Diagram	108D684 Sh. 19
04020	Logic Diagram	108D684 Sh. 18

TABLE 1.7-1 LOGIC DIAGRAMS

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-28	SWEC Drawing Number 12179-LSK-	Diagram Title
022-017	1-6A	Reactor Trips Control Logic Description Logic Diagram
022-018	1-6B	Reactor Trips Control Logic Description Logic Diagram
022-019	1-6C	Reactor Trips Control Logic Description Logic Diagram
022-020	1-6D	Reactor Trips Control Logic Description Logic Diagram
022-021	1-6E	Reactor Trips Control Logic Description Logic Diagram
022-022	1-6F	Reactor Trips Control Logic Description Logic Diagram
022-023	1-6G	Reactor Trips Control Logic Description Logic Diagram
022-024	1-6H	Reactor Trips Control Logic Description Logic Diagram
022-025	1-6J	Reactor Trips Control Logic Description Logic Diagram
022-026	1-6K	Reactor Trips Control Logic Description Logic Diagram
022-027	1-6L	Reactor Trips Control Logic Description Logic Diagram

TABLE 1.7-1 LOGIC DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-28	SWEC Drawing Number 12179-LSK-	Diagram Title
022-028	1-6M	Reactor Trips Control Logic Description Logic Diagram
022-029	1-6N	Reactor Trips Control Logic Description Logic Diagram
022-030	1-6P	AMSAC Logic Diagram
022-031	1-6Q	AMSAC Logic Diagram
025-001	2-1.1A	Circulating Water Pump Breaker
025-002	2.1-1B	Circulating Water Discharge & Water Box Valves Control Logic Diagram
025-003	2.1-1C	Circulating Water Discharge & Water Box Valves Control Logic Diagram
025-004	2.1-1D	Circulating Water Discharge & Water Box Valves Control Logic Diagram
025-005	2.1-1E	Circulating Water Discharge & Water Box Valves Control Logic Diagram
025-006	2.1-1F	Logic Diagram Circulating Water Pump
025-007	2.1-1G	Logic Diagram Circulating Water Pump
025-008	2.1-1H	Logic Diagram Circulating Water Pump
025-009	2.1-1J	Logic Diagram Circulating Water Pump
036-001	3-1.1A	Turbine Bypass Control Logic Diagram
036-002	3-1.1B	Turbine Bypass Control Logic Diagram
036-003	3-1.1C	Turbine Bypass Control Logic Diagram
036-004	3-1.1D	Turbine Bypass Control Logic Diagram
036-005	3-1.1E	Turbine Bypass Control Logic Diagram
036-006	3-1.1F	Turbine Bypass Control Logic Diagram
036-007	3-1.1G	Turbine Bypass Control Logic Diagram
036-008	3-1.1H	Turbine Bypass Control Logic Diagram
036-009	3-1.2A	Main Steam Isolation Control Logic Diagram
036-010	3-1.2B	Main Steam Isolation Control Logic Diagram
036-011	3-1.2C	Main Steam Isolation Control Logic Diagram

TABLE 1.7-1 LOGIC DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-28	SWEC Drawing Number 12179-LSK-	Diagram Title
036-012	3-1.2D	Main Steam Isolation Control Logic Diagram
036-013	3-1.2E	Main Steam Isolation Control Logic Diagram
061-001	3-9A	Aux Stm Sys Control Logic Diagram
061-002	3-9B	Aux Stm Sys Control Logic Diagram
120-001	6-1.1A	Motor Drive Stm Gen Pump & Discharge Valve Control Logic Diagram
120-002	6-1.1B	Motor Drive Stm Gen Pump & Discharge Valve Control Logic Diagram
120-003	6-1.1C	Motor Drive Stm Gen Pump & Discharge Valve Control Logic Diagram
120-T019	6-1.1D	Motor Drive Stm Gen Pump & Discharge Valve Logic Diagram
120-004	6-1.2A	Fdwtr Sys Control Logic Diagram
120-005	6-1.2B	Fdwtr Sys Control Logic Diagram
120-006	6-1.2C	Fdwtr Sys Control Logic Diagram
120-007	6-1.2D	Fdwtr Sys Control Logic Diagram
120-008	6-1.2E	Fdwtr Sys Control Logic Diagram
126-001	6-2.1A	Motor Drive Aux Fdwtr Pump & Recirc Control Logic Diagram
126-002	6-2.1B	Motor Drive Aux Fdwtr Pump & Recirc Control Logic Diagram
126-003	6-2.1C	Motor Drive Aux Fdwtr Pump & Recirc Control Logic Diagram
126-004	6-2.1D	Motor Drive Aux Fdwtr Pump & Recirc Control Logic Diagram
126-005	6-2.1E	Motor Drive Aux Fdwtr Pump & Recirc Control Logic Diagram
126-006	6-2.1F	Motor Drive Aux Fdwtr Pump & Recirc Control Logic Diagram

TABLE 1.7-1 LOGIC DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-28	SWEC Drawing Number 12179-LSK-	Diagram Title
126-007	6-2.1G	Motor Drive Aux Fdwtr Pump & Recirc Control Logic Diagram
126-008	6-2.1H	Motor Drive Aux Fdwtr Pump & Recirc Control Logic Diagram
126-009	6-2.1J	Motor Drive Aux Fdwtr Pump & Recirc Control Logic Diagram
126-010	6-2.2A	Turbine-Driven Aux Fdwtr Pump & Recirc Index Control Logic Diagram
126-011	6-2.2B	Turbine-Driven Aux Fdwtr Pump & Recirc Index Control Logic Diagram
126-012	6-2.2C	Turbine-Driven Aux Fdwtr Pump & Recirc Index Control Logic Diagram
575	7-3.2	Aux Fdwtr Pump & Drive Lube Oil Control Logic Diagram
143-001	8-9A	Emergency Diesel Generator Fuel Control Logic Diagram
143-002	8-9B	Emergency Diesel Generator Fuel Control Logic Diagram
148-001	9-1A	Reactor Plant Component Cooling Water Control Logic Diagram
148-002	9-1B	Reactor Plant Component Cooling Water Control Logic Diagram
148-003	9-1C	Reactor Plant Component Cooling Water Control Logic Diagram
148-004	9-1D	Reactor Plant Component Cooling Water Control Logic Diagram
148-005	9-1E	Reactor Plant Component Cooling Water Control Logic Diagram
148-006	9-1F	Reactor Plant Component Cooling Water Control Logic Diagram
148-007	9-1G	Reactor Plant Component Cooling Water Control Logic Diagram

TABLE 1.7-1 LOGIC DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-28	SWEC Drawing Number 12179-LSK-	Diagram Title
148-008	9-1H	Reactor Plant Component Cooling Water Control Logic Diagram
148-009	9-1J	Reactor Plant Component Cooling Water Control Logic Diagram
148-010	9-1K	Reactor Plant Component Cooling Water Control Logic Diagram
165-001	9-2A	Chilled Water System Control Logic Diagram
165-002	9-2B	Chilled Water System Control Logic Diagram
165-003	9-2C	Chilled Water System Control Logic Diagram
165-004	9-2D	Chilled Water System Control Logic Diagram
165-005	9-2E	Chilled Water System Control Logic Diagram
165-006	9-2F	Chilled Water System Control Logic Diagram
165-007	9-2G	Chilled Water System Control Logic Diagram
165-008	9-2H	Chilled Water System Control Logic Diagram
165-009	9-2J	Chilled Water System Control Logic Diagram
165-010	9-2K	Chilled Water System Control Logic Diagram
173-001	9-4A	Charging Pumps Cooling Control Logic Diagram
173-002	9-4B	Charging Pumps Cooling Control Logic Diagram
173-003	9-4C	Charging Pumps Cooling Control Logic Diagram
173-004	9-4D	Charging Pumps Cooling Control Logic Diagram
179-001	9-5A	Safety Injection Pumps Cooling Control Logic Diagram
179-002	9-5B	Safety Injection Pumps Cooling Control Logic Diagram
184-001	9-7A	Turbine Plant Component Cooling Water Control Logic Diagram
184-002	9-7B	Turbine Plant Component Cooling Water Control Logic Diagram
184-003	9-7C	Turbine Plant Component Cooling Water Control Logic Diagram

TABLE 1.7-1 LOGIC DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-28	SWEC Drawing Number 12179-LSK-	Diagram Title
184-004	9-7D	Turbine Plant Component Cooling Water Control Logic Diagram
184-005	9-7E	Turbine Plant Component Cooling Water Control Logic Diagram
193-001	9-10A	Service Water System Control Logic Diagram
193-002	9-10B	Service Water System Control Logic Diagram
193-003	9-10C	Service Water System Control Logic Diagram
193-004	9-10D	Service Water System Control Logic Diagram
193-005	9-10E	Service Water System Control Logic Diagram
193-006	9-10F	Service Water System Control Logic Diagram
193-007	9-10G	Service Water System Control Logic Diagram
193-008	9-10H	Service Water System Control Logic Diagram
193-009	9-10J	Service Water System Control Logic Diagram
193-010	9-10K	Service Water System Control Logic Diagram
193-011	9-10L	Service Water System Control Logic Diagram
193-012	9-10M	Service Water System Control Logic Diagram
224-001	12-1A	Instrument Air Control Logic Diagram
224-002	12-1B	Instrument Air Control Logic Diagram
224-003	12-1C	Instrument Air Control Logic Diagram
224-004	12-1D	Instrument Air Control Logic Diagram
224-001	12-1A	Instrument Air Control Logic Diagram
224-002	12-1B	Instrument Air Control Logic Diagram
224-005	12-1E	Instrument Air Control Logic Diagram
224-006	12-1F	Instrument Air Control Logic Diagram
224-007	12-1G	Instrument Air Control Logic Diagram
224-008	12-1H	Instrument Air Control Logic Diagram
224-009	12-1J	Instrument Air Control Logic Diagram

TABLE 1.7-1 LOGIC DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-28	SWEC Drawing Number 12179-LSK-	Diagram Title
234-003	12-3C	Containment Instrument Air Control Logic Diagram
267-001	14-1A	Nitrogen System Control Logic Diagram
267-002	14-1B	Nitrogen System Control Logic Diagram
267-003	14-1C	Nitrogen System Control Logic Diagram
267-004	14-1D	Nitrogen System Control Logic Diagram
283-001	15-03A	Fire Protection Low Press CO_2 Logic Diagram
283-002	15-03B	Fire Protection Low Press CO_2 Logic Diagram
283-003	15-03C	Fire Protection Low Press CO_2 Logic Diagram
283-004	15-03D	Fire Protection Low Press CO_2 Logic Diagram
283-005	15-03E	Fire Protection Low Press CO_2 Logic Diagram
320-001	22-1A	Reactor Plant Ventilation Control Logic Diagram
320-002	22-1B	Reactor Plant Ventilation Control Logic Diagram
320-003	22-1C	Reactor Plant Ventilation Control Logic Diagram
320-004	22-1D	Reactor Plant Ventilation Control Logic Diagram
320-005	22-1E	Reactor Plant Ventilation Control Logic Diagram
320-006	22-1F	Reactor Plant Ventilation Control Logic Diagram
320-007	22-1G	Reactor Plant Ventilation Control Logic Diagram
320-008	22-1H	Reactor Plant Ventilation Control Logic Diagram
320-009	22-1J	Reactor Plant Ventilation Control Logic Diagram
320-010	22-1K	Reactor Plant Ventilation Control Logic Diagram
320-011	22-1L	Reactor Plant Ventilation Control Logic Diagram
320-012	22-1M	Reactor Plant Ventilation Control Logic Diagram
320-013	22-1N	Reactor Plant Ventilation Control Logic Diagram
320-014	22-1P	Reactor Plant Ventilation Control Logic Diagram
320-015	22-1Q	Reactor Plant Ventilation Control Logic Diagram
320-016	22-1R	Reactor Plant Ventilation Control Logic Diagram

TABLE 1.7-1 LOGIC DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-28	SWEC Drawing Number 12179-LSK-	Diagram Title
320-017	22-1S	Reactor Plant Ventilation Control Logic Diagram
320-018	22-1T	Reactor Plant Ventilation Control Logic Diagram
320-019	22-1U	Reactor Plant Ventilation Control Logic Diagram
320-020	22-1V	Reactor Plant Ventilation Control Logic Diagram
320-021	22-1W	Reactor Plant Ventilation Control Logic Diagram
320-022	22-1X	Reactor Plant Ventilation Control Logic Diagram
320-023	22-1Y	Reactor Plant Ventilation Control Logic Diagram
320-024	22-1Z	Reactor Plant Ventilation Control Logic Diagram
320-025	22-1AA	Reactor Plant Ventilation Control Logic Diagram
320-026	22-1AB	Reactor Plant Ventilation Control Logic Diagram
339-001	22-7A	Diesel Gen Bldg Ventilation Control Logic Diagram
339-002	22-7B	Diesel Gen Bldg Ventilation Control Logic Diagram
339-003	22-7C	Diesel Gen Bldg Ventilation Control Logic Diagram
342-001	22-8A	Yard Structure Ventilation Control Logic Diagram
342-002	22-8B	Yard Structure Ventilation Control Logic Diagram
342-003	22-8C	Yard Structure Ventilation Control Logic Diagram
342-004	22-8D	Yard Structure Ventilation Control Logic Diagram
342-005	22-8E	Yard Structure Ventilation Control Logic Diagram
342-006	22-8F	Yard Structure Ventilation Control Logic Diagram
342-007	22-8G	Yard Structure Ventilation Control Logic Diagram
342-008	22-8H	Yard Structure Ventilation Control Logic Diagram
342-009	22-8J	Yard Structure Ventilation Control Logic Diagram
342-010	22-8K	Yard Structure Ventilation Control Logic Diagram
342-011	22-8L	Yard Structure Ventilation Control Logic Diagram
342-012	22-8M	Yard Structure Ventilation Control Logic Diagram
342-013	22-8N	Yard Structure Ventilation Control Logic Diagram
342-014	22-8P	Yard Structure Ventilation Control Logic Diagram

TABLE 1.7-1 LOGIC DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-28	SWEC Drawing Number 12179-LSK-	Diagram Title
342-016	22-8S	Yard Structure Ventilation Control Logic Diagram
345-001	22-9A	Control Bldg A/C Control Logic Diagram
345-002	22-9B	Control Bldg A/C Control Logic Diagram
345-003	22-9C	Control Bldg A/C Control Logic Diagram
345-004	22-9D	Control Bldg A/C Control Logic Diagram
345-005	22-9E	Control Bldg A/C Control Logic Diagram
345-006	22-9F	Control Bldg A/C Control Logic Diagram
345-007	22-9G	Control Bldg A/C Control Logic Diagram
345-008	22-9H	Control Bldg A/C Control Logic Diagram
345-009	22-9J	Control Bldg A/C Control Logic Diagram
345-010	22-9K	Control Bldg A/C Control Logic Diagram
345-011	22-9L	Control Bldg A/C Control Logic Diagram
345-012	22-9M	Control Bldg A/C Control Logic Diagram
345-013	22-9N	Control Bldg A/C Control Logic Diagram
345-014	22-9P	Control Bldg A/C Control Logic Diagram
366-001	22-12A	Chilled Water System Control Logic Diagram
366-002	22-12B	Chilled Water System Control Logic Diagram
366-003	22-12C	Chilled Water System Control Logic Diagram
366-004	22-12D	Chilled Water System Control Logic Diagram
366-005	22-12E	Chilled Water System Control Logic Diagram
366-006	22-12F	Chilled Water System Control Logic Diagram
366-007	22-12G	Chilled Water System Control Logic Diagram
366-008	22-12H	Chilled Water System Control Logic Diagram
366-009	22-12J	Chilled Water System Control Logic Diagram
366	22-12K	Chilled Water System Control Logic Diagram
366	22-12L	Chilled Water System Control Logic Diagram
366	22-12M	Chilled Water System Control Logic Diagram

TABLE 1.7-1 LOGIC DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-28	SWEC Drawing Number 12179-LSK-	Diagram Title
373-001	22-16A	Hot Water Heating Logic Diagram
373-002	22-16B	Hot Water Heating Logic Diagram
373-003	22-16C	Hot Water Heating Logic Diagram
373-004	22-16D	Hot Water Heating Logic Diagram
373-005	22-16E	Hot Water Heating Logic Diagram
587-001	22-26A	ESF Bldg Ventilation Control Logic Diagram
587-002	22-26B	ESF Bldg Ventilation Control Logic Diagram
587-003	22-26C	ESF Bldg Ventilation Control Logic Diagram
587-004	22-26D	ESF Bldg Ventilation Control Logic Diagram
587-005	22-26E	ESF Bldg Ventilation Control Logic Diagram
587-006	22-26F	ESF Bldg Ventilation Control Logic Diagram
587-007	22-26G	ESF Bldg Ventilation Control Logic Diagram
587-008	22-26H	ESF Bldg Ventilation Control Logic Diagram
587-009	22-26J	ESF Bldg Ventilation Control Logic Diagram
592-001	22-27A	Containment Structure Vent Control Logic Diagram
592-002	22-27B	Containment Structure Vent Control Logic Diagram
592-003	22-27C	Containment Structure Vent Control Logic Diagram
592-004	22-27D	Containment Structure Vent Control Logic Diagram
592-005	22-27E	Containment Structure Vent Control Logic Diagram
592-006	22-27F	Containment Structure Vent Control Logic Diagram
592-007	22-27G	Containment Structure Vent Control Logic Diagram
595-001	22-28A	Main Stm Valve Bldg Ventilation Control Logic Diagram
595-002	22-28B	Main Stm Valve Bldg Ventilation Control Logic Diagram
391-001	22-33A	Hot Water Preheating Sys Control Logic Diagram
391-002	22-33B	Hot Water Preheating Sys Control Logic Diagram
708-001	24-3A	Reserve Station Service Breaker Controls Control Logic Description

TABLE 1.7-1 LOGIC DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-28	SWEC Drawing Number 12179-LSK-	Diagram Title
708-002	24-3B	Reserve Station Service Breaker Controls Control Logic Diagram
708-003	24-3C	Reserve Station Service Breaker Controls Control Logic Diagram
708-004	24-3D	Reserve Station Service Breaker Controls Control Logic Diagram
708-005	24-3E	Reserve Station Service Breaker Controls Control Logic Diagram
708-006	24-3F	Reserve Station Service Breaker Controls Control Logic Diagram
708-007	24-3G	Reserve Station Service Breaker Controls Control Logic Diagram
708-008	24-3H	Reserve Station Service Breaker Controls Control Logic Diagram
708-009	24-3J	Reserve Station Service Breaker Controls Control Logic Diagram
708-010	24-3K	Reserve Station Service Breaker Controls Control Logic Diagram
709-001	24-4A	Medium Voltage Bus Tie Breaker Controls Control Logic Diagram
709-002	24-4B	Medium Voltage Bus Tie Breaker Controls Control Logic Diagram
713-001	24-8A	Low Voltage Switchgear Supply Breaker Controls Control Logic Diagram
713-002	24-8B	Low Voltage Switchgear Supply Breaker Controls Control Logic Diagram
721-001	24-9.2A	Emergency Gen Breaker Controls Control Logic Diagram
721-002	24-9.2B	Emergency Gen Breaker Controls Control Logic Diagram
721-003	24-9.2C	Emergency Gen Breaker Controls Control Logic Diagram
721-004	24-9.2D	Emergency Gen Breaker Controls Control Logic Diagram

TABLE 1.7-1 LOGIC DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-28	SWEC Drawing Number 12179-LSK-	Diagram Title
722-001	24-9.3A	Emergency Diesel Gen Control & Protection Control Logic Diagram
722-002	24-9.3B	Emergency Diesel Gen Control & Protection Control Logic Diagram
722-003	24-9.3C	Emergency Diesel Gen Control & Protection Control Logic Diagram
722-004	24-9.3D	Emergency Diesel Gen Control & Protection Control Logic Diagram
722-005	24-9.3E	Emergency Diesel Gen Control & Protection Control Logic Diagram
722-006	24-9.3F	Emergency Diesel Gen Control & Protection Control Logic Diagram
722-007	24-9.3G	Emergency Diesel Gen Control & Protection Control Logic Diagram
722-008	24-9.3H	Emergency Diesel Gen Control & Protection Control Logic Diagram
722-009	24-9.3J	Emergency Diesel Gen Control & Protection Control Logic Diagram
722-010	24-9.3K	Emergency Diesel Gen Control & Protection Control Logic Diagram
722-011	24-9.3L	Emergency Diesel Gen Control & Protection Control Logic Diagram
722-012	24-9.3M	Emergency Diesel Gen Control & Protection Control Logic Diagram
722-013	24-9.3N	Emergency Diesel Gen Control & Protection Control Logic Diagram
722-014	24-9.3O	Emergency Diesel Gen Control & Protection Control Logic Diagram
722-015	24-9.3P	Emergency Diesel Gen Control & Protection Control Logic Diagram
723-001	24-9.4A	Emergency Gen Load Sequence Control Logic Diagram

TABLE 1.7-1 LOGIC DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-28	SWEC Drawing Number 12179-LSK-	Diagram Title
723-002	24-9.4B	Emergency Gen Load Sequence Control Logic Diagram
723-003	24-9.4C	Emergency Gen Load Sequence Control Logic Diagram
723-004	24-9.4D	Emergency Gen Load Sequence Control Logic Diagram
723-005	24-9.4E	Emergency Gen Load Sequence Control Logic Diagram
723-006	24-9.4F	Emergency Gen Load Sequence Control Logic Diagram
723-007	24-9.4G	Emergency Gen Load Sequence Control Logic Diagram
723-008	24-9.4H	Emergency Gen Load Sequence Control Logic Diagram
723-009	24-9.4J	Emergency Gen Load Sequence Control Logic Diagram
723-010	24-9.4K	Emergency Gen Load Sequence Control Logic Diagram
723-011	24-9.4L	Emergency Gen Load Sequence Control Logic Diagram
723-012	24-9.4M	Emergency Gen Load Sequence Control Logic Diagram
723-013	24-9.4N	Emergency Gen Load Sequence Control Logic Diagram
723-014	24-9.4P	Emergency Gen Load Sequence Control Logic Diagram
723-015	24-9.4Q	Emergency Gen Load Sequence Control Logic Diagram
723-016	24-9.4R	Emergency Gen Load Sequence Control Logic Diagram
723-017	24-9.4S	Emergency Gen Load Sequence Control Logic Diagram
723-018	24-9.4T	Emergency Gen Load Sequence Control Logic Diagram
723-019	24-9.4U	Emergency Gen Load Sequence Control Logic Diagram
723-020	24-9.4V	Emergency Gen Load Sequence Control Logic Diagram
723-021	24-9.4W	Emergency Gen Load Sequence Control Logic Diagram
723-022	24-9.4X	Emergency Gen Load Sequence Control Logic Diagram
723-023	24-9.4Y	Emergency Gen Load Sequence Control Logic Diagram
723-024	24-9.4Z	Emergency Gen Load Sequence Control Logic Diagram
714-001	24-10A	Battery Power Supply Control Logic Functional Diagram
714-002	24-10B	Battery Power Supply Control Logic Diagram
714-003	24-10C	Battery Power Supply Control Logic Diagram

TABLE 1.7-1 LOGIC DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-28	SWEC Drawing Number 12179-LSK-	Diagram Title
714-004	24-11A	Instrument and Control AC Power Supply Control Logic Functional Diagram
714-005	24-11B	Instrument and Control AC Power Supply Control Logic Functional Diagram
714-006	24-11C	Instrument and Control AC Power Supply Control Logic Functional Diagram
724-001	24-12.5A	Synchronizing Check Control Logic Diagram
724-002	24-12.5B	Synchronizing Check Control Logic Diagram
724-003	24-12.5C	Synchronizing Check Control Logic Diagram
724-004	24-12.5D	Synchronizing Check Control Logic Diagram
404-001	25-1.1A	Reactor Coolant Pumps Control Logic Diagram
404-002	25-1.1B	Reactor Coolant Pumps Control Logic Diagram
404-003	25-1.1C	Reactor Coolant Pumps Control Logic Diagram
404-004	25-1.1D	Reactor Coolant Pumps Control Logic Diagram
404-005	25-1.2A	Pressurized Control Logic Diagram
404-006	25-1.2B	Pressurized Control Logic Diagram
404-007	25-1.2C	Pressurized Control Logic Diagram
404-008	25-1.2D	Pressurized Control Logic Diagram
404-009	25-1.2E	Pressurized Control Logic Diagram
404-010	25-1.2F	Pressurized Control Logic Diagram
404-011	25-1.2G	Pressurized Control Logic Diagram
404-012	25-1.2H	Pressurized Control Logic Diagram
404-013	25-1.J	Pressurized Control Logic Diagram
404-014	25-1.K	Pressurized Control Logic Diagram
404-015	25-1.L	Pressurized Control Logic Diagram
404-016	25-1.4A	Pressurizer Relief Tank Control Logic Diagram
404-017	25-1.4B	Pressurizer Relief Tank Control Logic Diagram
504-018	25-1.5A	Reactor Coolant Isolation Valves Control Logic Diagram

TABLE 1.7-1 LOGIC DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-28	SWEC Drawing Number 12179-LSK-	Diagram Title
504-019	25-1.5B	Reactor Coolant Isolation Valves Control Logic Diagram
504-020	25-1.5C	Reactor Coolant Isolation Valves Control Logic Diagram
414-001	26-2.1A	Reactor Coolant Letdown Control Logic Diagram
414-002	26-2.1B	Reactor Coolant Letdown Control Logic Diagram
414-003	26-2.1C	Reactor Coolant Letdown Control Logic Diagram
414-004	26-2.1D	Reactor Coolant Letdown Control Logic Diagram
414-005	26-2.2A	Volume Control Tank Control Logic Diagram
414-006	26-2.2B	Volume Control Tank Control Logic Diagram
414-007	26-2.2C	Volume Control Tank Control Logic Diagram
414-008	26-2.2D	Volume Control Tank Control Logic Diagram
414-010	26-2.3A	Charging Pumps Control Logic Diagram
414-011	26-2.3B	Charging Pumps Control Logic Diagram
414-012	26-2.3C	Charging Pumps Control Logic Diagram
414-013	26-2.3D	Charging Pumps Control Logic Diagram
414-014	26-2.3E	Charging Pumps Control Logic Diagram
414-015	26-2-3F	Charging Pumps Control Logic Diagram
414-016	26-2.3G	Charging Pumps Control Logic Diagram
414-017	26-2.3H	Charging Pumps Control Logic Diagram
414-018	26-2.3J	Charging Pumps Control Logic Diagram
414-029	26-2.5A	Reactor Makeup & Boric Acid Blender Control Logic Diagram
414-030	26-2.5B	Reactor Makeup & Boric Acid Blender Control Logic Diagram
414-031	26-2.5C	Reactor Makeup & Boric Acid Blender Control Logic Diagram
414-032	26-2.5D	Reactor Makeup & Boric Acid Blender Control Logic Diagram

TABLE 1.7-1 LOGIC DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-28	SWEC Drawing Number 12179-LSK-	Diagram Title
414-033	26-2.5E	Reactor Makeup & Boric Acid Blender Control Logic Diagram
414-034	26-2.5F	Reactor Makeup & Boric Acid Blender Control Logic Diagram
414-035	26-2.5G	Reactor Makeup & Boric Acid Blender Control Logic Diagram
414-036	26-2.6A	Reactor Coolant Pumps Seal Water Control Logic Diagram
414-037	26-2.6B	Reactor Coolant Pumps Seal Water Control Logic Diagram
434-001	27-2A	High Pressure Safety Injection Control Logic Diagram
434-002	27-2B	High Pressure Safety Injection Control Logic Diagram
434-003	27-2C	High Pressure Safety Injection Control Logic Diagram
434-004	27-2D	High Pressure Safety Injection Control Logic Diagram
434-005	27-2E	High Pressure Safety Injection Control Logic Diagram
434-006	27-2F	High Pressure Safety Injection Control Logic Diagram
434-007	27-2G	High Pressure Safety Injection Control Logic Diagram
434-008	27-2H	High Pressure Safety Injection Control Logic Diagram
434-009	27-2J	High Pressure Safety Injection Control Logic Diagram
434-010	27-2K	High Pressure Safety Injection Control Logic Diagram
434-011	27-2L	High Pressure Safety Injection Control Logic Diagram
441-001	27-3A	Low Pressure Safety Injection Control Logic Diagram
441-002	27-3B	Low Pressure Safety Injection Control Logic Diagram
441-003	27-3C	Low Pressure Safety Injection Control Logic Diagram
441-004	27-3D	Low Pressure Safety Injection Control Logic Diagram
441-005	27-3E	Low Pressure Safety Injection Control Logic Diagram
441-006	27-3F	Low Pressure Safety Injection Control Logic Diagram
441-007	27-3G	Low Pressure Safety Injection Control Logic Diagram

TABLE 1.7-1 LOGIC DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-28	SWEC Drawing Number 12179-LSK-	Diagram Title
441-008	27-3H	Low Pressure Safety Injection Control Logic Diagram
449-001	27-7A	Residual Heat Removal Control Logic Diagram
449-002	27-7B	Residual Heat Removal Control Logic Diagram
449-003	27-7C	Residual Heat Removal Control Logic Diagram
449-004	27-7D	Residual Heat Removal Control Logic Diagram
449-005	27-7E	Residual Heat Removal Control Logic Diagram
449-006	27-7F	Residual Heat Removal Control Logic Diagram
449-007	27-7G	Residual Heat Removal Control Logic Diagram
449-008	27-7H	Residual Heat Removal Control Logic Diagram
449-009	27-7J	Residual Heat Removal Control Logic Diagram
449-010	27-7K	Residual Heat Removal Control Logic Diagram
454-001	27-10A	Containment Vacuum Control Logic Diagram
454-002	27-10B	Containment Vacuum Control Logic Diagram
457-001	27-11A	Containment Recirc Control Logic Diagram
457-002	27-11B	Containment Recirc Control Logic Diagram
457-003	27-11C	Containment Recirc Control Logic Diagram
457-004	27-11D	Containment Recirc Control Logic Diagram
457-005	27-11E	Containment Recirc Control Logic Diagram
457-006	27-11F	Containment Recirc Control Logic Diagram
457-007	27-11G	Containment Recirc Control Logic Diagram
457-008	27-11H	Containment Recirc Control Logic Diagram
457-019	27-11J	Containment Recirc Control Logic Diagram
457-010	27-11K	Containment Recirc Control Logic Diagram
457-011	27-11L	Containment Recirc Control Logic Diagram
463-001	27-12A	Quench Spray Control Logic Diagram
463-002	27-12B	Quench Spray Control Logic Diagram
463-003	27-12C	Quench Spray Control Logic Diagram

TABLE 1.7-1 LOGIC DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-28	SWEC Drawing Number 12179-LSK-	Diagram Title
463-004	27-12D	Quench Spray Control Logic Diagram
463-005	27-12E	Quench Spray Control Logic Diagram
463-006	27-12F	Quench Spray Control Logic Diagram
470-001	27-13A	DBA Hydrogen Recombiner Control Logic Diagram
470-002	27-13B	DBA Hydrogen Recombiner Control Logic Diagram
470-003	27-13C	DBA Hydrogen Recombiner Control Logic Diagram
470-004	27-13D	DBA Hydrogen Recombiner Control Logic Diagram
470-005	27-13E	DBA Hydrogen Recombiner Control Logic Diagram
719-001	27-17A	Safety Injection Actuation Control Logic Diagram
719-002	27-17B	Safety Injection Actuation Control Logic Diagram
719-003	27-17C	Safety Injection Actuation Control Logic Diagram
716-001	27-18A	Containment Spray Actuation Control Logic Diagram
716-002	27-18B	Containment Spray Actuation Control Logic Diagram
716-003	27-18C	Containment Spray Actuation Control Logic Diagram
720-001	27-19A	Containment Isolation Control Logic Diagram
720-002	27-19B	Containment Isolation Control Logic Diagram
720-003	27-19C	Containment Isolation Control Logic Diagram
720-004	27-19D	Containment Isolation Control Logic Diagram
720-005	27-19E	Containment Isolation Control Logic Diagram
720-006	27-19F	Containment Isolation Control Logic Diagram
720-007	27-19G	Containment Isolation Control Logic Diagram
482-012	31-1.1	High Level Waste Drn. TK/Pump
482-002	31-1.2A	Waste Eval Reblr. Pump/Waste Dist. Pump
482-003	31-1.2B	Waste Eval Reblr. Pump/Waste Dist. Pump
482-004	31-1.2C	Waste Eval Reblr. Pump/Waste Dist. Pump

TABLE 1.7-1 LOGIC DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-28	SWEC Drawing Number 12179-LSK-	Diagram Title
482-005	31-1.2D	Waste Eval Reblr. Pump/Waste Dist. Pump
482-006	31-1.2E	Waste Eval Reblr. Pump/Waste Dist. Pump
482-018	31-1.3A	Waste Test TK/Pump
482-019	31-1.3B	Waste Test TK/Pump
482-020	31-1.3C	Waste Test TK/Pump
482-021	31-1.4A	Waste Evap Bot Pump/Waste Bot. Coolant Pump
482-022	31-1.4B	Waste Evap Bot Pump/Waste Bot. Coolant Pump
482-023	31-1.4C	Waste Evap Bot Pump/Waste Bot. Coolant Pump
482-024	31-1.4D	Waste Evap Bot Pump/Waste Bot. Coolant Pump
482-025	31-1.4E	Waste Evap Bot Pump/Waste Bot. Coolant Pump
482-026	31-1.5A	Low Level Wast. Drn. TK/Pump
482-027	31-1.5B	Low Level Wast. Drn. TK/Pump
492-001	31-2.1A	Degasifier & Recirc Pumps Control Logic Diagram
492-002	31-2.1B	Degasifier & Recirc Pumps Control Logic Diagram
492-003	31-2.1C	Degasifier & Recirc Pumps Control Logic Diagram
492-004	31-2.1D	Degasifier & Recirc Pumps Control Logic Diagram
492-005	31-2.1E	Degasifier & Recirc Pumps Control Logic Diagram
492-006	31-2.1F	Degasifier & Recirc Pumps Control Logic Diagram
492-007	31-2.1G	Degasifier & Recirc Pumps Control Logic Diagram
492-008	31-2.1H	Degasifier & Recirc Pumps Control Logic Diagram
516-001	32-3A	Reactor Plant Gaseous Drains Control Logic Diagram
516-002	32-3B	Reactor Plant Gaseous Drains Control Logic Diagram
516-003	32-3C	Reactor Plant Gaseous Drains Control Logic Diagram
516-004	32-3D	Reactor Plant Gaseous Drains Control Logic Diagram
522-001	32-4A	Reactor Plant Aerated Drains Control Logic Diagram
522-002	32-4B	Reactor Plant Aerated Drains Control Logic Diagram

TABLE 1.7-1 LOGIC DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-28	SWEC Drawing Number 12179-LSK-	Diagram Title
522-003	32-4C	Reactor Plant Aerated Drains Control Logic Diagram
522-004	32-4D	Reactor Plant Aerated Drains Control Logic Diagram
522-005	32-4E	Reactor Plant Aerated Drains Control Logic Diagram
640-001	32-5.1A	Turbine Plant Misc Drains Control Logic Diagram
640-002	32-5.1B	Turbine Plant Misc Drains Control Logic Diagram
640-002A	32-5.1B1	Turbine Plant Misc Drains Control Logic Diagram
640-003	32-5.1C	Turbine Plant Misc Drains Control Logic Diagram
640-004	32-5.1D	Turbine Plant Misc Drains Control Logic Diagram
546-001	32-13A	Steam Generator Blowdown System Control Logic Diagram
546-002	32-13B	Steam Generator Blowdown System Control Logic Diagram
546-003	32-13C	Steam Generator Blowdown System Control Logic Diagram
554	33-1	Containment Leakage Monitoring Control Logic Diagram
557	33-2	Containment Atmosphere Monitoring Control Logic Diagram
560-001	34-1A	Fuel Pool Cooling & Purification System Control Logic Diagram
560-002	34-1B	Fuel Pool Cooling & Purification System Control Logic Diagram
560-003	34-1C	Fuel Pool Cooling & Purification System Control Logic Diagram
560-004	34-1D	Fuel Pool Cooling & Purification System Control Logic Diagram
560-006	34-1E	Fuel Pool Cooling & Purification System Control Logic Diagram
567-001	35-1A	Primary Grade Water Control Logic Diagram
567-002	35-1B	Primary Grade Water Control Logic Diagram
567-003	35-1C	Primary Grade Water Control Logic Diagram

TABLE 1.7-1 LOGIC DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-28	SWEC Drawing Number 12179-LSK-	Diagram Title
567-004	35-1D	Primary Grade Water Control Logic Diagram

TABLE 1.7-1 ONE-LINE DIAGRAMS - CLASS 1E

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Dwg. Number 25212-30	SWEC Drawing Number 12179-EE-	Diagram Title
001	1A	Main One-Line Diagram Power Distribution Composite
027	1AD	480 V MCC One-Line Diagram (Bus 32-1R, 1W) Aux Bldg (Sh. 3)
063	1AE	480 V MCC One-Line Diagram (Bus 32-5G, 5H, 5T, 5U) Circulating Water Pump-house
040	1AH	480 V MCC One-Line Diagram (Bus 32-2F, 4T) ESF Bldg (Sh. 1)
041	1AJ	480 V MCC One-Line Diagram (Bus 32-2J, 3U, 4U) ESF Bldg (Sh. 2)
059	1AK	480 V MCC One-Line Diagram Bus 32-3D, 1L, 1T, 1U) Diesel Enclosure & Auxiliary Bldg
047	1AQ	480 V MCC One-Line Diagram (Bus 32-2R, 2W) Rod Control Area (Sh. 2)
061	1AS	480 V MCC One-Line Diagram (Bus 32-2T, 2U) Control Bldg
065	1AT	480 V MCC One-Line Diagram (Bus 32-3A, 3P, 3T) Turbine Building (Sheet 4)
076	1BA	125 V dc and 120 V ac Distribution Sys Composite
077	1BB	125 V dc One-Line Diagram Batteries 301A-1 and 301A-2
078	1BC	125 V dc One-Line Diagram Batteries 301B-1 and 301B-2
079	1BD	125 V dc One-Line Diagram Batteries 301C-1
080	1BE	125 V dc One-Line Diagram Battery 301D-A
081	1BF	120 V ac One-Line Diagram Vital Bus I and III

TABLE 1.7-1 ONE-LINE DIAGRAMS - CLASS 1E (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Dwg. Number 25212-30	SWEC Drawing Number 12179-EE-	Diagram Title
082	1BG	120 V ac One-Line Diagram Vital Bus II and IV
083	1BH	120 V ac One-Line Diagram Vital Bus Back-Up 1 and 2
087	1BM	125 V Misc dc One-Line Diagram Battery 1
088	1BN	125 V Misc dc One-Line Diagram Battery 2 (Sh. 1)
106	1BR	125 V Misc dc Line Diagram Batteries 1 & 2
107	1BS	125 V dc One-Line Diagram Batteries 1 & 2
108 (Sh. 1)	1BT	125 V Misc dc One-Line Diagram
122 (Sh. 3)	1CC	120 V Misc ac One-Line Diagram
126 (Sh. 7)	1CG	120 V Misc ac One-Line Diagram
127 (Sh. 8)	1CH	120 V Misc ac One-Line Diagram
128 (Sh. 9)	1CJ	120 V Misc ac One-Line Diagram
129 (Sh. 10)	1CK	120 V Misc ac One-Line Diagram
125 (Sh. 12)	1CM	120 V Misc ac One-Line Diagram
004	1D	Main One-Line Diagram 4,160 & 480 Normal
028	1EF	480 V One-Line Diagram (Bus 32Y, X) [3EJS*US-4A, 4B]
30030	IEH	480 V One-Line Diagram Emergency Diesel Generator 480 V Distribution Panels
018	1K	4.16 kV One-Line Diagram Bus 34C [3ENS*SWG-A(-0)] (Sh. 1)
019	1L	4.16 kV One-Line Diagram Bus 34C [3ENS*SWG-A(0)] (Sh. 2)
020	1M	4.16 kV One-Line Diagram Bus 34D [3ENS*SWG-B(P)] (Sh. 1)
021	1N	4.16 kV One-Line Diagram Bus 34D [3ENS*SWG-B(P)] (Sh. 2)
055	1U	480 V One-Line Diagram (Bus 32T, U) [3EJS*US-1A, 1B]
032	1V	480 V One-Line Diagram (Bus 32S, V) [3EJS*US-2A, 2B]
033	1W	480 V One-Line Diagram (Bus 32R, W) [3EJS*US-3A, 3B]

TABLE 1.7-1 ELEMENTARY DIAGRAMS

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-32001	SWEC Drawing Number 12179-ESK-	Diagram Title
Sh. 2A	2A	Device Numbers and General Notes
Sh. 2B	2B	Code Information Guide
Sh. 2C	2C	Elementary Diagram Symbols
Sh. 2D	2D	Elementary Diagram Symbols
Sh. 2E	2E	Elementary Diagram Symbols
Sh. 2F	2F	Elementary Diagram Symbols
Sh. 2G	2G	One-Line Diagram Symbols
Sh. 2H	2H	One-Line Diagram Symbols
Sh. 2J	2J	One-Line Diagram Symbols
Sh. 2K	2K	One-Line Diagram Symbols
Sh. 2L	2L	Legend and Location Symbols
Sh. 2M	2M	Legend and Location Symbols
Sh. 2N	2N	Legend and Location Symbols
Sh. 2P	2P	Legend and Location Symbols
Sh. 2Q	2Q	Typ isolation CKT & presentation on ESK
32001	3A	Control Switch Contact Diagram
32001	3B	Control Switch Contact Diagram
32001	3C	Control Switch Contact Diagram
32001	3D	Control Switch Contact Diagram
32001	3E	Control Switch Contact Diagram
32001	3F	Control Switch Contact Diagram
32001	3G	Control Switch Contact Diagram
32001	3H	Control Switch Contact Diagram
32001	3J	Control Switch Contact Diagram
32001	3K	Control Switch Contact Diagram
32001	3L	Control Switch Contact Diagram

TABLE 1.7-1 ELEMENTARY DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-32001	SWEC Drawing Number 12179-ESK-	Diagram Title
32001	3M	Control Switch Contact Diagram
32001	3N	Control Switch Contact Diagram
32001	3P	Control Switch Contact Diagram
32001	3Q	Control Switch Contact Diagram
32001	3R	Control Switch Contact Diagram
32001	3S	Control Switch Contact Diagram
32001	3T	Control Switch Contact Diagram
32001	3U	Control Switch Contact Diagram
32001	3V	Control Switch Contact Diagram
32001	3W	Control Switch Contact Diagram
32001	3X	Control Switch (OIM) Contact Diagram
32001	3Y	Control Switch Contact Diagram
32001	3Z	Control Switch Contact Diagram
32001	3AA	Control Switch Contact Diagram
32001	3AB	Control Switch Contact Diagram
32001	3AC1	Control Switch Contact Diagram
32001	3AC2	Control Switch Contact Diagram
32001	3AD	Control Switch Contact Diagram
32001	3AE	Control Switch Contact Diagram
32001	4AAA	Outline Isolator Cabinets
32001	4AAH01	Fire Transfer Switch Panel (FTSP) 3CES*PNLFTSP
32001	4AAH02	Fire Transfer Switch Panel (FTSP) 3CES*PNLFTSP
32001	4AAH03	Fire Transfer Switch Panel (FTSP) 3CES*PNLFTSP
32001	4A01	Main Control Board Outline
32001	4BA03	Auxiliary Shutdown Panel 3RPS*PNLAS
32001	4BA04	Auxiliary Shutdown Panel 3RPS*PNLAS

TABLE 1.7-1 ELEMENTARY DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-32001	SWEC Drawing Number 12179-ESK-	Diagram Title
32001	4BA05	Auxiliary Shutdown Panel 3RPS*PNLAS
Sh. 4BA6	4BA6	Auxiliary Shutdown Panel (ASP) 3RPS*PNLAS (Sh.6)
Sh. 4BA7	4BA7	Auxiliary Shutdown Panel (ASP) 3RPS*PNLAS (Sh.7)
Sh. 4BA8	4BA8	Auxiliary Shutdown Panel (ASP) 3RPS*PNLAS (Sh.8)
Sh. 4BA9	4BA9	Auxiliary Shutdown Panel (ASP) 3RPS*PNLAS (Sh.9)
32001	4B02	MB1 Front Section
32001	4B03	MB1 Front Section
32001	4B04	MB1 Front Section
32001	4B05	MB1 Front Section
32001	4B06	Main Control Board Front Section MB1
32001	4B07	Main Control Board Front Section MB1
32001	4B08	Main Control Board Front Section MB1
32001	4B09	MB1 Front Section
32001	4B10	Main Control Board Front Section MB1
32001	4B11	MB1 Front Section
32001	4C02	Main Control Board Front Section MB2
32001	4C03	Main Control Board Front Section MB2
32001	4C04	Main Control Board Front Section MB2
32001	4C05	Main Control Board Front Section MB2
32001	4C06	Main Control Board Front Section MB2
32001	4C07	Main Control Board Front Section MB2
32001	4C08	Main Control Board Front Section MB2
32001	4C09	Main Control Board Front Section MB2
32001	4C10	Main Control Board Front Section MB2
32001	4C11	Main Control Board Front Section MB2
32001	4D02	Main Control Board Front Section MB3

TABLE 1.7-1 ELEMENTARY DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-32001	SWEC Drawing Number 12179-ESK-	Diagram Title
32001	4D03	Main Control Board Front Section MB3
32001	4D04	Main Control Board Front Section MB3
32001	4D05	Main Control Board Front Section MB3
32001	4D06	Main Control Board Front Section MB3
32001	4D07	Main Control Board Front Section MB3
32001	4D08	Main Control Board Front Section MB3
32001	4D09	Main Control Board Front Section MB3
32001	4D11	Main Control Board Front Section MB3
32001	4E02	Main Control Board Front Section MB4
32001	4E03	Main Control Board Front Section MB4
32001	4E04	Main Control Board Front Section MB4
32001	4E05	Main Control Board Front Section MB4
32001	4E06	Main Control Board Front Section MB4
32001	4E07	Main Control Board Front Section MB4
32001	4E08	Main Control Board Front Section MB4
32001	4F02	Main Control Board Front Section MB5
32001	4F03	Main Control Board Front Section MB5
32001	4F04	Main Control Board Front Section MB5
32001	4F05	Main Control Board Front Section MB5
32001	4F06	Main Control Board Front Section MB5
32001	4F07	Main Control Board Front Section MB5
32001	4F08	Main Control Board Front Section MB5
32001	4F09	Main Control Board Front Section MB5
32001	4F10	Main Control Board Front Section MB5
32001	4F11	Main Control Board Front Section MB5
32001	4F13	Main Control Board Front Section MB5

TABLE 1.7-1 ELEMENTARY DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-32001	SWEC Drawing Number 12179-ESK-	Diagram Title
32001	4G02	Main Control Board Front Section MB6
32001	4G03	Main Control Board Front Section MB6
32001	4G04	Main Control Board Front Section MB6
32001	4G05	Main Control Board Front Section MB6
32001	4G06	Main Control Board Front Section MB6
32001	4G07	Main Control Board Front Section MB6
32001	4G07A	Main Control Board Front Section MB6
32001	4H02	Main Control Board Front Section MB7
32001	4H03	Main Control Board Front Section MB7
32001	4H04	Main Control Board Front Section MB7
32001	4H05	Main Control Board Front Section MB7
32001	4H06	Main Control Board Front Section MB7
32001	4H07	Main Control Board Front Section MB7
32001	4H08	Main Control Board Front Section MB7
32001	4H09	Main Control Board Front Section MB7
32001	4H10	Main Control Board Front Section MB7
32001	4H12	Main Control Board Front Section MB7
32001	4J02	Main Control Board Front Section MB8
32001	4J03	Main Control Board Front Section MB8
32001	4J04	Main Control Board Front Section MB8
32001	4J05	Main Control Board Front Section MB8
32001	4J06	Main Control Board Front Section MB8
32001	4J07	Main Control Board Front Section MB8
32001	4J08	Main Control Board Front Section MB8
32001	4J09	Main Control Board Front Section MB8
32001	4J13	Main Control Board Front Section MB8

TABLE 1.7-1 ELEMENTARY DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-32001	SWEC Drawing Number 12179-ESK-	Diagram Title
32001	4K02	Main Control Board Front Section MBIR
32001	4K03	Main Control Board Front Section MB1R
32001	4K04	Main Control Board Front Section MB1R
Sh. 4LA3	4LA3	Main Vent and Air Cond Panel 3HVC*PNLVP1
Sh. 4LA4	4LA4	Main Vent and Air Cond Panel 3HVC*PNLVP1
Sh. 4LA5	4LA5	Main Vent and Air Cond Panel 3HVC*PNLVP1
Sh. 4LA6	4LA6	Main Vent and Air Cond Panel 3HVC*PNLVP1
Sh. 4LA7	4LA7	Main Vent and Air Cond Panel 3HVC*PNLVP1
Sh. 4LA8	4LA8	Main Vent and Air Cond Panel 3HVC*PNLVP1
Sh. 4LA9	4LA9	Main Vent and Air Cond Panel 3HVC*PNLVP1
Sh. 4LA10	4LA10	Main Vent and Air Cond Panel 3HVC*PNLVP1
Sh. 4LA11	4LA11	Main Vent and Air Cond Panel 3HVS*PNLVP1
Sh. 4LA12	4LA12	Main Vent and Air Cond Panel 3HVS*PNLVP1
Sh. 4LA13	4LA13	Main Vent and Air Cond Panel 3HVS*PNLVP1
Sh. 4LA14	4LA14	Main Vent & Air Cond Panel 3HVS*PNLVP1
Sh. 4LA15	4LA15	Main Vent and Air Cond Panel
Sh. 4L2	4L2	Main Control Board Rear Section MB2R
Sh. 4L3	4L3	Main Control Board Rear Section MB2R
Sh. 4L4	4L4	Main Control Board Rear Section MB2R
Sh. 4M2	4M2	Main Control Board Rear Section MB3R
Sh. 4M3	4M3	Main Control Board Rear Section MB3R
Sh. 4N2	4N2	Main Control Board Rear Section MB4R
Sh. 4N3	4N3	Main Control Board Rear Section MB4R
Sh. 4N4	4N4	Main Control Board Rear Section MB4R
Sh. 4P2	4P2	Main Control Board Rear Section MB5R
Sh. 4P3	4P3	Main Control Board Rear Section MB5R

TABLE 1.7-1 ELEMENTARY DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-32001	SWEC Drawing Number 12179-ESK-	Diagram Title
Sh. 4P4	4P4	Main Control Board Rear Section MB5R
Sh. 4P5	4P5	Main Control Board Rear Section MB5R
Sh. 4P6	4P6	Main Control Board Rear Section MB5R
Sh. 4P7	4P7	Main Control Board Rear Section MB5R
Sh. 4P8	4P8	Main Control Board Rear Section MB5R
Sh. 4Q2	4Q2	Main Control Board Rear Section MB6R
Sh. 4Q3	4Q3	Main Control Board Rear Section MB6R
Sh. 4Q4	4Q4	Main Control Board Rear Section MB6R
Sh. 4Q5	4Q5	Main Control Board Rear Section MB6R
Sh. 4Q6	4Q6	Main Control Board Rear Section MB6R
Sh. 4R2	4R2	Main Control Board Rear Section MB7R
Sh. 4R3	4R3	Main Control Board Rear Section MB7R
Sh. 4R4	4R4	Main Control Board Rear Section MB7R
Sh. 4R5	4R5	Main Control Board Rear Section MB7R
Sh. 4R6	4R6	Main Control Board Rear Section MB7R
Sh. 4S2	4S2	Main Control Board Rear Section MB8R
Sh. 4S3	4S3	Main Control Board Rear Section MB8R
Sh. 4S4	4S4	Main Control Board Rear Section MB8R
Sh. 4S5	4S5	Main Control Board Rear Section MB8R
Sh. 4S6	4S6	Main Control Board Rear Section MB8R
Sh. 4S7	4S7	Main Control Board Rear Section MB8R
Sh. 4U1	4U1	Post Accident Sampling Panel 3SSP*PNL3
Sh. 4U2	4U2	Post Accident Sampling Panel 3SSP*PNL3
Sh. 4U3	4U3	Post Accident Sampling Panel 3SSP*PNL3
Sh. 4U4	4U4	Post Accident Sampling Panel 3SSP*PNL3
Sh. 4U5	4U5	Post Accident Sampling Panel 3SSP*PNL3

TABLE 1.7-1 ELEMENTARY DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-32001	SWEC Drawing Number 12179-ESK-	Diagram Title
Sh. 4YA3	4YA3	Auxiliary Relay Rack (AR4) 3RPS*Rakotxa
Sh. 4YA4	4YA4	Auxiliary Relay Rack (AR4) 3RPS*Rakotxa
Sh. 4YA5	4YA5	Auxiliary Relay Rack (AR4) 3RPS*Rakotxa
Sh. 4ZA3	4ZA3	Auxiliary Relay Rack (AR5)3RPS*Rakotxb
Sh. 4ZA4	4ZA4	Auxiliary Relay Rack (AR5) 3RPS*Rakotxb
Sh. 4ZA5	4ZA5	Auxiliary Relay Rack (AR5) 3RPS*Rakotxb
Sh. 4ZB	4ZB	Aux Relay & Cont Panel 3HVC*PNL CHLIA Cont Bldg Chld Wtr Sys-Train "A" Aux
Sh. 4ZC	4ZC	Aux Relay & Cont Panel 3HVC*PNL CHIB Cont Bldg Chld Wtr Sys-Train "B"
Sh. 4ZD2	4ZD2	Transfer Switch Panel Train A (TSPA) 3 ES*PNL TSA
Sh. 4ZD3	4ZD3	Transfer Switch Panel Train A (TSPA) 3 ES*PNL TSA
Sh. 4ZD4	4ZD4	Transfer Switch Panel Train A (TSPA) 3CES*PNL TSA
Sh. 4ZE2	4ZE2	Transfer Switch Panel Train A (TSPA) 3CES*PNL TSA
Sh. 4ZE3	4ZE3	Transfer Switch Panel Train A (TSPA) 3CES*PNL TSA
Sh. 5A	5A	Typ Med Voltage Swgr ACB
Sh. 5BB	5BB	4.16 KV Norm Stat Service BKr
Sh. 5BC	5BC	Norm Sta Svce Bkr [3NNS-ACB-BN] 35A3-34B-2
Sh. 5BD	5BD	Rsv Sta Svce Bkr [3ENS*ACB-AR] 23SA3-34C-2
Sh. 5BE	5BE	Rsv Sta Svce Bkr [3ENS*ACB-BR] 23SA-3-34D-2
Sh. 5BF	5BF	Bus Tie Bkr [3ENS*ACB-TA] 34C-1T-2
Sh. 5BG	5BG	Bus Tie Bkr [3ENS*ACB-TB] 34D-1T-2
Sh. 5BT	5BT	US Fdr Bkr [3ENS*ACB-AA], 34C3-2
Sh. 5BU	5BU	US Fdr Bkr [3ENS*ACB-AB], 34C4-2
Sh. 5BV	5BV	US Fdr Bkr [3ENS*ACB-BA], 34D2-2
Sh. 5BW	5BW	US Fdr Bkr [3ENS*ACB-BB], 34D3-2
Sh. 5BX	5BX	US Fdr Bkr [3ENS*ACB-AC], 34C5-2

TABLE 1.7-1 ELEMENTARY DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-32001	SWEC Drawing Number 12179-ESK-	Diagram Title
Sh. 5BY	5BY	US Fdr Bkr [3ENS*ACB-BC], 34D4-2
Sh. 5CJ	5CJ	Service Wtr Pump [3SWP*P1A]
Sh. 5CK	5CK	Service Wtr Pump [3SWP*P1B]
Sh. 5CL	5CL	Service Wtr Pump [3SWP*P1C]
Sh. 5CM	5CM	Service Wtr Pump
Sh. 5CN	5CN	Cntmt Recirc Pump [3RSS*P1A]
Sh. 5CP	5CP	Cntmt Recirc Pump [3RSS*P1B]
Sh. 5CQ	5CQ	Cntmt Recirc Pump [3RSS*P1C]
Sh. 5CR	5CR	Cntmt Recirc Pump [3RSS*P1D]
Sh. 5CS	5CS	Charging Pump P3A [3CHS*P3A]
Sh. 5CT	5CT	Charging Pump P3B [3CHS*P3B]
Sh. 5CU	5CU	Charging Pump P3C (Spare) [3CHS*P3C]
Sh. 5CV	5CV	Charging Pump P3C (Spare) [3CHS*P3C]
Sh. 5DA	5DA	Reactor Plant Comp Cooling Wtr Pp [3CCP*P1A]
Sh. 5DB	5DB	Reactor Plant Comp Cooling Wtr Pp [3CCP*P1B]
Sh. 5DC	5DC	Reactor Plant Comp Cooling Wtr Swing Pp [3CCP*P1C]
Sh. 5DD	5DD	Reactor Plant Comp Cooling Wtr Swing Pp [3CCP*P1C]
Sh. 5DE	5DE	Residual Heat Removal Pump P1A [3RHS*P1A]
Sh. 5DF	5DF	Residual Heat Removal Pump P1B [3RHS*P1B]
Sh. 5DG	5DG	Quench Spray Pump P3A (3QSS*P3A)
Sh. 5DH	5DH	Quench Spray Pump P3B (3QSS*P3B)
Sh. 5DJ	5DJ	Safety Injection Pump P1A [3SIH*P1A]
Sh. 5DK	5DK	Safety Injection Pump P1B [3SIH*P1B]
Sh. 5DR	5DR	Emer Diesel Gen Bkr [3ENS*ACB-G-A] 15G-14U-2
Sh. 5DS	5DS	Emer Diesel Gen Bkr [3ENS*ACB-G-B] 15G-15U-2
Sh. 5DX	5DX	Stm Gen Aux Fdwtr Pp Mot Driven P1A [3FWA*P1A]

TABLE 1.7-1 ELEMENTARY DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-32001	SWEC Drawing Number 12179-ESK-	Diagram Title
Sh. 5DY	5DY	Stm Gen Aux Fdwtr Pp Mot Driven P1B [3FWA*P1B]
Sh. 5DZ	5DZ	Cntrl Bldg Chilled Wtr Chiller CHL1A [3HVK*CHL1A]
Sh. 5EA	5EA	Cntrl Bldg Chilled Wtr Chiller CHL1B [3HVK*CHL1B]
Sh. 5EB	5EB	Emer Gen Neutral Bkr [3ENS-ACB-GNA] 15G-14U-2N
Sh. 5EC	5EC	Emer Gen Neutral Bkr [3ENS-ACB-GNB] 15G-15U-2N
Sh. 5EX	5EX	Elem Diag Misc Ckts Sta 3VCE Bkr Control
Sh. 5EY	5EY	4.16 kV - US Fdr Bkr [3ENS*ACB-AD]
Sh. 5EZ	5EZ	4.16 kV - US Fdr Bkr [3ENS*ACB-BD]
Sh. 5FD	5FD	AH D.G. Bkr Backup Protection
Sh. 6A	6A	Typical 480 V ACB
Sh. 6M	6M	Emer Supply Bkr [3EJS*ACB-AA]
Sh. 6N	6N	Emer Supply Bkr [3EJS*ACB-AB]
Sh. 6P	6P	Emer Supply Bkr [3EJS*AC]
Sh. 6Q	6Q	Emer Supply Bkr [3EJS*ACB-BA]
Sh. 6R	6R	Emer Supply Bkr [3EJS*ACB-BB]
Sh. 6S	6S	Emer Supply Bkr [3EJS*ACB-BC]
Sh. 6AG	6AG	Fuel Pool Cooling Pump [3SFC*P1A]
Sh. 6AH	6AH	Fuel Pool Cooling Pump [3SFC*P1B]
Sh. 6AN	6AN	Pressurizer Htrs BU GP A [3RCS*PH1A]
Sh. 6AP	6AP	Pressurizer H Htrs BU GP B [3RCS*PH1A]
Sh. 6AV	6AV	Crdm Shroud [3HVU-FN2A]
Sh. 6AW	6AW	Crdm Shroud Fan [3HVU-FN2B]
Sh. 6AY	6AY	Inst Air Comp Fdr Bkr
Sh. 6BA	6BA	Cntmt Air Recirc Fan [3HVU-FN1A]
Sh. 6BB	6BB	Cntmt Air Recirc Fan [3HVU-FN1B]
Sh. 6DD	6DD	Serv Wtr Pp Str and Backwash V [3SWP*STR1A], [3SWP*MOV24A]

TABLE 1.7-1 ELEMENTARY DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-32001	SWEC Drawing Number 12179-ESK-	Diagram Title
Sh. 6DE	6DE	Serv Wtr Pp Str and Backwash V [3SWP*STR1B], [3SWP*MOV24B]
Sh. 6DF	6DF	Serv Wtr Pp Str and Backwash V [3SWP*STR1C], [3SWP*MOV24C]
Sh. 6DG	6DG	Serv Wtr Pp Str and Backwash V [3SWP*STR1D], [3SWP*MOV24D]
Sh. 6DX	6DX	Emer Gen Fuel Oil Xfer pps [3EGF*P1A,C]
Sh. 6DY	6DY	Emer Gen Fuel Oil Xfer Pps [3EGF*P1B,D]
Sh. 6GD	6GD	Boric Acid Transfer Pump [3CHS*P2A]
Sh. 6GE	6GE	Boric Acid Transfer Pump [3CHS*P2B]
Sh. 6GH	6GH	Charging Pumps Cooling Pumps [3CCE*P1A,B]
Sh. 6GT	6GT	Cont Bldg Chilled Wtr Pump [3HVK*P1A]
Sh. 6GU	6GU	Cont Bldg Chilled Wtr Pump [3HVK*P1B]
Sh. 6GV	6GV	Cont Bldg Chilled Lube Oil Pumps [3HVK*P3A] & [3HVK*P3B]
Sh. 6LD	6LD	Cntmt Recir Wtr Spray Hdr Isol VV20A
Sh. 6LE	6LE	Cntmt Recir Wtr Spray Hdr Isol VV20B
Sh. 6LF	6LF	Cntmt Recir Wtr Spray Hdr Isol VV20C
Sh. 6LG	6LG	Cntmt Recir Wtr Spray Hdr Isol VV20D
Sh. 6LH	6LH	Cont Recir Pump Suct Isol VV23A
Sh. 6LJ	6LJ	Cont Recir Pump Suct Isol VV23B
Sh. 6LK	6LK	Cont Recir Pmp Suct Isol VV23C
Sh. 6LL	6LL	Cont Recir Pump Suct Isol VV23D
Sh. 6LM	6LM	Rss to Rhr Cross Connect MV8837A
Sh. 6LN	6LN	Rss to Rhr Cross Connect MV8838A
Sh. 6LP	6LP	Rss to Rhr Cross Connect MV8837B
Sh. 6LQ	6LQ	Rss to Rhr Cross Connect MV8838B

TABLE 1.7-1 ELEMENTARY DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-32001	SWEC Drawing Number 12179-ESK-	Diagram Title
Sh. 6LS	6LS	Quench Spray Header Isol Valve (3QSS*MOV34A)
Sh. 6LT	6LT	Quench Spray Header Isol Valve (3QSS*MOV34B)
Sh. 6MF	6MF	Refueling Wtr Stor Tk to SIP [3SIH*MV8806]
Sh. 6MG	6MG	SIP Suction [3SIH*MV8923A]
Sh. 6MH	6MH	SIP Suction [3SIH*MV8923B]
Sh. 6MJ	6MJ	SIP Disch HDR Isol Valve [3SIH*MV8821A]
Sh. 6MK	6MK	SIP Disch Hdr Isol Valve [3SIH*MV8821B]
Sh. 6ML	6ML	SIP to Cold Legs [3SIH*MV8835]
Sh. 6MN	6MN	Hi Press SIP Mini Flow Isol [3SIH*MV8813]
Sh. 6MP	6MP	Suction Hdr Cross Connection [3SIH*MV8807A]
Sh. 6MQ	6MQ	Suction Header Cross Connection [3SIH*MV8807B]
Sh. 6MR	6MR	SIP to Hot Legs [3SIH*MV8802A]
Sh. 6MS	6MS	SIP to Hot Legs [3SIH*MV8802B]
Sh. 6MV	6MV	Boron Inj Tk Disch Isol [3SIH*MV8801A]
Sh. 6MW	6MW	Boron Inj Tk Disch Isol (3SIH*MV8801B)
Sh. 6MX	6MX	Rwst to Rhr P1A Isolation 3SIL*MV8812A
Sh. 6MY	6MY	Rwst to Rhr P1B Isolation [3SIL*MV8812B]
Sh. 6MZ	6MZ	Rhr to Cold Leg Isol [3SIL*MV8809A]
Sh. 6NA	6NA	Rhr To Cold Leg Isol [3SIL*MV8809B]
Sh. 6NB	6NB	Accumulator Isolation [3SIL* MV8808A]
Sh. 6NC	6NC	Accumulator Isolation [3SIL*MV8808B]
Sh. 6ND	6ND	Accumulator Isolation [3SIL*MV8808C]
Sh. 6NE	6NE	Accumulator Isolation [3SIL*MV8808D]
Sh. 6NF	6NF	Rhr P1A to Charging Pump [3SIL*MV8804A]
Sh. 6NG	6NG	Rhr P1B to Charging Pump [3SIL*MOV8804B]
Sh. 6NH	6NH	Residual Ht Removal to Hot Legs [3SIL*MV8840]

TABLE 1.7-1 ELEMENTARY DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-32001	SWEC Drawing Number 12179-ESK-	Diagram Title
Sh. 6NJ	6NJ	Cross Connection Isolation VV22A (3RHS*MV8716A)
Sh. 6NK	6NK	Cross Connection Isolation Valve [3RHS*MV8716B]
Sh. 6NU	6NU	HI Press SIP Mini Flow Isol [3SIH*MV8814]
Sh. 6NV	6NV	Hi Press SIP Mini Flow Isol [3SIH*MV8920]
Sh. 6NW	6NW	SI Charging Suction Cross Connection [3SIH*MV8924]
Sh. 6PA	6PA	Rcp Seal Wtr Isol [3CHS*MV8109A]
Sh. 6PB	6PB	Rcp Seal Wtr Isol [3CHS*MV8109B]
Sh. 6PC	6PC	Rcp Seal Wtr Isol [3CHS*MV8109C]
Sh. 6PD	6PD	Rcp Seal Wtr Isol [3CHS*MV8109D]
Sh. 6PE	6PE	Charging Pump Mini-flow Isol Valve [3CHS*V8110]
Sh. 6PF	6PF	Charging Pump Mini-flow Isol Valve [3CHS*MV8111]
Sh. 6PG	6PG	Charging Pump to Rx Clt Sys Isol Valve [3CHS*MV8105]
Sh. 6PH	6PH	Charging Pump to Rx Clt Sys Isol Valve [3CHS*MV8106]
Sh. 6PJ	6PJ	Boric Acid Fltr to Charging Pmp Valve [3CHS*MV8104]
Sh. 6PK	6PK	Vol Cont Tk Outlet Isol VV112B
Sh. 6PL	6PL	Vol Cont Tk Outlet Isol VV112C
Sh. 6PM	6PM	Refueling Wtr Stor Tk to Charging Pmp VV112D
Sh. 6PN	6PN	Refueling Wtr Stor Tk to Charging Pmp V112E
Sh. 6PP	6PP	Rx Clt Pmp Seal Wtr Isol [3CHS*MV8112]
Sh. 6PQ	6PQ	Rx Clt Pmp Seal Wtr Isol [3CHS*MV8100]
Sh. 6QR	6QR	RHS Inlet Isolation Valve [RHS*MV8701C]
Sh. 6QS	6QS	RHS Inlet Isolation Valve [3RHSMV8702C]
Sh. 6QT	6QT	RHS Inlet Isolation Valve 3RHSMV8701A
Sh. 6QU	6QU	RHS Inlet Isolation Valve 3RHSMV8702B
Sh. 6QV	6QV	RHS Inlet Isolation Valve 3RHSMV8701B
Sh. 6QW	6QW	RHR Inlet Isolation Valve 3RHS*MC8702A

TABLE 1.7-1 ELEMENTARY DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-32001	SWEC Drawing Number 12179-ESK-	Diagram Title
Sh. 6QX	6QX	RHR Pmp P1A Mini-flow Recirc VV610
Sh. 6QY	6QY	RHR Pmp P1B Mini-flow Recirc VV611
Sh. 6RJ	6RJ	Cntrl Rm Area A/C Units ACU1A, 1B
Sh. 6RL	6RL	Battery Rms 1, 2&5 ACU3A, 3B
Sh. 6RM	6RM	Cable Spread & Swgr, Equip Rm ACU4A & 4B
Sh. 6RP	6RP	Cntmt Bldg Chill Wtr Pp Aux Ckt A
Sh. 6RQ	6RQ	Cntmt Bldg Chill Wtr Pp Aux Ckt B
Sh. 6RR	6RR	Cntmt Bldg Air Cond Booster Pp (3SWP*P2A, B)
Sh. 6RV	6RV	Chiller Equip Space Supply Fan - FN2A, FN2B
Sh. 6RW	6RW	Chiller Equip Space Exhaust Fan - FN7A, 7B
Sh. 6RX	6RX	Cntmt Bldg Emerg Vent Fan - FN1A & Inlet Damper MOD33A
Sh. 6RY	6RY	Cntmt Bldg Emerg Vent Fan - FN1B & Inlet Damper MOD33B
Sh. 6SB	6SB	Aux Bldg Exh Fan & Assoc Dmprs [3HVK*FNGA, AOD20A & 26A]
Sh. 6SC	6SC	Aux Bldg Fltr Exh Exh Fan & Assoc Dmprs
Sh. 6SD	6SD	Chg Pp Cub Sply Fan & Assoc Dmprs
Sh. 6SE	6SE	Chg Pp Cub Sply Fan & Assoc Dmprs
Sh. 6SF	6SF	Chg Pp Exh Fan & Assoc Dmprs
Sh. 6SG	6SG	Chg Pp Exh Fan & Assoc Dmprs
Sh. 6SH	6SH	Fuel Bldg Exh Fan & Assoc Dmprs
Sh. 6SJ	6SJ	Fuel Bldg Exh Fan & Assoc Dmprs
Sh. 6SK	6SK	Slcr Exh Fan & Assoc Dmprs
Sh. 6SL	6SL	Slcr Exh Fan & Assoc Dmprs
Sh. 6SM	6SM	MCC, Rod Cont & Cable Vault Area ACU
Sh. 6TD	6TD	Pressurizer Relief Isol Valve [3RCS*MV8000A]
Sh. 6TE	6TE	Pressurizer Relief Isol Valve [3RCS*MV8000B]
Sh. 6TF	6TF	Reac Clnt LP1 Hot Leg Stop Valve (3RCS*MV8001A)

TABLE 1.7-1 ELEMENTARY DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-32001	SWEC Drawing Number 12179-ESK-	Diagram Title
Sh. 6TG	6TG	Reac Clnt LP2 Hot Leg Stop Valve (3RCS*MV8001B)
Sh. 6TH	6TH	Reac Clnt LP3 Hot Leg Stop Valve (3RCS*MV8001C)
Sh. 6TJ	6TJ	Reac Clnt LP4 Hot Leg Stop Valve (3RCS*MV8001D)
Sh. 6TK	6TK	Reac Clnt LP1 Cold Leg Stop Valve (3RCS*MV8002A)
Sh. 6TL	6TL	Reac Clnt LP2 Cold Leg Stop Valve (3RCS*MV8002B)
Sh. 6TM	6TM	Reac Clnt LP3 Cold Leg Stop Valve (3RCS*MV8002C)
Sh. 6TN	6TN	Reac Clnt LP4 Cold Leg Stop Valve (3RCS*MV8002D)
Sh. 6TP	6TP	Reactor Coolant Loop 1 Bypass Leg Stop Valve (3RCS*MV8003A)
Sh. 6TQ	6TQ	Reactor Coolant Loop 2 Bypass Leg Stop Valve (3RCS*MV8003B)
Sh. 6TR	6TR	Reactor Coolant Loop 3 Bypass Leg Stop Valve (3RCS*MV8003C)
Sh. 6TS	6TS	Reactor Coolant Loop 4 Bypass Leg Stop Valve (3RCS*MV8003D)
Sh. 6TW	6TW	Swgr Area & Battery Rooms Supply Fans
Sh. 6TX	6TX	Battery Rooms 1 & 2 Exhaust Fans
Sh. 6TY	6TY	Battery Rooms 3 & 4 Exhaust Fans
Sh. 6TZ	6TZ	Battery Room 5 Exhaust Fan
Sh. 6VK	6VK	Instrument RK & Computer Rooms A/C Unit 3HVC*ACU2A
Sh. 6VL	6VL	Instrument RK & Computer Rooms A/C Unit 3HVC*ACU2B
Sh. 6VM	6VM	Stm Gen Aux Fdwtr Isol Valve [3FWA*MOV35A]
Sh. 6VN	6VN	Stm Gen Aux Fdwtr Isol Valve [3FWA*MOV35B]
Sh. 6VP	6VP	Stm Gen Aux Fdwtr Isol Valve [3FWA*MOV35C]
Sh. 6VQ	6VQ	Stm Gen Aux Fdwtr Isol Valve [3FWA*MOV35D]
Sh. 6VV	6VV	Aux Fdwtr Pump Turb Stm Supply Non-return Valve [3MSS*MOV17A]

TABLE 1.7-1 ELEMENTARY DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-32001	SWEC Drawing Number 12179-ESK-	Diagram Title
Sh. 6VW	6VW	Aux Fdwtr Pump Turb Stm Supply Non-return Valve [3MSS*MOV17B]
Sh. 6VX	6VX	Aux Fdwtr Pump Turb Stm Supply Non-return Valve [3MSS*MOV17D]
Sh. 6VY	6VY	Main Stm Press Rel Isol Valve [3MSS*MOV18A] (AO)
Sh. 6VZ	6VZ	Main Stm Press Rel Isol Valve [3MSS*MOV18B] (BP)
Sh. 6WA	6WA	Main Stm Press Rel Isol Valve [3MSS*MOV18C] (CO)
Sh. 6WB	6WB	Main Stm Press Rel Isol Valve [3MSS*MOV18D] (DP)
Sh. 6ZH	6ZH	EGE MCC Fdr Bkr [3EHS*MCC1A1]
Sh. 6ZJ	6ZJ	Cont Bldg MCC Fdr Bkr [3EHS*MCC1A2]
Sh. 6ZK	6ZK	Aux Bldg MCC Fdr Bkr [3EHS*MCC3A1]
Sh. 6ZL	6ZL	Sfgrds Area MCC Fdr Bkr [3EHS*MCC1A4]
Sh. 6ZM	6ZM	Circ & Serv Wtr Pp Hse MCC Fdr Bkr [3EHS*MCC1A5]
Sh. 6ZN	6ZN	EGE MCC Fdr Bkr [3EHS*MCC1B1]
Sh. 6ZP	6ZP	Cont Bldg MCC Fdr Bkr [3EHS*MCC1B2]
Sh. 6ZQ	6ZQ	Aux Bldg MCC Fdr Bkr [3EHS*MCC3B1]
Sh. 6ZR	6ZR	Sfgrds Area MCC Fdr Bkr [3EHS*MCC1B4]
Sh. 6ZS	6ZS	Circ & Serv Pp Hse MCC Fdr Bkr [3EHS*MCC1B5]
Sh. 6ZT	6ZT	Rod Contmt Area MCC Fdr Bkr [3EHS*MCC2A1]
Sh. 6ZU	6ZU	Rod Contmt Area MCC Fdr Bkr [3EHS*MCC2B1]
Sh. 6ZV	6ZV	Turb Bldg MCC Fdr Bkr [3EHS*MCC2A2]
Sh. 6AAA	6AAA	Contmt Recirc Cir Out VV57A
Sh. 6AAB	6AAB	Contmt Recirc Cir Out VV57B
Sh. 6AAC	6AAC	Contmt Recirc Cir Out VV57C
Sh. AAD	AAD	Contmt Recirc Cir Out VV57D
Sh. 6AAF	6AAF	Contmt Recirc Cir Supply VV54A
Sh. 6AAG	6AAG	Contmt Recirc Cir Supply VV54B

TABLE 1.7-1 ELEMENTARY DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-32001	SWEC Drawing Number 12179-ESK-	Diagram Title
Sh. 6AAH	6AAH	Contmt Recirc Cir Supply VV54C
Sh. 6AAJ	6AAJ	Contmt Recirc Cir Supply VV54D
Sh. 6AAK	6AAK	Rx Plt Compnt Clg Hx Supply VV50A
Sh. 6AAL	6AAL	Rx Plt Compnt Clg Hx Supply VV50B
Sh. 6AAM	6AAM	TPCCW Hx Inlet Valve [3SWP*MOV71A]
Sh. 6AAN	6AAN	TPCCW Hx Inlet Valve [3SWP*MOV71B]
Sh. 6AAU	6AAU	Service Wtr Pmp Disch Valve [3SWP*MOV102A]
Sh. 6AAV	6AAV	Service Wtr Pmp Disch Valve [3SWP*MOV102B]
Sh. 6AAW	6AAW	Service Wtr Pmp Disch Valve [3SWP*MOV102C]
Sh. 6AAX	6AAX	Service Wtr Pmp Disch Valve [3SWP*MOV102D]
Sh. 6AAY	6AAY	SI Pump Cooling
Sh. 6ABG	6ABG	Diesel Gen "A" Enc Ventilation Supply Fan [3HVP*FN1A,C]
Sh. 6ABH	6ABH	Diesel Gen "B" Enc Ventilation Supply Fan [3HVP*FN1B,D]
Sh. 6ABP	6ABP	480 V Motor Cont Emer Gen A&B Misc Equipment Fdr
Sh. 6ABZ	6ABZ	Emer Gen Crankcase Vac Pmp [3EGD*P1A, B]
Sh. 6ACA	6ACA	Emerg Gen "A" Air Compr [3EGA-C1A, C2A]
Sh. 6ACB	6ACB	Emer Gen "B" Air Compr [3EGA-C1B, C2B]
Sh. 6ACL	6ACL	480 V Mc Diesel Gen Enc+ "A" Ventilation Dampers [3HVP*MOD20A, 23A, 26A, 20C]
Sh. 6ACM	6ACM	480 V Mc Diesel Gen Enc+ "B" Ventilation Dampers [3HVP*MOD20B, 23B, 26B, 20D]
Sh. 6ACN	6ACN	Diesel Gen "A" Enc Ventilation Supply Fan [3HVP*FN1C]
Sh. 6ACP	6ACP	Diesel Gen "B" Enc Ventilation Supply Fan [3HVP*FN1D]
Sh. 6ACZ	6ACZ	Service Water Pumphouse Exh Fans & Inlet Dampers 3HVY*FN2A&B; 3HVY*AOD23A&B
Sh. 6ADH	6ADH	Main Steam Valve Bldg Exh Fan 1B
Sh. 6ADR	6ADR	SI, QS & RHR Pps Area A/C Units [3HVQ*ACUS1A,B]

TABLE 1.7-1 ELEMENTARY DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-32001	SWEC Drawing Number 12179-ESK-	Diagram Title
Sh. 6ADS	6ADS	Contmnt Recirc Pp & Clr Area A/C Units [3HVQ*ACUS2A,B]
Sh. 6ADU	6ADU	Aux Fdwtr Pp Area Emer Sply Fans [3HVQ*FN5A,B]
Sh. 6ADV	6ADV	Aux Fdwtr Pp Area Emer Vent Dmprs [3HVQ*MOD26A1, B1,C1]
Sh. 6ADW	6ADW	Aux Fdwtr Pp Area Emer Exh Fans [3HVQ*FN6A,B]
Sh. 6ADX	6ADX	Charging Pp Aux Lube Oil Pp (3CHS*P6C)
Sh. 6AEA	6AEA	Charging Pp Area Inlet & Outlet Air Dampers [3HVR*MOD49A,50A, 49B*50B]
Sh. 6AEB	6AEB	Charging Pp Area Inlet & Outlet Air Dampers [3HVR*MOD49C1, 50C1,49C2,50C2]
Sh. 6AEC	6AEC	Elem Diag 120V ac Aux & Fuel Bldg Vent Dampers [3HVR*MOD28A,28B]
Sh. 6AFA	6AFA	Bus 32T Tie Bk to Bus 32S 32T11-2 [3EJS*ACB-T11]
Sh. 6AFB	6AFB	Bus 32S Tie Bkr to Bus 32R 32S11-2 [3EJS*ACB-S11]
Sh. 6AFC	6AFC	Bus 32U Tie Bkr to Bus 32V 32U11-2 [3EJS*ACB-U11]
Sh. 6AFD	6AFD	Bus 32V Tie Bkr to Bus 32W 32V11-2 [3EJS*ACB-V11]
Sh. 6AFE	6AFE	Aux Bldg Filter Unit [3HVR*FLT1A]
Sh. 6AFF	6AFF	Aux Bldg Filter Unit [3HVR*FLT1B]
Sh. 6AFG	6AFG	Fuel Bldg Filter Unit [3HVR*FLT2A]
Sh. 6AFH	6AFH	Fuel Bldg Filter Unit [HVR*FLT2]
Sh. 6AFK	6AFK	Cont Bldg Chiller Condenser Inlet Temp Cont Valves (3SWP*TV35A&B)
Sh. 6AFL	6AFL	Cntmt Recirc Pp Miniflow Valve [3RSS*MOV38A]
Sh. 6AFM	6AFM	Cntmt Recirc Pp Miniflow Valve [3RSS*MOV38B]
Sh. 6AFZ	6AFZ	Containment Open Pressure Tap Isolation Valve [3LMS*MOV40A]
Sh. 6AGA	6AGA	Containment Open Pressure tap Isolation Valve [3LMS*MOV40B]

TABLE 1.7-1 ELEMENTARY DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-32001	SWEC Drawing Number 12179-ESK-	Diagram Title
Sh. 6AGB	6AGB	Containment Open Pressure Tap Isolation Valve [3LMS*MOV40C]
Sh. 6AGC	6AGC	Containment Open Pressure Tap Isolation Valve [3LMS*MOV40D]
Sh. 6AGD	6AGD	Aux Fd Pps Area Emer Vent Dmprs [3HVQ*MOD26A2, B2, C2]
Sh. 6AGF	6AGF	Circulating Water Valve 3SWP*MOV115A
Sh. 6AGG	6AGG	Circulating Water Pump Lube Water Valve 3SWP*MOV115B
Sh. 6AGH	6AGH	Containment Instr Air Isolation Valve [3IAS*MOV72]
Sh. 6AGX	6AGX	Fuel Bldg Vent Dampers [3HVR*MOD72A,B]
Sh. 6AGY	6AGY	Aux Fdwtr Pp Area Emer Sply Fan
Sh. 6AHB	6AHB	Suppl Leak Coll Rel Sys Flt Htrs
Sh. 6AHL	6AHL	Contmt ATM Monit Inside Contmt Isol Valve [3CMS*MOV24]
Sh. 6AJH	6AJH	Elem Diag 480 V MC Reactor Vessel to Excess Ltdn Valve (3RCS*V8098)
Sh. 6AJJ	6AJJ	Elem Diag 480 V MC Boric Acid Gravity Feed Valve (3CHS*V8507A)
Sh. 6AJK	6AJK	Elem Diag 480 V MC Boric Acid Gravity Feed Valve (3CHS*MV8507B)
Sh. 6AJL	6AJL	Elem Diag 480 V MC LPSI Charging Pp Suct Valve (3CHS*V8468A)
Sh. 6AJM	6AJM	Elem Diag 480 V MC LPSI Charging Pipe Suct Valve (3CHS*MV8468B)
Sh. 6AJN	6AJN	Elem Diag 480 V MC Charging Pp Recirc Valve (3CHS*MV8111B)
Sh. 6AJP	6AJP	Elem Diag 480 V MC Charging Pp Recirc (3CHS*MV8111C)
Sh. 6AJQ	6AJQ	Elem Diag 480 V MC Charging Hdr Isol Valve (3CHS*MV8438A)
Sh. 6AJR	6AJR	Elem Diag 480 V MC Charging Hdr Isol Valve (3CHS*MV8438B)

TABLE 1.7-1 ELEMENTARY DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-32001	SWEC Drawing Number 12179-ESK-	Diagram Title
Sh. 6AJS	6AJS	Elem Diag 480 V MC Charging Hdr Isol Valve (3CHS*MV-8438C)
Sh. 6AJU	6AJU	Elem Diag 480 V MC Contmt Air Recirc Coil Sply Valve (CCP*MOV222)
Sh. 6AJV	6AJV	Elem Diag 480 V MC Contmt Air Recirc Clg Coil Sply Valve (3CCP*MOV223)
Sh. 6AJW	6AJW	Elem Diag 480 V MC Contmt Air Recirc Clg Coil Sply Valve (3CCP*MOV224)
Sh. 6AJX	6AJX	Elem Diag 480 V Contmt Air Recirc Clg Coil Sply Valve (3CCP*MOV225)
Sh. 6AJY	6AJY	Elem Diag 480 V Sply Valve Contmt Air Recirc Clg Coil Sply Valve (3CCP*MOV226)
Sh. 6AJZ	6AJZ	Elem Diag 480 V MC Contmt Air Recirc Clg Coil Sply Valve (3CCP*MOV227)
Sh. 6AKA	6AKA	Elem Diag 480 V MC Contmt Air Recirc Coil Sply Valve (3CCP*MOV228)
Sh. 6AKB	6AKB	Elem Diag 480 V MC Contmt Air Recirc Clg Coil Sply Valve (3CCP*MOV229)
Sh. 6AKC	6AKC	Elem Diag 480 V MC Charging Hdr Isol Valve (3CHSX-MV8116)
Sh. 6AKE	6AKE	Contmt Isol Valve (3CCP*MOV45A)
Sh. 6AKF	6AKF	Contmt Isol Valve (3CCP*MOV45B)
Sh. 6AKG	6AKG	Contmt Isol Valve (3CCP*MOV48A)
Sh. 6AKH	6AKH	Contmt Isol Valve (3CCP*MOV48B)
Sh. 6AKJ	6AKJ	Contmt Isol Valve (3CCP*MOV49A)
Sh. 6AKK	6AKK	Contmt Isol Valve (3CCP*MOV49B)
Sh. 6AKL	6AKL	Elem Diag 480 V MC Mn Stm Press Relief Cntrl Bypass Valve 3MSS*MOV74A
Sh. 6AKM	6AKM	Elem Diag 480 V MC Mn Stm Press Relief Cntrl Bypass Valve 3MSS*MOV74B

TABLE 1.7-1 ELEMENTARY DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-32001	SWEC Drawing Number 12179-ESK-	Diagram Title
Sh. 6AKN	6AKN	Elem Diag 480 V MC Mn Stm Press Relief Cntrl Bypass Valve 3MSS*MOV74C
Sh. 6AKP	6AKP	Elem Diag 480 V MC Mn Stm Press Relief Cntrl Bypass Valve 3MSS*MOV74D
Sh. 6AKX	6AKX	Elem Diag 480 V ESF Bldg MCC Fdr Bkr (3EHS*MCC1B3)
Sh. 6AKY	6AKY	Elem Diag 480 V Emer Supply Bkr (3EJS*ACB-AD)
Sh. 6AKZ	6AKZ	Elem Diag 480 V Emer Supply Bkr [3EJS*ACB-BD]
Sh. 6ALA	6ALA	Elem Diag 480 V Bus 32Y Tie to Bus 32R 32T 4A-2 [3EJS*ACB T4A]
Sh. 6ALB	6ALB	Elem Diag 480 V Bus 32X Tie Bkr to Bus 32W 32T4B-2 [3ETS*ACB-T4B]
Sh. 6ALC	6ALC	Elem Diag 480 V US Spare MCC Fdr Bkr [3EHS*MCC4A1]
Sh. 6ALD	6ALD	Elem Diag 480 V US Spare MCC Fdr Bkr [3EHS*Bkr MCC4B1]
Sh. 6ALG	6ALG	Elem Diag 480 V MC (3SWP-3A & B)
Sh. 6ALH	6ALH	Elem Diag 480 V MC (3STR 2A & B)
Sh. 6AMB	6AMB	Elem Diag 480 V MC Cold Shutdown Air Compressor [3IAS-C2A]
Sh. 6AMC	6AMC	Elem Diag 480 V MC Cold Shutdown Air Compressor [3IAS-C2B]
Sh. 6AMG	6AMG	Mn Stm Valve Bldg Vent Exh Fan Dampers
Sh. 6AMH	6AMH	Mn Stm Valve Bldg Vent Exh Fan Dampers
Sh. 6AMJ	6AMJ	Mn Stm Valve Bldg Vent Inlet Dampers
Sh. 6AMK	6AMK	Mn Stm Valve Bldg Vent Inlet Dampers
Sh. 6BAA	6BAA	Air Sample Pump
Sh. 6BAB	6BAB	Elem Diag 480 V MC Charging Pp Miniflow Cntrl Valve [3CHS*MV8512A]
Sh. 6BAC	6BAC	Elem Diag 480 V MC Charging Pp Miniflow Cntrl Valve [3CHS*MV8512B]

TABLE 1.7-1 ELEMENTARY DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-32001	SWEC Drawing Number 12179-ESK-	Diagram Title
Sh. 6BAD	6BAD	Elem Diag 480 V MC Charging Pp Miniflow Cntrl Valve [3CHS*MV8511A]
Sh. 6BAE	6BAE	Elem Diag 480 V MC Charging Pp Miniflow Cntrl Valve [3CHS*MV8511B]
Sh. 6BAF	6BAF	Elem Diag 480 V Fuel Bldg Fltr Exh Fan [3HVR*FN10A2]
Sh. 6BAG	6BAG	Elem Diag 480 V Fuel Bldg Fltr Exh Fan [3HVR*FN10B2]
Sh. 6BAN	6BAN	Elem Diag. Flow to Monitor for Pzr SVs
Sh. 6BAW	6BAW	CRDM Fan 3-HVU-FN2A PENSEC PROT Bkr
Sh. 6BAX	6BAX	CRDM Fan 3-HVU-FN2B PENSEC PROT Bkr
Sh. 6BAZ	6BAZ	Cont Struc Air Recirc FN 3-HVU-FN1A
Sh. 6BBA	6BBA	Cont Struc Air Recirc FN 3-HVU-FN1B PENE Sec Prot Bkr
Sh. 6BBE	6BBE	120 V DC Halon Disch Timer
Sh. 7J	7J	4.16 kV Bus 34C [3NNS-SWG-A] Aux Ckt
Sh. 7L	7L	4.16 kV Bus 34D [3ENS*SWG-B] Aux Ckt
Sh. 7Q	7Q	Emer Diesel Gen Bkr [3ENS*ACB-G-A] Aux Circuit 15G-14U-2
Sh. 7R	7R	Emer Diesel Gen Bkr [3ENS*ACB-G-B] Aux Circuit 15G-15U-2
Sh. 7W	7W	Emerg Diesel Clr Outlet Valves [3SWP*AOV39A&B]
Sh. 7AD	7AD	Stm Jet Air Ejector Stm Valves [3ASS-AOV22A,B]
Sh. 7AJ	7AJ	Aux Fdwtr Alt Suct Valves [3FWA*AOV23A,23B]
Sh. 7AL	7AL	Dwst Htr Circ Line Isol Valve [3FWA*AOV25, 26]
Sh. 7AM	7AM	Refuel Wtr Recirc Pp Suct Isol Valve (3QSS*AOV27, 28)
Sh. 7AN	7AN	Accumulator Nit Test Line Isol SIP Hot Leg Test Line [3SIL*CV8880] [SIL*CV8825]
Sh. 7AP	7AP	Test Line Header [3SIH*CV8964] & [3SIH*8871]
Sh. 7AQ	7AQ	Accumulator Gas Sply & Vent [3SIL*AV8875A]
		Accumulator Fill Line Isol [3SIL*AV8878A]

TABLE 1.7-1 ELEMENTARY DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-32001	SWEC Drawing Number 12179-ESK-	Diagram Title
Sh. 7AR	7AR	Accumulator Gas Sply & Vent [3SIL*AV8875B]
		Accumulator Fill Line Isol [3SIL*AV8878B]
Sh. 7AS	7AS	Accumulator Gas Sply & Vent [3SIL*AV8875A]
		Accumulator Fill Line Isol [3SIL*AV8878C]
Sh. 7AT	7AT	Accumulator Gas Sply & Vent [3SIL*AV8875D]
		Accumulator Fill Line Isol [3SIL*AV8878D]
Sh. 7AU	7AU	RHR Pmp Hot Leg Test Line [3SIL*CV8890A, CV8890B]
Sh. 7AY	7AY	Mn Stm Isol Trip Valve [3MSS*CTV27A]
Sh. 7AZ	7AZ	Mn Stm Isol Trip Valve [3MSS*CTV27B]
Sh. 7BA	7BA	Mn Stm Isol Trip Valve [3MSS*CTV27C]
Sh. 7BB	7BB	Mn Stm Isol Trip Valve [3MSS*CTV27D]
Sh. 7BC	7BC	Mn Stm Line Isol Bypass Valve [3MSS*HV28A]
Sh. 7BD	7BD	Mn Stm Line Isol Bypass Valve [3MSS*HV28B]
Sh. 7BE	7BE	Mn Stm Line Isol Bypass Valve [3MSS*HV28C]
Sh. 7BF	7BF	Mn Stm Line Isol Bypass Valve [3MSS*HV28D]
Sh. 7BG	7BG	SIP Hot Leg Test Line [3SIH*CV8824] & [3SIH* CV8881]
Sh. 7BH	7BH	Stm Gen Blwdn Cntrl Isol Valve [3BDG*CTV22A]
Sh. 7BJ	7BJ	Stm Gen Blwdn Cntrl Isol Valve [3BDG*CTV22B]
Sh. 7BK	7BK	Stm Gen Blwdn Cntrl Isol Valve [3BDG*CTV22C]
Sh. 7BL	7BL	Stm Gen Blwdn Cntrl Isol Valve [3BDG*CTV22D]
Sh. 7BP	7BP	Rx Pl Gaseous Drs Inside Isol Valve [3DGS*CTV24]
Sh. 7BQ	7BQ	Rx Pl Gaseous Drs Outside Cntmt Isol Valve [3DGS*CTV25]
Sh. 7BR	7BR	Cntmt Drs Isol Valves [3DAS**CTV24] & [3DAS*CTV25]
Sh. 7BT	7BT	Cntmt Drs Isol Valve [3DAS*CTV25]
Sh. 7CM	7CM	Chg to Rx Cool Sys Isol Valves [3CHS*AV8147, 8146]
Sh. 7CS	7CS	Rx Coolant Makeup Cont Aux, Circ, RPS-RAK Aux A,C

TABLE 1.7-1 ELEMENTARY DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-32001	SWEC Drawing Number 12179-ESK-	Diagram Title
Sh. 7CT	7CT	BA Inj Valve to BA Blender VV 110A BA Mkup Inj Valve to Charging Pp Hdr VV 110B
Sh. 7CU	7CU	Ba Mkup Wtr Inj Valve to BA Blender VV 111A BA Dil Inj Value to Vol Cntrl Tk VV 111B
Sh. 7CW	7CW	Vol Cntrl Tk Isol & Inert Valves [3CHS*AV8101, LCV112A] Elem Diag 480 V
Sh. 7CY	7CY	Emerg Bux 32R, S, T, Y Aux Ckt Elem Diag
Sh. 7CZ	7CZ	480 V Emerg Bus Bus 32U, V, W, X Aux Ckts
Sh. 7DF	7DF	Cntmt Purge Norm & Fltr Exh Dmprs
Sh. 7DG	7DG	Chg Pp & CCW Norm & Fltr Exh Dmprs
Sh. 7DH	7DH	Waste Dspl & Pipe Tunnel Norm & Fltr Exh Dmprs
Sh. 7DJ	7DJ	Aux Bldg Norm & Fltr Exh Dmprs
Sh. 7DK	7DK	Aux Bldg Norm & Fltr Exh Dmprs
Sh. 7DL	7DL	Fuel Bldg Fltr & Bypass Dmprs
Sh. 7DV	7DV	Reac Coolant Letdn Drn Valves & Excess Letdn Isol Valves [3RCS*AV8032, 8153]
Sh. 7DW	7DW	Pressurizer Power Power Relief Valves [3RCS*PCV455A, 456]
Sh. 7DW1	7DW1	Pressurizer Pwr Relief U/V 3RCS*456
Sh. 7DX	7DX	Letdown Line Isol Valves [3RCS*LCV459, 460]
Sh. 7DZ	7DZ	Reactor Coolant System Spray VV 8145 Pressurizer Relief Tank Press VV 469
Sh. 7EK	7EK	Chg Pump Test Line Isol [3SIH*CV8843] & [3SIH*AV8882]
Sh. 7EM	7EM	Accumulator Fill Line 3SIH*CV8888 Skip Cold Leg Test Line 3SIH*CV8823
Sh. 7EZ	7EZ	Pri Gr Wtr Cntmt Isol Valves [3PGS*CV8026, 8046]
Sh. 7FG	7FG	4.16 kV Bus [3ENS*SWG-A] Aux Circuit
Sh. 7FH	7FH	4.16 kV Bus [3ENS*SWG-B] Aux Circuit
Sh. 7FP	7FP	Main Steam Line Drains [3DTM*AOV63, 64]

TABLE 1.7-1 ELEMENTARY DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-32001	SWEC Drawing Number 12179-ESK-	Diagram Title
Sh. 7FS	7FS	Main Steam Line Drains [3DTM*AOV29A,B]
Sh. 7FT	7FT	Main Steam Line Drains [3DTM*AOV29C, D]
Sh. 7FU	7FU	Main Steam Line Drains [3DTM*AOV61A, B]
Sh. 7FV	7FV	Main Steam Line Drains [3DTM*AOV61C, D]
Sh. 7HK	7HK	SG Fdwtr Bypass Valve & Cntrl Valves Auxiliary Ckt [3FWS*LV550]
Sh. 7HL	7HL	SG Fdwtr Cntrl Valve Bypass Valve [3FWS*LV560]
Sh. 7HM	7HM	SG Fdwtr Cntrl Valve Bypass Valve [3FWS*LV570]
Sh. 7HN	7HN	SG Fdwtr Cntrl Valve Bypass Valve [3FWS*LV580]
Sh. 7HP	7HP	SG Mn Fdwtr Flow Cntrl Valve [3FWS-FCV510]
Sh. 7HQ	7HQ	SG Mn Fdwtr Flow Cntrl Valve [3FWS-FCV520]
Sh. 7HR	7HR	SG Mn Fdwtr Flow Cntrl Valve (3FWS*FCV530)
Sh. 7HS	7HS	SG Mn Fdwtr Flow Cntrl Valve 3FWS*FCV540
Sh. 7HX	7HX	Turb Bypass Cntrl Sys Aux Ckt
Sh. 7JC	7JC	Letdown Orifice Isol Valve [3CHS*AV8149A]
Sh. 7JD	7JD	Letdown Orifice Isol Valve [3CHS*AV8149B]
Sh. 7JE	7JE	Letdown Orifice Isol Valve [3CHS*AV8149C]
Sh. 7JK	7JK	Chg Pps Clrs Out Crossover Valves [3CCE*AOV26A,B]
Sh. 7JL	7JL	Chg Pmps Disch Crossover Valves [3CCE*AOV30A,B]
Sh. 7JN	7JN	SG Fdwtr Isol Valve [3FWS*CTV41A]
Sh. 7JP	7JP	SG Fdwtr Isol Valve [FWS*CTV41B]
Sh. 7JQ	7JQ	SG Fdwtr Isol Valve [3FWS*CTV41C]
Sh. 7JR	7JR	SG Fdwtr Isol Valve [3FWS*CTV41D]
Sh. 7JY	7JY	Cntmt Inst Air Isol Valve [3IAS*CTV15]
Sh. 7KJ	7KJ	120 VAC Cont Bldg Aux Ckts
Sh. 7KK	7KK	CVS Isol Valves (3CVS*CTV20A,B)
Sh. 7KL	7KL	CVS Isol Valves (3CVS*CTV21A,B)

TABLE 1.7-1 ELEMENTARY DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-32001	SWEC Drawing Number 12179-ESK-	Diagram Title
Sh. 7KQ	7KQ	Serv Wtr Pps Train A & B Aux Ckts
Sh. 7KT	7KT	SG #1 Blwdwn Sample Isol VV19A
Sh. 7KU	7KU	SG #2 Blwdwn Isol VV19B
Sh. 7KV	7KV	SG #3 Blwdwn Isol VV19C
Sh. 7KW	7KW	SG #4 Blwdwn Sample Isol VV19D
Sh. 7KX	7KX	Pzr Vapor Space Sample Isol VVs 20&21
Sh. 7KZ	7KZ	Rx Clnt Hot Leg Sample Isol VVs 26&27
Sh. 7LA	7LA	Rx Clnt Cold Leg Sample Isol VVs 29&30
Sh. 7LB	7LB	SI Inj Accumulator Sample Isol VVs 32&33 Aux Circuit
Sh. 7LC	7LC	Pzr Rlf Tk Gas Space Sample Isol Valves (3SSR*CV8026, 8025)
Sh. 7LX	7LX	Non-Sfty Hdr Sply Rtn Valves [3CCP*AOV10A, B,&19A,B]
Sh. 7MA	7MA	Cntmt Isol Valves [3CCP*CTV49A,B]
Sh. 7MB	7MB	Residual Heat Removal Outlet Valves [3CCP*AOV66A&B]
Sh. 7MC	7MC	Thermal Barrier Coolant Return Valves [3CCP*AOV178A&B]
Sh. 7MD	7MD	Thermal Barrier Coolant Return Valves [3CCP*AOV178C&D]
Sh. 7ME	7ME	Comp Cooling Cross Conn VVs 179A&B
Sh. 7MF	7MF	Comp Cooling Cross Conn VVs 180A&B
Sh. 7MG	7MG	Non-Sfty Hdr Sply & Rtn Values [3CCP*AOV194A, B, & 197A,B]
Sh. 7MW	7MW	Ctmt Spray Test Inter Circuitary
Sh. 7MX	7MX	Diesel Gen "B" Enclosure Vent Outlet Damper (3HVP*MOD20F,H)
Sh. 7MY	7MY	Diesel Gen "B" Enclosure Vent Outlet Damper (3HVP*MOD20F,H)
Sh. 7MZ	7MZ	Diesel Gen "B" Enclosure Vent Outlet Damper (3HVP*MOD20F,H)
Sh. 7NA	7NA	125V Hx Temp Cont VIV 3CCP*SOU32A & 32C-1

TABLE 1.7-1 ELEMENTARY DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-32001	SWEC Drawing Number 12179-ESK-	Diagram Title
Sh. 7NB	7NB	125V Hx Temp Cont VIV 3CCP*SOU32B & 32C-2
Sh. 7PB	7PB	Control Room Ventilation Outlet Air Isol VVs 20,21 [HVA*AOV20,21]
Sh. 7PC	7PC	Control Room Ventilation Outlet Air Isol VVs 22,23 [HVC*AOV22,23]
Sh. 7PD	7PD	Control Room Ventilation Outlet Air Isol VVs 25,26 [HVC*AOV25,26]
Sh. 7PJ	7PJ	Purge Supply Fan Inlet Damper AOD135 Emerg Vent Air Return Damper AOD119
Sh. 7PK	7PK	Cntrl Bldg Makeup Air Dmprs, (3HVC*AOD 27A,B)
Sh. 7PL	7PL	Air Stor Tk 1A Outlet VVs SOV 74A & 74B
Sh. 7QD	7QD	Turbine-Driven Aux Fdwtr Pp Aux Oil Pp (3FWL*PS), Trip & Throttle Valve (3MSS*M5V5)
Sh. 7QE	7QE	Mn Stm Rel & Drain Valves [3MSS*PV20A, 3SVV-AOV20A]
Sh. 7QF	7QF	Mn Stm Rel & Drain Valves [3MSS*PV20B, 3SVV-AOV20B]
Sh. 7QG	7QG	Mn Stm Rel & Drain Valves [3MSS*PV20C, 3SVV-AOV20C]
Sh. 7QH	7QH	Mn Stm Rel & Drain Valves [3MSS*PV20D, 3SVV-AOV20D]
Sh. 7QM	7QM	Chilled Wtr Cntmt Isol Valves [3CDS*CTV38A,B]
Sh. 7QN	7QN	Chilled Wtr Cntmt Isol Valves [3CDS*CTV39A,B]
Sh. 7QP	7QP	Chilled Wtr Cntmt Isol Valves [3CDS*CTV40A,B]
Sh. 7QT	7QT	Reactor Coolant Letdown Valves [3CHS*TCV12G, AV8143]
Sh. 7QU	7QU	Letdown Line Isol Valves [3CHS*CV81528160]
Sh. 7QV	7QV	Aux Bldg Inlet Dampers [3HVR*AOD33A, 35A]
Sh. 7QW	7QW	Aux Bldg Inlet Dampers [3HVR*AOD33B, 35B]
Sh. 7QX	7QX	Pipe Tunnel Inlet Dampers [3HVR*AOD85,86]
Sh. 7QY	7QY	Cntmt Purge Inlet Dampers [3HVR*AOD55A/B & 174A/B]
Sh. 7QZ	7QZ	Chg & Compnt Clg Pps Temp Cntrl Dmprs [3HVR*AOD45B1, B2, C1, C2]

TABLE 1.7-1 ELEMENTARY DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-32001	SWEC Drawing Number 12179-ESK-	Diagram Title
Sh. 7RB	7RB	Reac Clnt Loop Stop Valve Aux Buffering Circuit
Sh. 7RC	7RC	Reac Clnt Loop Stop Valve Aux Buffering Circuit
Sh. 7RF	7RF	TDAFW Pmp Mtr Speed changr 3FWA*M7
Sh. 7RG	7RG	ESF Bldg Vent Dampers
Sh. 7RH	7RH	ESF Bldg Vent Dampers
Sh. 7RK	7RK	Containment Purge Inlet & Outlet Isol Valves [3HVU*CTV32A&B], [3HVU*CTV33A&B]
Sh. 7RM	7RM	Containment Atmosphere Monitoring
Sh. 7RZ	7RZ	SG Level Aux Ckts: Mn Turb Trip
Sh. 7SC	7SC	SG Chemical Feed Pp Isolation Valves [3SGF*AOV24A&B]
Sh. 7SD	7SD	SG Chemical Feed Pp Isol Valve [3SGF*AOV24C&D]
Sh. 7SF	7SF	Misc Level Ind Lights DAS System
Sh. 7SH	7SH	Rx Pl Gas Vents Hdr Cntrl Trip Valves (3VRS*CTV20,21)
Sh. 7SM	7SM	Gas Waste to Unit 1 Stack Isol Dmprs (3GWS*AOD78A,B)
Sh. 7SP	7SP	Pzr Relief Tk Nitrogen Sply Isol Valves (3GSN*CV8033, CTV105)
Sh. 7SS	7SS	ESF Manual Actuation Ckts
Sh. 7ST	7ST	ESF Manual Reset Ckts
Sh. 7SX	7SX	4.16 kV Bus 34C (3ENS*SWG-A) Undervoltage (Hi STPT) Trip Ckt
Sh. 7SY	7SY	4.16 kV Bus 34D (3ENS*SWG-B) Undervoltage (Hi STPT) Trip Ckt
Sh. 7SZ	7SZ	4.16 kV Bus 34C&D (3ENS*SWG-A&B) Undervoltage (Hi STPT) Relays
Sh. 7TA	7TA	Cntrl Bldg Isol (Train A) Sh. 1
Sh. 7TB	7TB	Cntrl Bldg Isol (Train A) Sh. 2
Sh. 7TC	7TC	Cntrl Bldg Isol (Train B) Sh. 1
Sh. 7TD	7TD	Cntrl Bldg Isol (Train B) Sh. 2

TABLE 1.7-1 ELEMENTARY DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-32001	SWEC Drawing Number 12179-ESK-	Diagram Title
Sh. 7TN	7TN	Containment Fire Protection Water Isolation Valves 3FPW & CTV 48,49
Sh. 7TP	7TP	ESF Aux Relays (Train A)
Sh. 7TQ	7TQ	ESF Aux Relays (Train A)
Sh. 7TR	7TR	ESF Aux Relays (Train B)
Sh. 7TS	7TS	ESF Aux Relays (Train B)
Sh. 7TY	7TY	Accumulator N2 Line Isol Valve 3SIL*CVS968
Sh. 7TZ	7TZ	Chilled Wtr Cntmt Isol Valves [3CDS*CTV91A,B]
Sh. 7UF	7UF	Auto and Manual Rod Withdrawal Block
Sh. 7UK	7UK	Accumulator Gas Sply & Vent Valves (3SIL*SV8875F&G)
Sh. 7UL	7UL	Accumulator Gas Sply & Vent Valves (3SIL*SV8875F&G)
Sh. 7UM	7UM	Rx Vessel Head Vent Isol Valve (3RCS*SV8095A1B)
Sh. 7UN	7UN	Rx Vessel Head Vent Isol Valve (3RCS*SV0896A1B)
Sh. 7UR	7UR	Elem Diag Water Feed to Chlorination System [3WTC*AOV25A&B]
Sh. 7UT	7UT	Elem Diag SI Accumulator Tk Vent Valves
Sh. 7UU	7UU	Elem Diag ESF Manual Reset Caskets
Sh. 7UV	7UV	Elem Diag Reactor Reactor Vessel Head Vent Valves
Sh. 7UW	7UW	Elem Diag Mn Stm Isol Trip Valve Aux Ckt (3MSS*CTV27A)
Sh. 7UX	7UX	Elem Diag Mn Stm Trip Valve Aux Ckt (3MSS*CTV27B)
Sh. 7UY	7UY	Elem Diag Mn Stm Isol Trip Valve Aux Ckt (3MSS*CTV27C)
Sh. 7UZ	7UZ	Elem Diag Mn Stm Mn Stm Isol Trip Valve Aux Ckt (3MSS*CTV27D)
Sh. 7VA	7VA	Analog Isolation Ckts
Sh. 7VB	7VB	Analog Isolation Ckts
Sh. 7VC	7VC	Elem Diag Change Hdr Flow Valves
Sh. 7VD	7VD	Elem Diag (3DTM*AOV63B, 64B) Main Steam Line Drains

TABLE 1.7-1 ELEMENTARY DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-32001	SWEC Drawing Number 12179-ESK-	Diagram Title
Sh. 7VE	7VE	Elem Diag (3DTM*AOV64B, 64C) Main Steam Line Drains
Sh. 7VG	7VG	Elem Diag ASP Lamp Test Aux Ckts
Sh. 7VH	7VH	Elem Diag ASP Lamp Test Aux Ckts Sequencer Status Lights
Sh. 7VJ	7VJ	Elem Diag 125 V dc Cntmt Purge Inlet Dampers (3HVR*AOD55B & 174B)
Sh. 7VK	7VK	Elem Diag 125 V dc 125 V dc Digital Isolator Circuits
Sh. 7VM	7VM	Elem Diag 120 V ac Main Vent & Air Cond Panel Lamp Test Ckt, Train "A"
Sh. 7VN	7VN	Elem Diag 120 V ac Main Vent & Air Cond Panel Lamp Test Ckt, Train "B"
Sh. 7VQ	7VQ	Elem Diag 120 V ac Main Vent & Air
Sh. 7VX	7VX	DC Digital Isol Circ 3BYS-PNL-5 & 6
Sh. 7VZ	7VZ	Elem Diag Digital Isolator Circuits
Sh. 7WB	7WB	Elem Diag 120 V ac MCB Lamp Test Ckt
Sh. 7WC	7WC	Elem Diag 120 V ac MCB Lamp Test Ckt
Sh. 7WD	7WD	Elem Diag 120 V ac MCB Lamp Test Ckt
Sh. 7WF	7WF	Elem Diag 120 V ac MCB Lamp Test Ckt
Sh. 7WG	7WG	Elem Diag 120 V ac MCB Lamp Test Ckt
Sh. 7WJ	7WJ	Elem Diag 120 V ac MCB Lamp Test Ckt
Sh. 7WL	7WL	Elem Diag 120 V ac MCB Lamp Test Ckt
Sh. 7WM	7WM	Elem Diag 120 V ac MB4
Sh. 7WP	7WP	Elem Diag 120 V ac MB3
Sh. 7WR	7WR	Elem Diag 120 V ac MB3
Sh. 7WS	7WS	Elem Diag 120 V ac MB2
Sh. 7WU	7WU	Elem Diag 120 V ac MB2
Sh. 7WV	7WV	Elem Diag 120 V ac MB1
Sh. 7WX	7WX	Elem Diag 120 V ac MB1

TABLE 1.7-1 ELEMENTARY DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-32001	SWEC Drawing Number 12179-ESK-	Diagram Title
Sh. 7XA	7XA	Aux Fdwtr Control Valves
Sh. 7XB	7XB	Aux Fdwtr Control Valves
Sh. 7XC	7XC	Aux Fdwtr Control Valves
Sh. 7XD	7XD	Aux Fdwtr Control Valves
Sh. 7XJ	7XJ	Elem Diag 125 V dc Chilled Water Diff Press Cntrl Valve [3HVK*POV32A, 32B]
Sh. 7XK	7XK	Elem Diag 125 V dc Inst Rack & Comp Rms A/C Unit Temp Cntrl Valve
Sh. 7XL	7XL	Elem Diag 125 V dc Cntrl Rm A/C Unit Temp Cntrl Valve [3HVK*TV41A, 41B]
Sh. 7XM	7XM	Elem Diag 125 V dc E&W Swgr Rms A/C Units Temp Cntrl Valve [3HVK*TV76A, 76B]
Sh. 7XN	7XN	Elem Diag 125 V dc E&W Swgr Rms Backup A/C Units Temp Cntrl Valve [3HVK*TV77A, 77B]
Sh. 7XP	7XP	Chg Pps Clg Valves
Sh. 7XQ	7XQ	Elem Diag (3HVZ*MOD20A, 21A)
Sh. 7XR	7XR	Elem Diag (3HVZ*MOD20B, 21B)
Sh. 7XV	7XV	Elem Diag 120 V ac MB4
Sh. 7XW	7XW	Elem Diag 120 V ac MCB Lamp Test Ckt
Sh. 7XZ	7XZ	Elem Diag CDA Signal Reset Aux Ckt
Sh. 7AAG	7AAG	Elem Diag Control Bldg Chilled Wtr Isol Valves
Sh. 7AAH	7AAH	Elem Diag Control Bldg Chilled Wtr Isol Valves
Sh. 7AAJ	7AAJ	Elem Diag Control Bldg Chilled Wtr Isol Valves
Sh. 7AAK	7AAK	Elem Diag Control Bldg Chilled Wtr Isol Valves
Sh. 7AAP	7AAP	Aux Fdwtr Cntrl Valves [3FWA*HV31A, 32A & 36B]
Sh. 7AAQ	7AAQ	Aux Fdwtr Cntrl Valves [3FWA*HV31B, 32B & 36A]
Sh. 7AAR	7AAR	Aux Fdwtr Cntrl Valves [3FWA*HV31C, 32C &36D]
Sh. 7AAS	7AAS	Aux Fdwtr Cntrl Valves [3FWA*HV31D, 32D & 36C]

TABLE 1.7-1 ELEMENTARY DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-32001	SWEC Drawing Number 12179-ESK-	Diagram Title
Sh. 7ABC	7ABC	Elem Diag Prst Accid Simpl VVS 3SSP*SOV1A, 1B
Sh. 7ABD	7ABD	Elem Diag Prst Accid Simpl VVS 3SSP*SOV1C, 1D
Sh. 7ABE	7ABE	Elem Diag Prst Accid Simpl VVS 3SSP*SOV2A, 2B
Sh. 7ABF	7ABF	Elem Diag Prst Accid Simpl VVS 3SSP*SOV3, 5
Sh. 7ABG	7ABG	Elem Diag Post-Accident Sample Valves 3SSP*CTV7,8
Sh. 7ABP	7ABP	Elem Diag Cntmt Recirc Isol Valves 3SSP*SOV25A,25B
Sh. 7ABU	7ABU	Elem Diag Rx Trip on Turb Trip Isol
Sh. 7ABV	7ABV	Elem Diag Rx Trip on Turb Trip Inputs
Sh. 7ABZ	7ABZ	Aux Stm to Aux Bldg Isol Valves [3ASS-AOV102A & B]
Sh. 7ABZ1	7ABZ1	Aux Stm to Aux Bldg Temp Switches
Sh. 7ACA	7ACA	Aux Bldg Hot Wtr Htg Sys Inlet & Outlet Valves
Sh. 7ACB	7ACB	Aux Bldg Hot Wtr Htg. Sys Inlet & Outlet Valves
Sh. 7ACD	7ACD	Elem Diag 125 V dc Hot Wtr Preheating Sply & Return Isol
Sh. 7ACE	7ACE	Elem Diag 125 V dc Hot Wtr Preheating Sply & Return Isol
Sh. 7ACF	7ACF	Fuel Bldg Air Suply Isol Damper 3HVR*184
Sh. 7ACG	7ACG	Dig Isol CK & for Fuel Bldg A/S Isol Dmpr
Sh. 7ACL	7ACL	Elem Diag 125 V DC Demin Wtr Stor TK to Aux Fd Wtr Pmp Suc 3FWA*AOV61A, 61B
Sh. 7ACM	7ACM	Aux Fd Wtr Pmp Disch Crossover 3FWA*ALU62A, 62B
Sh. 7ACP	7ACP	Chrg Pmp Comp Cooling Pmp Cubicle to Aux Bldg Flt Damper 3HVR*MOD 46A, 46B
Sh. 7ACQ	7ACQ	Aux Bldg Exh Fn Var Inlet Valve 3HVR*MOD 140A, 140B
Sh. 7ACW	7ACW	Mn. Stm Bldg Exh Fn Dampers 3HVV*AOD50A2, 50B2
Sh. 7ACY	7ACY	Tr A Reset for low-low Stm Gen Level on Aux Fd Start
Sh. 7ACZ	7ACZ	Tr B Reset for low-low Stm Gen Level on Aux Fd Start
Sh. 7ADA	7ADA	RHR Hx Flow Control Safety Grade Cold Shutdown
Sh. 8BA	8BA	Res Sta Svce X Fmr A Prot (3RTX-XSR-A) three line

TABLE 1.7-1 ELEMENTARY DIAGRAMS (CONTINUED)

(Refer to Plant Document Control for latest Document Rev. and Date)

NUSCO Drawing Number 25212-32001	SWEC Drawing Number 12179-ESK-	Diagram Title
Sh. 8DA	8DA	Synchronizing
Sh. 8DB	8DB	Synchronizing
Sh. 8DC	8DC	Synchronizing
Sh. 8HC	8HC	RSV Sta Svce Bckup Prot Transf Trip RCV CK 3SPRN03
Sh. 8HG	8HG	Rsv Sta Svce Pri Prot Xfmr Trip Rcvr
Sh. 8JB	8JB	Norm Sta Svce Xfmr Bu Prot
Sh. 8JC	8JC	Norm Sta Svce Xfmr Bu Prot
Sh. 8JD	8JD	Rsv Sta Svce Xfmr Pri Prot
Sh. 8JF	8JF	Rsv Sta Svce Xfmr Pri Prot
Sh. 8KA	8KA	Emer Diesel Gen Excita [3EGS*G-A]
Sh. 8KB	8KB	Emer Diesel Gen Excita [3EGS*G-B]
Sh. 8KC	8KC	Emer Diesel Gen Engine Control [3EGS*G-A]
Sh. 8KD	8KD	Emer Diesel Gen Engine Control [3EGS*G-A]
Sh. 8KE	8KE	Emer Gen "A" Governor [3EGS*EG-A] 15G-14U
Sh. 8KF	8KF	Emer Diesel Gen Engine Control [3EGS*G-B]
Sh. 8KG	8KG	Emer Diesel Gen Engine Control [3EGS*G-B]
Sh. 8KH	8KH	Emer Gen Governor [3EGS*EG-B] 15G-15U
Sh. 8KJ	8KJ	Elem Diag Emerg Diesel Gen 3EGS*EG-A, Shutdown Ckt
Sh. 8KK	8KK	Elem Diag Emerg Diesel Gen 3EGS*EG-B, Shutdown Ckt
Sh. 8KL	8KL	Alt AC DG 125V DC Hyd Govnr Cntr 3BGS-BG-A
Sh. 11A	11A	Reactor Trip Breaker [3RPS*ACB-RTA]
Sh. 11B	11B	Reactor Trip Breaker [3RPS*ACB-RTB]
Sh. 11C	11C	Reactor Trip Bypass Bkr [3RPS*ACB-BYA]
Sh. 11D	11D	Reactor Trip Bypass Bkr [3RPS*ACB-BYB]
Sh. 11H	11H	Nuclear Inst Sys

TABLE 1.7-2 PIPING AND INSTRUMENTATION DIAGRAMS

(See Summary Table of Contents for a Complete List of NUSCO Numbers and Sheets for each FSAR Figure or refer to Plant Document Control for Latest Document Rev and Date)

FSAR Figure Number	Piping and Instrumentation Diagram Title	NUSCO Number
1.2–3	Legend: Piping and Instrumentation Diagram	26900
5.1–1	Reactor Coolant System	26902
5.4.5	Low Pressure Safety Injection System	26912
6.2–36	Quench Spray and Hydrogen Recombiner System	26915
6.2–37	Low Pressure Safety Injection System	26912
6.2–53	Containment Monitoring System	26954
6.3–2	High Pressure Safety Injection System	26913
9.1–6	Fuel Pool Cooling and Purification System	26911
9.2–1	Service Water System	26933
9.2–2	Reactor Plant Component Cooling System	26921
9.2–3	Reactor Plant Chilled Water System	26922
9.2–4	Safety Injection Pump and Neutron Shielding Tank Cooling System	26914
9.2–5	Charging Pump Sealing and Lubrication System	26905
9.2–6	Condensate Demineralizer Liquid Waste System	26929
9.2–7	Water Treatment System	26920
9.2–8	Domestic Water and Sanitary System	26947
9.2–9	Condensate System	26926
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9.3–2	Reactor Plant Sampling System	26944
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TABLE 1.7-2 PIPING AND INSTRUMENTATION DIAGRAMS (CONTINUED)

(See Summary Table of Contents for a Complete List of NUSCO Numbers and Sheets for each FSAR Figure or refer to Plant Document Control for Latest Document Rev and Date)

FSAR Figure Number	Piping and Instrumentation Diagram Title	NUSCO Number
9.3–6	Radioactive Liquid Waste and Aerated Drains System	26906
9.3–7	Reactor Coolant Pump Seals System	26903
9.3–8	Chemical and Volume Control System	26904
9.3–9	Boron Recovery System	26908
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		26908
9.3–10	Post Accident Sample Sheet	26955
9.4–1	Control Building Heating, Ventilation, and Air-Conditioning System	26951
9.4–2	Reactor Plant Ventilation System	26948
9.4–3	Turbine Plant Ventilation System	26950
9.4–4	Office, ESF, and MSV Building Heating, Ventilation, and Air-Conditioning System	26952
9.4–5	Containment Structure Ventilation System	26953
9.4–6	Service Building Ventilation System	26949
9.4–7	Auxiliary Boiler and Ventilation System	26936
9.4–8	Hot Water Heating System	26937
9.4–9	Technical Support Center HVAC	26956
9.5–1	Fire Protection System	26946
9.5–2	Emergency Generator Fuel Oil System	26917
9.5–3	Emergency Generator Cooling, Starting Air, and Lube Oil System	26916
9.5–5	Nitrogen and Hydrogen System	26939
10.2–1	Electro-Hydraulic Control System	26940
10.2–2	Turbine Generator and Feed Pump Oil Generator System	26941
10.2–3	Turbine Generator Support System	26942
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TABLE 1.7-2 PIPING AND INSTRUMENTATION DIAGRAMS (CONTINUED)

(See Summary Table of Contents for a Complete List of NUSCO Numbers and Sheets for each FSAR Figure or refer to Plant Document Control for Latest Document Rev and Date)

FSAR Figure Number	Piping and Instrumentation Diagram Title	NUSCO Number
10.3–2	Turbine Plant Miscellaneous Drains System	26945
10.3–3	Chemical Feed System	26931
10.4–1	Condensate System	26926
10.4–2	Condenser Air Removal and Waterbox Priming System	26927
10.4–3	Extraction Steam and Turbine Gland Seal and Exhaust System	26924
10.4–4	Circulating Water System	26932
10.4–5	Condensate Demineralizer Mixed Bed System	26928
10.4–6	Feedwater System	26930
10.4–7	Feedheater and MSR Vents and Drains System	26925
10.4–9	Auxiliary Steam, Feedwater, and Condensate System	26935
11.2-1	Radioactive Liquid Waste and Aerated Drains System	26906
11.2-1	Condensate Demineralizer Liquid Waste System	26929
11.3-1	Radioactive Gaseous Waste System	26909
11.4-1	Radioactive Solid Waste System	26910
12.3–5	Containment Monitoring System	26954

TABLE 1.7-3 OMITTED

TABLE 1.7-4 SPECIAL REPORTS AND PROGRAMS

The information contained in Table 1.7-4 is retained for historical purposes. Information provided here was relevant at the time of operating license application.

1. Transmitted at Time of FSAR Submittal
   - A. Failure Modes and Effects Analysis (FMEA)
   - B. Fire Protection Evaluation Report (FPER)
   - C. Millstone Nuclear Power Station Emergency Plan
   - D. Modified Amended Security Plan (MASP) (Under Separate Enclosure)
   - E. Millstone 3 Design Basis Response to Regulatory Guide 1.97, Revision 2
2. Previously Transmitted

   Report on Faults and Soil Features Mapped in the Discharge Tunnel Excavation
3. Transmitted after Submittal
   - A. Probabilistic Safety Study (PSS)
   - B. Control Room Design Review (Chapter 18)
   - C. Inservice Inspection Program
   - D. Environmental Qualification of Electrical Equipment Report (EEQ)
   - E. Environmental Qualification of Mechanical Equipment Report (MEQ)
4. Program to be Reviewed at Millstone 3 Equipment Qualification Documentation (EQD)

1.8 CONFORMANCE TO NRC REGULATORY GUIDES

Table 1.8-1 lists NRC Division 1 Regulatory Guides. This table is annotated whenever a new or revised Regulatory Guide is invoked or adopted. It identifies applicable FSAR sections, and indicates the degree of Millstone 3 compliance.

TABLE 1.8-1 NRC REGULATORY GUIDES

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
1.1	Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps (Rev. 0, November 2, 1970)	(1) Comply with the following exception: For the recirculation system phase, the vapor pressure of the water in the sump is assumed to be equal to the containment pressure. The vapor pressure of the sump water cannot exceed the containment total pressure; therefore, assuming they are equal gives the limiting low value of available NPSH.	6.2.2 6.3
1.2	Thermal Shock to Reactor Pressure Vessels (Rev. 0, November 2, 1970)	(1) See Section 1.8N.	5.2.3.3
1.3	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors (Rev. 2, June 1974)	Not applicable. Applicable only to BWRS.	
1.4	Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors (Rev. 2, June 1974)	Comply R.G. 1.4 is only used for evaluation of original plant shielding design.	12.2.1.1
1.5	Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors (Rev. 0, March 10, 1971)	Not applicable. Applicable only to BWRS.	
1.6	Independence between Redundant Standby (On site) Power Sources and Their Distribution Systems (Rev. 0, March 10, 1971)	Comply	8.3.1.4 8.3.2
1.7	Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident (Rev.3, May 2003)	(1) Comply	6.2.5
1.8	Personnel Selection and Training (Rev. 1-R, May 1977)	Compliance is as described in the QAPD Topical Report	13.2 17.1

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
1.9	Selection of Diesel Generator Set Capacity for Standby Power Supplies (Rev. 2, December 1979) (2)	Comply, with the following exceptions: The magnetizing inrush current due to the four 4,160-480 V load center transformers may cause a momentary (3 to 5 cycles) dip in voltage prior to the first load block. This momentary voltage dip to levels outside that allowed by the Regulatory Guide for load sequencing is considered inconsequential to the successful loading of the standby generator unit. C.11 Section 6.5, Site Acceptance Testing, and Section 6.6, Periodic Testing, of IEEE Std. 387-1977 should be supplemented by R.G. 1.108. The Millstone 3 position on R.G. 1.108 has several clarifications. (See R.G. 1.108).	8.3.1
1.10	Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures (Rev. 1, January 2, 1973)	Withdrawn: Withdrawal of this guide does not alter any prior or existing licensing commitments based on its use. A position statement follows. 1. Reinforce bars with a radius curve of 60 ft.-0 in. or greater are tested at the sampling frequency specified in paragraph C4a. Reinforcing bars with a radius of curvature of less than 60 ft.-0 in. are tested using only sister splices with the following frequency for each splicing crew: • One sister splice for the first 10 production splices. • Four sister splices for the next 90 production splices. • Three sister splices for the next and subsequent units of 100 production splices. If any sister splice used for tensile testing fails to equal or exceed 125% of the minimum yield strength specified for the reinforcing bar or the average tensile strength of each group of 15 consecutive samples fails to equal or exceed the guaranteed minimum tensile strength of the reinforcing bar, the individual Cadwelder shall be stopped and the procedure in Section C.5 of the Regulatory Guide will be followed.	3.8.1.6.2

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		The second paragraph of Position C.3 of R.G. 1.10 states that production mechanical splice samples for tensile testing should not be used from curved reinforcing sections, and then refers to Paragraph 4b for sampling frequency. Paragraph 4b provides for a combination of production and sister splices which appears to be inconsistent with the requirement to use straight sister splices. In preparing the Regulatory Guide, the NRC (formerly USAEC) assumed that the Cadwelders performed splices in the horizontal, vertical, and diagonal directions on the same day. Thus, there would be occasional splices on straight vertical bars which could be alternated with the curve bar splicing to permit the frequency of testing in paragraph 4b of R.G. 1.10, which requires both production and sister splices. However, construction is very apt to perform splices in one position only for more splices than those requiring another set of tests. The NRC agreed that it would accept a testing program using only sister splices whenever curved bars with a radius of curvature less than 60 ft. are mechanically spliced. The 60 ft. was established at SWEC's suggestion as we have obtained satisfactory test results for splices on bars of this or greater radii. Both the NRC and SWEC concede that the value obtained by test at this curvature of rebar will be slightly less than if a straight pull could be made. However, if otherwise practical, the NRC does not like a testing program using only sister splices. 2. If any completed mechanical splice fails to pass the visual inspection specified in Paragraph C.2 and the rate of splices that fail the visual inspection does not exceed 1 for each 15 consecutive observed splices, the sampling program will be started anew without requalifying the crew. If the failure rate exceeds 1 in 15, the crew will be re-qualified.	

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		The NRC issued a memo on a meeting held in Bethesda, Maryland on May 8, 1973 between the Commission and Erico Products, Inc. on Cadwelds, stating that requalification of a splicer should not necessarily be based on a single visual inspection. The memo stated that statistical sampling procedures permit a discard sample. Since the tensile test sampling program is based on 1 in 15 consecutive samples, the same statistical sampling procedure is used for the visual inspection sampling program.	
1.11	Instrument Lines Penetrating Primary Reactor Containment (Rev. 0, March 10, 1971)	Comply	6.2.4
1.12	Instrumentation for Earthquakes (Rev. 1, April 1974)	Comply, except as noted in FSAR section 3.7.4.1.	3.7.4.1
1.13	Spent Fuel Storage Facility Design Basis (Rev. 1, December 1975)	(1) Comply, except for Fuel Building Ventilation and Filtration requirements. Fuel building ventilation and filtration systems are not credited in the radiological accident analyses.	9.1.2
1.14	Reactor Coolant Pump Flywheel Integrity (Rev. 1, August 1975)	(1) See Section 1.8N.	5.4.1.1
1.15	Testing of Reinforcing Bars for Category I Concrete Structures (Rev. 1, December 28, 1972)	Withdrawn: Withdrawal of this Guide is not intended to alter any prior or existing licensing commitments based on its use.	3.8.1.6.2
1.16	Reporting of Operating Information - Appendix A Technical Specifications (Rev. 4, August 1975)	Comply. Monthly Operating Reports superseded by Generic Letter 97-02 and License Amendment No. 223.	16
1.17	Protection of Nuclear Power Plants Against Industrial Sabotage (Rev. 1, June 1973)	Comply, with the following clarification: The plant security system is in compliance with 10 CFR 73.55, since it is a federal regulation and, therefore, supersedes the outdated R.G. 1.17.	13.6
1.18	Structural Acceptance Test for Concrete Primary Reactor Containments (Rev. 1, December 28, 1972)	Withdrawn: Withdrawal of this Guide is not intended to alter any prior or existing licensing commitments based on its use.	3.8.1.7

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
1.19	Nondestructive Examination of Primary Containment Liner Welds (Rev. 1, August 11, 1972)	Withdrawn: Withdrawal of this Guide is not intended to alter any prior or existing licensing commitments based on its use. A position statement follows. 1. Position C.1.c Leak chase channels were used in areas of inaccessibility, such as the containment liner bottom or weld configurations in which shape discontinuities were evident, such as the personnel and equipment hatches and penetrations. Leak chase channels were pressurized to containment design pressure, and soapsuds were applied to the opposite side for a more sensitive test, since the channels were tested at a greater pressure difference than would have been possible with a vacuum box. In the case of the liner bottom weld seam, which was accessible from one side only, a vacuum box test was performed before the channels were "installed," where possible. However, to verify leak tightness, all the liner bottom welds were leak chase channel pressure tested as stated above. In either case, defects were repaired and the test repeated using the same techniques to ensure compliance with leak-tightness requirements. 2. Position C.1.d Where leak chase channels were installed, the leak-tightness of the liner/penetration to channel weld was verified by pressurizing the channel with Freon 22 (after evacuating the air from the channels) to the containment structure design pressure and examining these welds with a halogen detector. This method provides a more sensitive test and greater assurance that the completed structure will satisfy the leak rate test requirements.	3.8.1.2.3 3.8.1.7.2
1.20	Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing (Rev. 2, May 1976)	(1) See Section 1.8N.	3.9.2

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
1.21	Measuring, Evaluating, and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants (Rev. 1, June 1974)	Degree of compliance to RG 1.21 is defined for the various elements contained within the guidance in the Radiological Effluent Monitoring and Off site Dose Calculation Manual (REMODCM). Exception is being taken to RG 1.21 requirement to estimate population doses as well as individual doses. The calculation of maximum individual doses in combination with a land use census of the area around the plant is sufficient to assess the radiological impact of the plant on man and to implement the requirements of the REMODCM. Population doses will not be routinely calculated and will not be included in the annual reports to the NRC. Online monitors for all potentially significant paths for release of radioactive material are provided. For those effluent paths having two or more significant contributing sources, online monitoring of the contributing sources is provided as recommended in the Guide.	11.5.1 11.5.2 12.3.4
1.22	Periodic Testing of Protection System Actuation Functions (Rev. 0, February 17, 1972)	(1) Comply, with the following exception: RG 1.22 requires that where the ability of a system to respond to an accident signal is intentionally bypassed for the purpose of testing, positive means should be provided to prevent expansion of the bypass condition to redundant or diverse systems, and each bypass condition should be individually and automatically indicated to the reactor operator in the main control room. Permanently wired interlocks and annunciator circuitry are required to fully comply with the preceding requirement. Test circuitry is provided for items 1 through 8, shown in Table 1.8N-1. Items 9 through 13 rely on administrative controls to provide indication and prevent expansion of the bypass condition.	7.2 7.3.2.2
1.23	On site Meteorological Programs (Rev. 0, February 17, 1972)	Comply	2.3.3

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
1.24	Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure (Rev. 0, March 23, 1972)	Not applicable. No tanks are used for the storage of radioactive gases. However, an analysis of the most severe potential radioactive gaseous release is presented in FSAR Chapter 15.	
1.25	Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors (Rev. 0, March 23, 1972)	Regulatory Guide 1.25 is not used for fuel handling accident analyses. Current analyses are based on Regulatory Guide 1.183.	15.7.4
1.26	Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste Containing Components of Nuclear Power Plants (Rev. 3, February 1976)	(1) Comply, with the following exceptions: 1. The safety class terminology of ANSI N18.2 and ANSI N18.2-a,1975 is used instead of the quality group terminology. Thus, the terms Safety Class 1, Safety Class 2, Safety Class 3, and non nuclear safety (NNS) are used instead of Quality Groups A, B, C, and D, respectively. 2. Regarding Regulatory Positions C.1.e and C.2.c, one safety valve designed, manufactured, and tested in accordance with ASME III Division 1 (i.e., a code safety valve) is considered acceptable as the boundary between the reactor coolant pressure boundary and a lower safety class or NNS line. 3. Regarding Regulatory Positions C.1 and C.2, all instrument tubing, classified as Safety Class 2 or 3, is designed to ASME Section III rules, with Seismic Category I supports installed with a 10 CFR 50, Appendix B program as described in Section 3.2.3.	3.2.2

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		4. Groundwater removal post accident via the porous concrete groundwater removal system is a safety-related function. The components within the Engineered Safety Features Building (ESFB) that are accessible/retrievable from the roof are classified as non-nuclear safety instead of Safety Class 3 commensurate with their importance to safety. The sump and piping in the ESFB are Safety Class 3, however, the sump is designed to the guidance of the AISC Code and the piping is designed to the ANSI B31.1 Code. The electrical power for the sump pump is supplied from a single safety related source and is routed via non-safety related power cables. 5. The service water system supply and return piping for the post accident sample cooler is designed to ANSI B31.1 requirements and is seismically qualified.	
1.27	Ultimate Heat Sink for Nuclear Power Plants (Rev. 2, January 1976)	Comply	2.4.11.6 9.2.5
1.28	Quality Assurance Program Requirements (Design and Construction) (Rev. 2, February 1979) (3)	1. (1) Prior to 12/14/05, Millstone Unit 3 complied as follows: Construction - Millstone 3 complied with R.G. 1.28, Rev. 0. Operation - Millstone 3 complies with R.G. 1.28, Rev. 2. 2. Current compliance is as described in the QAPD Topical Report.	17.1
1.29	Seismic Design Classification (Rev. 3, September 1978)	(1) Comply	3.2.1
1.30	Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment (Rev. 0, August 11, 1972)	No longer comply - 12/14/05 (QA standards are described in QAPD Topical Report.)	17.1

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
1.31	Control of Ferrite Content in Stainless Steel Weld Metal (Rev. 2, May 1977) (2)	(1) Comply, with the following clarification: The control of ferrite content in stainless steel weld metal meets the requirements of R.G. 1.31, Rev. 1, dated June 1973, or Rev. 3, dated May 1977, on a case-by-case basis. Rev. 1 requirements are met by our method of qualifying welding procedures and controlling heat input during welding. This is subsequently verified by magnetic measurements of production welds. Rev. 3 requirements are met by purchasing to specifications which require the manufacturer to test each heat and lot for delta ferrite content. The weld pad used is described in SFA 5.4 (1971 version). This weld pad differs in dimensions and in the chill bars (steel versus copper) from the weld pad described in Rev. 3 of R.G. 1.31. However, it should be noted that both weld pads are designed to produce undiluted weld metal for test purposes. Measurements are required of the "Delta ferrite content at 5 places on the as-deposited, undiluted weld metal using a magnetic device calibrated to a single set of traceable standards." Thus, the intent of the Regulatory Guide (Rev. 3) is met; i.e., to test as-deposited, undiluted weld filler metal for each heat, lot, and process to be used in production. Since the program meets the intent of either version of R.G. 1.31, field welding of ASME III austenitic stainless steel has been performed with weld filler metal purchased as indicated above and no subsequent testing or production welds is performed.	4.5.1 4.5.2.4 5.2.3.4 5.3.1.4 6.1.1.1 10.3.6.2
1.32	Criteria for Safety-Related Electric Power Systems for Nuclear Power Plants (Rev. 2, February 1977) (3)	Comply, with the following exception: Battery Discharge Test must be performed at the specified testing frequency as described in the Surveillance Frequency Control Program.	8.1.5 8.3.1 8.3.2

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
1.33	Quality Assurance Program Requirements (Operation) (Rev. 2, February 1978)	Compliance is as described in the QAPD Topical Report.	13.4 13.5.1.1 17.1
1.34	Control of Electroslag Weld Properties (Rev. 0, December 28, 1972)	Not applicable. There electroslag welding process for fabricating components was not used.	
1.35	Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures (Rev. 2, January 1976)	Not applicable. There is no prestressed concrete containment structure.	
1.36	Nonmetallic Thermal Insulation for Austenitic Stainless Steel (Rev. 0, February 23, 1973)	1. (1) Prior to 12/14/05, Millstone 3 complied, with the following clarifications and exceptions: Position C.1 states the packaging and shipping requirements of this guide. In lieu of controlled packaging and shipping, receipt inspection and tests are required, by specification. This consists of visual inspection for physical or water damage to all cartons. Damaged cartons are segregated. The potentially contaminated insulation is not accepted unless randomly selected samples from each carton are shown to be acceptable after being re-subjected to the production test outlined in the Guide. 2. Current compliance is as described in the QAPD Topical Report.	5.2.3 6.1.1 17.1
1.37	Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants (Rev. 0, March 16, 1973)	1. (1) Prior to 12/14/05, Millstone 3 complied. 2. Current compliance is as described in the QAPD Topical Report.	4.5.1 17.1
1.38	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants (Rev. 2, May 1977)	1. (1) Prior to 12/14/05, Millstone 3 complied. 2. Current compliance is as described in the QAPD Topical Report.	17.1

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
1.39	Housekeeping Requirements for Water-Cooled Nuclear Power Plants (Rev. 2, September 1977)	1. Prior to 12/14/05, Millstone 3 complied as follows: Construction - Millstone 3 complied with Rev. 0 of the Guide. Operation - Millstone 3 complied with Rev. 2 of the Guide. 2. Current compliance is as described in the QAPD Topical Report.	12.5.3 17.1
1.40	Qualification Tests of Continuous Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants (Rev. 0, March 16, 1973)	(1) Comply	3.11 8.3
1.41	Preoperational Testing of Redundant On site Electric Power Systems to Verify Proper Load Group Assignments (Rev. 0, March 16, 1973)	Comply	8.1 8.3
1.42	Interim Licensing Policy As Low As Practicable for Gaseous Radio Iodine Releases from Light-Water-Cooled Nuclear Power Plants	Withdrawn	
1.43	Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components (Rev. 0, May 1973)	(1) Comply	5.3.1.4
1.44	Control of the Use of Sensitized Stainless Steel (Rev. 0, May 1973)	(1) Comply, with the following clarification: The intent of Paragraph C.6 is complied with in varying degrees as follows: 1. Field fabrication and erection of ASME III piping complies with Paragraph C.6, except that the ASTM A708-74 standard is used to perform the intergranular corrosion testing. The radius of the bend specimen is as specified in ASME IX with the weld metal-base metal interface located at the centerline of the bend. This meets the intent of Paragraph C.6.	4.5.1 4.5.2.4 5.2.3.4 5.3.1.4 6.1.1.1 10.3.6.2

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		R.G. 1.44 requires that an intergranular corrosion test, such as ASTM A262 Practice A or E, should be performed to evaluate sensitization of the heat affected zone in stainless steel weldments having a carbon content greater than 0.03%. The ASTM A262 is a very severe test and most of unstabilized stainless steel weldments with a carbon content above 0.03% would not pass it. According to the NRC Regulatory Standard Review Plan (Section 4.5.1), ASTM A708-74 test (previously A393) is acceptable for testing the qualification welds for degree of sensitization. Paragraph 7.1.1 of ASTM A708-74 requires that the test specimen is bent 180° over a diameter equal to the thickness. This requirement is applicable to the base metal, but ductility of the weld metal and the heat affected zone may not be that good even without any corrosion attack. his is recognized in ASME IX, which required testing of weldments over a diameter equal to four times the thickness. 2. Shop fabrication of ASME III piping, tanks, and valves and field fabrication of ASME III tanks require control of heat input during welding so as to avoid severe sensitization of the weld zone. In addition, the maximum interpass temperature is limited to 350°F. While no testing for intergranular corrosion during weld procedure qualification is required, the above controls assure that base material will not be severely sensitized during welding and meets the intent of Paragraph C.6.	

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		3. Fabrication of ASME III forged stainless steel instrumentation valves and ASME III components other than those identified in Items 1 and 2 above is performed in fabrication shops and requires a maximum interpass temperature of 350°F. While no testing for intergranular corrosion during weld procedure qualification is required, this specific control reduces the possibility of a severely sensitized heat-affected zone during welding. In addition, the need for the shop fabricator to provide unsensitized heat-affected zone during welding. In addition, the need for the shop fabricator to provide unsensitized components is specifically identified in all procurement specifications by requiring supplied material to be capable of meeting ASTM A262, Practice A or E. Shop practice generally recognizes the need to limit heat input during the welding through good fit-up, adequate welder accessibility, proper positioning, and close supervision. Finally, most of the pressure retaining components of the RCPB piping system are castings and, because of their delta ferrite content, are highly resistant to stress corrosion cracking.	
1.45	Reactor Coolant Pressure Boundary Leakage Detection Systems (Rev. 0, May 1973)	Comply with the following interpretation given to Regulatory Position C.5: The sensitivity and response time of each leakage detection system employed to collect unidentified leakage are as shown in the following table: **System** — **Sensitivity and Response Time** Containment Drain Sump Level or Pumped Capacity Monitoring System — 1 gpm in less than 1 hour Containment Atmosphere Humidity, Temperature and Pressure — Humidity, temperature or pressure monitoring of the containment atmosphere are considered as alarms or indirect indication of leakage to the containment	5.2.5

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance		FSAR Section Reference
		Containment Atmosphere Gaseous and Particulate Radioactivity Monitoring	1 gpm in less than 1 hour, provided that the equilibrium activity of the reactor coolant is sufficiently high and the equilibrium activity of the containment atmosphere is below a level that would mask the change in activity corresponding to this leak rate	
		Comply with the following interpretation given to Regulatory Position C.7 Indicators and alarms for each leakage detection system should be provided in the main control room. Procedures for converting sump level and pumped capacity indications to a common leakage equivalent should be available to the operators. The calibration of the indicators should account for needed independent variables. Due to the numerous factors that can affect the readings from the Containment Atmosphere Gaseous and Particulate Radioactivity Monitors, they can not be used to reliably quantify a leak rate, although they are a very sensitive indicator of a leak. Factors affecting the monitor readings include RCS radioactivity levels, containment air radioactivity levels, radionuclide mix, location of leak and removal mechanisms.		5.2.5
1.46	Protection Against Pipe Whip Inside Containment (Rev. 0, May 1973)	(1) Comply with the following clarifications and exceptions: <u>Paragraph C.1.b</u> At intermediate locations between terminal ends selected by either of the following criteria: 1. At each pipe fitting, welded attachment, and valve; or 2. At locations where the primary plus secondary stress intensities (circumferential or longitudinal) derived on an elastically calculated basis under loadings associated with specified seismic events and operational plant conditions exceed 2.4 S_m <u>in lieu of</u>		3.6

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		"Any intermediate locations...conditions exceed 2.0 S_m for ferritic steel and 2.4 S_m for austenitic steel." Paragraph C.2.b At intermediate locations between terminal ends selected by either of the following criteria: At intermediate locations between terminal ends selected by either of the following criteria: 1. At each pipe fitting, weld attachment, and valve; or 2. At locations where either the circumferential or longitudinal stresses derived on an elastically calculated basis under the loadings associated with specified seismic events and operational plant conditions exceed 0.8 (1.2 S_h + S_a) in lieu of "Any intermediate locations...conditions exceed 0.8 (S_h + S_a)." Paragraph C.3.a, Footnote 10 Longitudinal breaks are assumed to result in axial split without pipe severance. Splits oriented (but not concurrently) at two diametrically opposed points on the piping circumference such that jet reactions cause out-of-plane bending of the piping configuration. Alternatively, a single split may be assumed at the section of highest stress as determined by a detailed stress analysis (e.g., finite element analysis).	

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		The dynamic force of the fluid jet discharge should be based on circular or elliptical (2D x 1/2D) break area equal to the effective cross-sectional flow area of the pipe at the break location and on a calculated fluid pressure modified by an analytically or experimentally determined thrust coefficient as determined for a circumferential break at the same location. Line restrictions, flow limiters, positive pump-controlled flow, and the absence of energy reservoirs may be taken into account as applicable, in the reduction of jet discharge. in lieu of Footnote 10 of R.G. 1.46. Paragraph C.3.b, Footnote 11 Pipe whipping is assumed to occur in the plane defined by the piping configuration and geometry and to cause pipe movement in the direction of the jet reaction. in lieu of Dynamic forces resulting...cause whipping in any direction normal to the pipe axis (Footnote 11 of R.G. 1.46). Implementation of additional criteria is documented and approved by NRC as providing an acceptable level of plant safety. These criteria potentially include those bases provided by NRC Standard Review Plans 3.6.1 and 3.6.2, the approved topical reports and SARs submitted by NSSS, and the draft of ANSI-N176, Standard Design Basis for Protection of Nuclear Power Plants Against Effects of Postulated Pipe Rupture.	

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		Certain provisions of R.G. 1.46 are impractical. While NRC is in the process of revising R.G. 1.46, the NRC Standard Review Plant 3.6.2, Paragraph 11.1 stated "if the criteria specified in R.G. 1.46 are impractical for a specific application, the criteria of Branch Technical Position BTP-MEB-3-1 will be considered," and "BTP-MEB-3-1 may be used for all applications, in lieu of References 3 and 4, at the option of the Applicants." The modifications or exceptions taken by Millstone 3 on position of R.G. 1.46, are based upon BTP-MEB-3-1. These positions were incorporated in the PSAR of WEPCO and SWESSAR, and the Safety Evaluation Reports (SER) issued by the NRC for these two projects found it to be acceptable.	
1.47	Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems (Rev. 0, May 1973)	Comply, with the following clarifications: 1. An indicator of bypass/inoperability, located in the control room, will be provided for redundant or diverse portions of each safety system. Bypass indication will be provided for any deliberate action that renders a safety system inoperable. 2. Bypass of redundant portions of engineered safety feature support systems warrants indicators that must be differentiated from safety system bypass indicators.	7.5.3
1.48	Design Limits and Loading Combinations for Seismic Category I Fluid System Components (Rev. 0, May 1973)	(1) Comply, with the following exception: Class 2 and 3 piping analysis to follow criteria specified in 1971 ASME III Code, and all Addenda up to and including Summer 1973, except that piping systems other than safety injection and quench/recirculation spray use an increased allowable (design limit) of 2.4 S for plant faulted loading conditions. Components, except piping, use stress criteria given in Tables 3.9B–5 and 3.9B–7. A comparison of these valves and R.G. 1.48 are given in Tables 3.9B–6 and 3.9B–8. R.G. 1.48 has been superseded by NUREG 0800, SRP 3.9.3. Refer to FSAR Section 1.9, SRP 3.9.3 for further discussion.	3.9.3

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
1.49	Power Levels of Nuclear Power Plants (Rev. 1, December 1973)	Comply, with the following exception: The analyses and evaluations for those conditions described in Position C.2 and the analyses of the possible off site radiological consequences described in Position C.3 are made at an assumed core power level using the guidelines in Position C.3. The margin specified in Position C.2 is equal to 1.02 times the licensed power level to allow for possible instrument errors in determining the power level. The words "...equal to 1.02..." are interpreted to mean "...equal to at least 1.02...". This interpretation allows the analyses and evaluations described in Position C.2 to be made, at an Applicant's discretion, at a somewhat higher power level to account for the margin which may be provided in turbine generator designs above rated capacity. This interpretation can be found in Position C.3, and is, therefore, also considered applicable to Position C.2. As described in FSAR Chapter 15, most safety analyses demonstrating that the DNB design criterion is met do not explicitly use 1.02 times the licensed thermal power for full power initial condition assumptions.	1.1 15.0
1.50	Control of Preheat Temperature for Welding of Low-Alloy Steel (Rev. 0, May 1973)	(1) Comply, with the following exception: In cases where it is impractical to maintain preheat until a postweld heat treatment has been performed, the preheat temperature is maintained for sufficient time to assure that the residual hydrogen has effused from the weld zone This reduces the tendency to form cracks in the weldment and, therefore, complies with the requirements of this Guide.	5.2.3.3
1.51	Inservice Inspection of ASME Code Class 2 and 3 Nuclear Power Plant Components	Withdrawn	

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
1.52	Design, Testing, and Maintenance Criteria for Engineered-Safety-Feature Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants (Rev. 2, March 1978)	Comply, with the following clarifications and exceptions: Paragraph C.2.g The filter trains are not instrumented to monitor, alarm and record flow rates in the control room. Flow through the filters are verified on a monthly basis. The flow through the fuel building filters are not required to be verified. Paragraph C.2.h The following exceptions are taken to the requirement that "all instrumentation and equipment controls should be designed to IEEE 279": 1. All instruments and equipment controls that sense or process one or more variables and that act to accomplish the protective function are designed in accordance with IEEE 279. These include sensors, signal conditioners, logic, and actuation device control circuitry. (The protective function with which the subject guide is concerned is atmospheric cleanup to mitigate accident doses.) 2. In addition, a very limited class of analog indicators may be designed in accordance with selected applicable paragraphs of IEEE 279. The basis for selecting specific indicators to be so designed, is their significance to safety. All paragraphs of IEEE 279 are applicable, except 4.12, 4.13, 4.15, 4.16, and 4.17. For this limited class of indicators, redundant analog channels are provided. One channel is recorded. The systems are designed to operate before and after, but not necessarily during, a safe shutdown earthquake. 3. Annunciator functions are incorporated in overall system design. Annunciators are not safety-related; therefore, they are not designed in accordance with IEEE 279. Paragraph C.2.i (Clarification)	6.5.1.2

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		Fuel building exhaust system actuation is manual from the control room. Fuel building high radiation will annunciate in the control room if the radiation level reaches the predetermined setpoint during normal plant operation, as well as DBA. When the radiation alarm from the particulate and gas monitor for the exhaust duct work has been verified, the filtration system may then be actuated manually. Paragraph C.2.j (Clarification) Adsorber design provides for the replacement of charcoal by an external vacuum system. The structural design of the filter train will provide for the removal of the filter components on a cell by cell basis. Demisters, heaters, fans and casings will be decontaminated by wash down process; wash down liquid will drain to an aerated drain system. Paragraph C.2.l Housing leak tests are performed in accordance with the provisions of Section 6 of ANSI N510-1980. Leak rate of 0.1 percent in accordance with Table 4-3 of ANSI N509-1980 is acceptable. However, ductwork leak tests shall be performed in accordance with the procedures delineated in Chapter 8, Leak Testing, of the Manual for the Balancing and Adjustment of Air Distribution Systems, published by SMACNA, 1967. All ESF and non-ESF ventilation systems interconnecting with safety-related filter trains comply with ANSI N509-1980 leakage requirements, with the following clarifications: 1. The following fan systems will exceed ANSI N509-1980 leakage requirements on the suction side: • Containment purge exhaust fan. • Suction side on fuel building filters and fuel building normal exhaust fan. The leakage flow into the ductwork will have no significant effect on air cleaning effectiveness since the fans and filters have enough capacity to handle the additional flow.	

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		2. The fuel building ductwork on the discharge side of the filter exhaust fans will exceed ANSI N509-1980 leakage requirements. This is a radiation protection consideration because the system will exhaust air from contaminated areas then leak into relatively cleaner areas. The consequences of leakage from this system during normal operation is inconsequential since the activity in the duct will be less than 1 MPC. However, personnel access may have to be limited to the top floor of the auxiliary building, where the ductwork is located, following a fuel handling accident. 3. The leakage on the suction side of the SLCRS exceeds ANSI requirements, but will have no adverse effect on maintaining negative pressure within the cubicles since the fan can accommodate the added flow. The leakage on the discharge side of the fan also exceeds ANSI requirements, but will be into the same areas being exhausted. This type of leakage affects only effectiveness of the SLCRS to draw down the pressure. The fans are sized to compensate for this leakage. Also, the leakage between the filter outlet and the fan suction which is located in the auxiliary building is in excess of ANSI requirements. The effect of this leakage on system performance has been evaluated and was found to be acceptable. 4. Leak test boundaries for the control building ESF filters are defined in accordance with ANSI N509-1980, Appendix B, Figures B-3 and B-4, schemes 13 and 22, to be from the filter inlet to filter outlet. 5. Millstone 3 complies with ANSI N509-1980 requirements stated in Paragraph 4.6.3, which calls for the use of design static pressure (working pressure) in testing for duct leakage.	

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		However, fan peak pressure is not used during leakage testing of segments of ductwork which could be isolated by a closed damper or a clogged filter as implied by Paragraph 4.6.2 of ANSI N509-1980. The bases for the clarification are as follows: • Total clogging of filter or closure of a damper on fan suction will have greater impact on air cleaning effectiveness than the increased leakage resulting from fan shut-off pressure. Therefore, duct leakage is determined at maximum working pressure. • Closure of dampers located on the discharge side of the fan will result in low or no flow in the system, thus causing that particular fan to stop and the redundant one to start. Such action will preclude the possibility of any significant duct outleakage and eliminate this health/physics concern. Paragraph C.3.d (Clarification) Filters have been purchased to ANSI N509-1976. Filter media will be subjected to velocities recommended by the HEPA filter manufacturer which exceeds ANSI N509-1976 requirements given in Section 4.3.1. The HEPA filter cell testing is conducted initially at the manufacturer's facilities and again after installation at the plant site. All HEPA filters furnished are equipped with face guards in accordance with MIL-F-51068. When installed in the filter housing, the HEPA filters and housing are inspected for defects and tested for leak-tightness in accordance with ANSI N510-1980. Paragraph C.3.e (Clarification) Filter mounting frame is constructed and designed in accordance with the recommendations of Section 4.3 of ERDA 76-21, except for the frame tolerance guidelines in Table 4.2. The tolerances selected for HEPA mountings are sufficient to satisfy the bank leak test criteria of Paragraphs C.5.c and C.5.d of RG. 1.52, Rev. 2. Paragraph C.3.g	

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		Millstone 3 complies with ANSI N509-1980 paragraph 4.6.2.2 with respect to designing inlet units and components which can be isolated from the fan to withstand a peak negative pressure by ensuring that such isolation is precluded via the design control logic between the fans and the inlet dampers. Compliance with designing inlet units and components, as noted in the same paragraph with respect to the plugging of such components, is demonstrated via routine surveillance and subsequent filter replacement as necessary. Millstone 3 is in accordance with ANSI N509, except access to the control building filter units is not provided with hinged doors or inspection windows. Access is via 20-inch by 40-inch bolted panels. Other units are provided with hinged doors or bolted panels with inspection windows. There is no internal lighting. Even though doors are not available to access both sides of each bank of components, Millstone 3 complies with the intent of the requirements of Paragraph 5.6 of ANSI N509-1976 to provide access to each side of each component of the ESF ventilation filtration systems filter housings for maintenance and testing. Paragraph C.3.h Exception is taken to the recommendations of Section 4.5.8 of ERDA 76-21 relative to drain sizes and arrangement. Normally closed manual valves, instead of water seals and traps, will be provided to control the discharge of the fire sprinkler flow. Sprinkler flow will be a timed discharge, and the water will be contained within the housing until it is removed to the liquid radwaste system at a controlled rate. Condensate from the moisture separator chamber will continually drain via normally open drain valves. Paragraph C.3.i	

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		The dwell time for the minimum 2 inches of the carbon adsorber unit of the Auxiliary Building Filtration System is 0.22 sec. All filters use a 4-inch thick charcoal bed which exceeds the minimum 2 inches recommended by R.G. 1.52, Rev. 2. The additional 2 inches will result in a dwell time of 0.43 sec. for the Auxiliary Building Filtration System. Carbon is purchased to Table 5-1 of ANSI N509-1980. Pre Generic Letter 99-02 testing of the charcoal was based on a maximum face velocity of 46 fpm and a 2-inch thick bed. Testing of new and used ESF filter system charcoal, post Generic Letter 99-02 (Ref. Amendment 184) uses ASTM D3803-89 testing standards with face velocities based on the greater of the individual system upper Technical Specification air flow limit (cfm) or 12.2 meters/min. (40 fpm) Paragraph C.3.k When conservative calculations show that the maximum decay heat generation from collected radioiodines is insufficient to raise the carbon bed temperature above 250°F with no system air flow, ESF atmosphere cleanup systems may be designed without a decay heat removal mechanism. (See FSAR Table 6.5–1 for applicability.) In addition, exception is taken to provide humidity control for the decay heat removal system cooling air flow which uses room air of less than 70% relative humidity. Paragraph C.3.l System resistances will be determined in accordance with Section 5.7.1 of ANSI N509-1976 except that fan inlet and outlet losses will not be calculated in accordance with AMCA 201. Fan blast area data necessary to calculate inlet and outlet losses, per AMCA 201, are the responsibility of fan manufacturers, and are not available from them. The following vibration performance criteria will be used: Acceptable Alert Action ≤ 0.325 in/sec > 0.325 in/sec > 0.700 in/sec instead of ANSI N509-1976 Section 5.7.3.	

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		Documentation will not be furnished in accordance with Section 5.7.5 where AMCA certification ratings are submitted. An engineering evaluation and a test determined that the SLCRS exhaust fans were correctly selected to operate, and do operate, on the stable portion of the fan curve during single or parallel operation. Paragraph C.3.n Exception is taken to Section 5.10.3.5 of ANSI N509-1976; ductwork, as a structure, will have a resonant frequency above 25 Hz, but this may not be true for the unsupported plate or sheet sections. The design provides for specification of the resonant frequency range of the support hangers. Specifying the resonant frequency of the unsupported plate or sheet has no meaning in the design. Exception is taken to Section 5.10.5 of ANSI N509-1976 on welding in accordance with AWS D1.1 or ASME Section IX. AWS D1.1 General Provisions Section stipulates that the code is not intended to apply to welding base less than 1/8 inch thick. Since ductwork thicknesses are below 1/8 inch, the AWS D9.1 code, Specification for Welding of Sheet Metal, is used. In addition, exception is taken to the following: Workmanship samples shall be inspected with liquid penetrant or magnetic particle on both root and face surfaces. Workmanship samples shall be inspected by macro-sectioning butt welds. Macro-sectioning is a satisfactory method to determine butt weld quality, including both root and face surfaces. Exception is taken to Section 5.10 of ANSI N509-1976 regarding allowable stresses for the structural analysis of ductwork. Requirements shall be as described in Section 5.10.3.3 of ANSI N509-1980. Exception is taken to Section 5.10.9 of ANSI N509-1976 regarding balancing of all duct systems. All duct systems shall be balanced to within ±10 percent of the specified design air flow.	

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		This value is in agreement with ANSI N509-1980. Paragraph C.3.p Exception is taken to the provisions in Section 5.9 of ANSI N509-1976 of designing dampers to ANSI B31.1 and to using butterfly valves. Class B dampers may be designed and tested to meet the verification of strength and leak-tightness necessary for use in a contaminated air stream. (Note: This exception does not pertain to containment penetrations.) In addition, exception is taken to the following: All Class II dampers on ESF atmosphere cleanup system air filtration and adsorption units have been tested for leakage rates except for two backdraft dampers on the SLCRS fan discharge. The size of the two untested backdraft dampers is bounded by both larger and smaller size dampers which have been satisfactorily tested to 50% of allowable leakage rates. Damper leakage will not impact on the air cleaning effectiveness of ESF systems. Paragraph C.4.a Exception is taken to full compliance with Section 2.3.8 of ERDA 76-21; i.e., the plant does not use any communications system, floor drains are as noted in Paragraph C.3.h above, decontamination areas and showers are not "nearby," filters are not used at duct inlets, and duct inspection hatches are not provided. Paragraph C.4.b Partial compliance, with a minimum spacing between filter frame of 2 ft.-6 in. instead of a minimum of 3 ft. This is deemed adequate since replacement of filter elements would be minimal due to system function, use, and location. Paragraph C.4.d (Clarification) ESF atmosphere cleanup systems are run a minimum of 15 Minutes per month. Paragraph C.5.a Visual inspection is performed in accordance with ANSI N510-1980.	NRC Question 460.12

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

<table>
<tr><th>R.G. No.</th><th>Title</th><th>Degree of Compliance</th><th>FSAR Section Reference</th></tr>
<tr><td></td><td></td><td>
Paragraph C.5.b
Test for air flow distribution to the HEPA filters and the adsorbers shall be conducted in accordance with Section 8 of ANSI N510-1980 with the following exceptions and clarifications:
Paragraph 8.3.1.6 - Test shall be conducted to verify the following pressure drop values across the combined HEPA filter and charcoal adsorber banks and across the entire filter housing:
<table>
<tr><th>System</th><th>ΔP Across HEPA & Charcoal (in. wg)</th><th>ΔP Across Housing (in. wg)</th></tr>
<tr><td>Supplementary Leak Collection & Release</td><td>6.25</td><td>7.75</td></tr>
<tr><td>Control Room Emergency Air Filtration</td><td>6.75</td><td>9.86</td></tr>
<tr><td>Aux. Bldg. Filter</td><td>6.80</td><td>9.80</td></tr>
</table>
Paragraph 8.3.1.7 - Exception is taken to the requirement regarding testing at 50 percent of the pressure drop value. Testing the filter at this condition will require the removal of certain filter components with potential damage due to mishandling. The above condition is not considered a design parameter of the system.
Paragraph 8.3.1.8 - Acceptance criteria for the control building shall be 1,225 cfm ±10 percent at clean filter condition, and 1,000 cfm ±10 percent at dirty filter condition. Calculations have demonstrated that the control room pressurization and dose limits are not affected by the above range of air flow.
Exception is taken to the SLCRS fan flowrate acceptance criteria of + or − 10% requirement of ANSI N510-1980. Instead a range of flows is specified; the higher end of which is established by the capacity of the filter train and the lower end by an Inservice Test.
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</table>

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		Paragraph 8.3.2.3 - Air distribution tests across the prefilter and moisture separator banks are not required by project specifications. Paragraph C.5.c HEPA Filter DOP testing is conducted in accordance with ANSI N510-1980. Exception taken to the requirement that test should be performed at least once per 18 months. Based on acceptance of 24 month fuel cycle; this test can be extended to at least once per refueling interval. Paragraph C.5.d Charcoal adsorber leak testing with refrigerant is conducted in accordance with ANSI N510-1980. Exception taken to the requirement that test should be performed at least once per 18 months. Based on acceptance of 24 month fuel cycle; this test can be extended to at least once per refueling interval. Paragraph C.6.a/b The Auxiliary Building Filtration System activated carbon adsorber section has a 4 inch bed and operating face velocity of 46 fpm (0.43 sec residence time) based on nominal design air flows. The accident dose analysis in Chapter 15 of the FSAR is based on a 95 percent decontamination efficiency. Table 2 of R.G. 1.52, Rev. 2 assigns a 95 percent decontamination efficiency for an activated carbon sample having a methyl iodide penetration of less than 1 percent. Therefore, within 31 days after removal, a 4 inch laboratory sample from the installed sample canisters of ESF filtration systems activated carbon absorber, need only demonstrate a removal efficiency of 99 percent for methyl iodide when tested in accordance with ANSI N510-1980 at 80°C and 70 percent relative humidity. Pre Generic Letter 99-02 Technical Specifications demonstrate removal efficiency of 99.825%. Technical Specification testing of used ESF filter system charcoal, post Generic Letter 99-02 (Ref. Amendment 184), uses ASTM D3803-89 testing standards assuring charcoal efficiency of 97.5% or greater.	

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
1.53	Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems (Rev. 0, June 1973)	Table 2, Note C Table 2, Note C of R.G. 1.52, Rev. 2 states that testing should be performed ". . . (2) at least once per 18 months thereafter. . ." Exception taken to the requirement that test should be performed at least once per 18 months. This test is to be performed at least once per refueling interval. (1) Comply, with the following clarifications: 1. Regulatory Position C.1 Due to the trial-use status of the source document, IEEE 379 1972, departure from certain provisions may occur. The phrase "any and all combinations of nondetectable failures" in Paragraph 3(3) of IEEE 379 is interpreted to mean that an accumulation of single nondetectable failures taken collectively,culminate in a nondetectable combination. 2. Regulatory Position C.2 The protection system, as defined by IEEE 279-1971, incorporates the capabilities for test and calibration as set forth in Paragraphs 4.9 and 4.10 of IEEE 279-1971. Final actuation devices (as defined by IEEE 379-1972) are capable of periodic testing in accordance with R.G. 1.22. Those final actuation devices which cannot be fully tested during reactor operation (for reasons as stated in regulatory positions 4a through 4c of R.G. 1.22) are subjected to a partial test with the unit online and full operational testing during reactor shutdown. Taken as a whole, the operability of all active components necessary to achieve protective functions is demonstrated via the testing program described above. 3. Regulatory Position C.3	3.1.1

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		Single switches supplying signals to redundant channels are designed with at least 6 inch separation or with suitable barriers between redundant circuits. 4. Compliance with single-failure criteria is verified based on a collective analysis of both the protective system, as defined in IEEE 279-1971, and the final actuation devices or actuators, as defined in IEEE 379-1972.	
1.54	Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants (Rev. 0, June 1973)	1. (1) Prior to 12/14/05, Millstone 3 complied, with the following clarification and exception: Clarification Compliance will not be invoked for equipment of a miscellaneous nature and all insulated surfaces. Due to the impracticability of imposing Regulatory Guide requirements to the standard shop process used in painting valve bodies, handwheels, electrical cabinetry and control panels, loudspeakers, emergency light cases and other miscellaneous equipment, the Regulatory Guide will not be invoked for these items since the total surface area for such items is relatively small when compared to the total surface area for which the requirements are imposed. Exception Quality Assurance Program recommendations stated in R.G. 1.54 are followed except for the inspection defined in Section 6.2.4 of ANSI N101.4-1972. Inspection is in accordance with ANSI N5.12-1974, Section 10. Testing of coating materials is performed in accordance with ANSI N101.2, or ASTM D3911 as noted in Section 6.1.2.1. 2. Current compliance is as described in the QAPD Topical Report.	6.1.2 17.1
1.55	Concrete Placement in Category I Structures (Rev. 0, June 1973)	Withdrawn Withdrawal of this Guide is not intended to alter any prior licensing commitments based on its use. A position statement follows:	3.8.1.6.1

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		1. Shop detail drawings for the reactor containment mat, shell, containment intervals, spent fuel pool, and dome reinforcement are checked by the designer. All other reinforcing shop details are checked by engineers at the job site. The Regulatory Guide calls for all shop drawings to be checked by the designer. The mat, shell, and dome of the reactor containment structure are checked by the designer, as these are complicated arrangements. Further, the large size bars normally used require special detailing practices to permit bending and to satisfy development of strength requirements. Therefore, these detail drawings are checked by designers who have had previous experience with the large bars. Details for conventionally reinforced structures are normally checked by engineers in the field. This allows the field to review the proposed locations of construction joints. 2. A slump of 4 inches is used for mass concrete in areas where the density of reinforcing steel requires a more plastic mix for placement. 3. Curing of the concrete for the reactor containment shell and dome conformed to Chapter 12 of ACI 301 instead of Subsection CC-4240 of ACI 359. Both have the same requirements for temperature and duration, but Chapter 12 allows curing compounds to retain the moist environment. Curing compounds are not used on the internal structures of the reactor containment. 4. The ACI and ASTM specifications are supplemented as necessary with mandatory requirements relating to types and strengths of concrete, minimum concrete densities, proportioning of ingredients reinforcing steel requirements, joint treatments, testing requirements, and quality control.	
1.56	Maintenance of Water Purity in Boiling Water Reactors (Rev. 1, July 1978)	Not applicable. Only applicable to BWRs.	
1.57	Design Limits and Loading Combinations for Metal	Not applicable.	

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
	Primary Reactor Containment System Components (Rev. 0, June 1973)	Only applicable to those plants with a metal primary reactor containment.	
1.58	Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel (Rev. 1, September 1980)	1. (1) Prior to 12/14/05, Millstone 3 complied. 2. Current compliance is as described in the QAPD Topical Report.	17.1
1.59	Design Basis Floods for Nuclear Power Plants (Rev. 2, August 1977)	Comply with the following clarification: Position C.1 specifies designing hardened protection for all safety related structures, systems, and components. Position C.1 requires hardened protection to be passive and in place, as it is to be used for flood protection, during normal plant operation. Flood protection for each service water pump cubicle is provided in part by a watertight door and a cubicle sump drain line. During normal operations, the drain line of each cubicle is open and the door of one cubicle only is open. Plant procedures require isolation of the sump drain line and watertight door of each cubicle on approaching severe weather. These actions are performed prior to sea level rising to the level which requires entry into the LCO for Technical Specification 3.7.6. A Technical Specification to ensure closing of the cubicle watertight doors in advance of a potential flooding event was required by Section 2.4 of the SER for MP3. Section 2.4 of the SER concluded that the guidelines of RG 1.59 had been met. The NRC therefore recognized that the cubicle watertight doors may be open during normal operations and found this to be acceptable.	2.4.2.2
1.60	Design Response Spectra for Seismic Design of Nuclear Power Plants (Rev. 1, December 1973)	(1) Not applicable. The NRC granted that Millstone Nuclear Power Station Unit 3 will not be required to comply with Regulatory Guide 1.60, Design Response Spectra for Seismic Design of Nuclear Power Plants, dated October 1973. Plants docketed prior to April 1, 1973 are not required to consider this Regulatory Guide. PSAR Section 2.6.2.7 specifies the response spectra used for the design of Millstone 3.	

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
1.61	Damping Values for Seismic Design of Nuclear Power Plants (Rev. 0, October 1973)	(1) Not applicable. The NRC granted that Millstone Nuclear Power Station Unit 3 will not be required to comply with R.G. 1.61, Damping Values for Seismic Design of Nuclear Power Plants, dated October 1973. Plants docketed prior to April 1, 1973 are not required to address this Regulatory Guide. FSAR Tables 3.7B-1 and 3.7N-1 list the damping values used with Seismic Design of Millstone 3.	
1.62	Manual Initiation of Protective Actions (Rev. 0, October 1973)	(1) Comply, with the following clarification and exceptions: 1. Regulatory Position C.1 a. Manual initiation at the system level is interpreted to mean no more than three operator actions will be required to initiate at least one train, division, or channel of final actuation devices, including support systems. b. Engineering judgment will be exercised to assure that a minimum of operator actions are required to achieve system level manual initiation without unnecessarily jeopardizing the return to operation of the power plant. For protective actions that significantly affect return to operation, or for those protective actions that may, if inadvertently initiated, result in a less safe plant condition, operator actions on two control devices will be required. c. Designs requiring more than two operator actions per train, division, or channel to achieve protective action are to be limited to those actions required only in the long term and will be evaluated on a case-by-case basis. 2. Regulatory Position C.2 All equipment that contributes to the protective action will be initiated at the system level.	7.3.2.2.7

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		There is no auxiliary feedwater system low-low SG water level manual initiation design feature which replicates the automatic protective action design feature. Except for the safety injection and containment depressurization manual initiation design features which will initiate motor-driven auxiliary feedwater pumps, auxiliary feedwater manual initiation requires that the operator start the pumps and isolate steam generator blowdown lines. During normal power operation, plant operating and surveillance procedures require that all auxiliary feedwater valves be aligned in the positions which provides a path to the steam generators. No operator action to perform auxiliary feedwater valve position changes is required to establish a flow path to the SG. The motor-driven AFW pump automatic initiation system closes the steam generator blowdown sample valves (3SSR*CTV19A-D). Engineering judgment is exercised to conclude that steam generator blowdown sample line isolation is inconsequential to the accomplishment of the AFW safety function because these lines are 3/8 inch lines and have insignificant flow capacity. Therefore, the sample valves are not counted as a required operator action to manually initiate the auxiliary feedwater system protective action. At below 10% rated thermal power levels, plant operators may throttle or close flow control valves (3FWA*HV31A/B/C/D) for steam generator water inventory control (or to support system alignment or restoration from normal use during startup, shutdown, and hot standby). These control valves do not receive an open signal from safety injection nor containment depressurization system manual initiation design feature. In this case, manual operator action is credited to open these control valves, if required, and to support auxiliary feedwater system operability. During auxiliary feedwater system normal use during startup, normal shutdown, and hot standby conditions, the turbine-driven auxiliary pump feedwater control valves are maintained normally fully open.	

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		During startup, normal shutdown, and hot standby conditions, the motor-driven auxiliary feedwater pumps may be aligned to take suction from the non-safety grade condensate storage tank (CST). Motor-driven auxiliary feedwater pump suction automatically switches to the demineralized water storage tank (DWST), including isolation from the CST, in the event of an SIS, LOP, CDA, two of four low-low water level condition in any one steam generator, or AMSAC signal. Relative to a low-low SG level condition operator action to realign the motor-driven auxiliary feedwater pump suction source may be required in this mode of operation. 3. Regulatory Position C.3 Switches for manual initiation are located in the control room in such a manner as to permit deliberate expeditious action by the operator. 4. Regulatory Position C.4 a. Equipment common to both manual and automatic initiation will be minimized. Where manual and automatic action sequencing functions and interlocks that contribute to the protective action are common, component or channel level initiation will also be provided in the control room. b. Manual initiation portions of the protection system meet the single failure criterion. c. Manual initiation portions of the protection system will not impair the ability of the automatic system to meet the single failure criterion. 5. Regulatory Position C.6 Manual initiation portions of the protection system are designed such that once initiated, a protective action at the system level (initiation of the final action device associated with a given protective function) goes to completion.	

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		Having gone to completion (i.e., when sufficient breakers are closed or sufficient MOVs or other actuators are operated), a device shall be returned to its pre-initiation status only by deliberate operator action. This action shall be similar in nature for all protection systems. This design is in compliance with the applicable section of IEEE 279 (Paragraph 4.16). In addition, manual initiation is provided to allow the operator to take early action based on observation of plant parameters. It is not to be treated as a "backup" to automatic features. Operator actions will not be required to compensate for single failures.	
1.63	Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants (Rev. 2, July 1978)	Comply, with the following clarifications: 1. The single failure provision of R.G. 1.63 shall apply to both Class 1E and non-1E overcurrent protection devices. 2. An acceptable method of compliance with the "Single Failure Criterion" of R.G. 1.63 may be the use of redundant or backup interrupting devices. Tripping coordination between primary and backup interrupting devices is not required. 3. While satisfying the "Single Failure Criterion" in IEEE 279-1971, Section 4.2, the overcurrent protective devices are not required to comply with other criteria listed in IEEE-279-1971. 4. Unless required for other considerations, the protection schemes and fault isolating devices need not be Class 1E or seismically qualified for protection of the penetrations. Overcurrent protection devices are not within the scope of IEEE 279-1971 as written. However, those principles developed in IEEE 279, which ensure a highly reliable design will be used for guidance in the protection system design.	8.3

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
1.64	Quality Assurance Requirements for the Design of Nuclear Power Plants (Rev. 2, June 1976) [3]	1. [1] Prior to 12/14/05, Millstone 3 complied as follows: Construction - Millstone 3 complied with Rev. 0 of the Guide Operation - Millstone 3 complied with Rev. 2 of the Guide. 2. Current compliance is as described in the QAPD Topical Report.	17.1
1.65	Materials and Inspections for Reactor Vessel Closure Studs (Rev. 0, October 1973)	[1] See Section 1.8N.	5.3.1
1.66	Nondestructive Examination of Tubular Products	Withdrawn	
1.67	Installation of Overpressure Protection Devices (Rev. 0, October 1973)	[1] Comply	5.4.11.3
1.68	Initial Test Programs for Water-Cooled Nuclear Power Plant (Rev. 2, August 1978)	Millstone 3 initial startup test program complies with R.G. 1.68 with exceptions as stated in FSAR Section 14.2.7.	14.2.7
1.68.1	Preoperational and Initial Startup Testing of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants (Rev. 1, January 1977)	Not applicable. Applicable only to BWRs.	
1.68.2	Initial Startup Test Program to Demonstrate Remote Shutdown Capability for Water-Cooled Nuclear Power Plants (Rev. 1, July 1978)	Comply	14.2.7
1.68.3	Preoperational Testing of Instrument and Control Air Systems (Rev. 0, April 1982)	Millstone 3 initial startup test program complies with R.G. 1.68.3 with exceptions and clarifications as stated in FSAR Section 14.2.7.	9.3.1 14.2.7
1.69	Concrete Radiation Shields for Nuclear Power Plants (Rev. 0, December 1973)	Comply	12.3.2

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
1.70	Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (Rev. 3, November 1978)	Comply	1.1
1.71	Welder Qualification for Areas of Limited Accessibility (Rev. 0, December 1973)	(1) Comply, with the following clarification: An acceptable alternative to this position is contained in the following exception: In lieu of Paragraphs C.1 and C.2a, all applicable welds of limited accessibility are volumetrically inspected to the requirements and standards of ASME III, Class 1.	5.2.3 5.3.1.4 10.3.6.2
1.72	Spray Pond Piping Made from Fiberglass-Reinforced Thermosetting Resin (Rev. 2, November 1978)	Not applicable. Fiberglass pipe is not used for QA Category I applications on Millstone 3.	
1.73	Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants (Rev. 0, January 1974)	(1) Comply	3.11.2 8.3.1
1.74	Quality Assurance Terms and Definitions (Rev. 0, February 1974)	(1) Comply	17.1.2 17.2
1.75	Physical Independence of Electric Systems (Rev. 2, September 1978)	(1) Comply, with the following exceptions and clarifications: 1. General (Clarification Ventilated tray covers are considered equivalent to solid tray covers.	7.1 8.3.1.4

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		Short lengths of cable (generally less than 10 feet) enclosed in a protective wrap of woven silicon dioxide are considered to be protected from electrically induced problems in adjacent cables to the same degree as the same cable in an enclosed raceway. The protective wrap of woven silicon dioxide (trade name -SIL-TEMP) is 54 mils thick and is wrapped longitudinally around cable(s) with a 50% overlap to ensure that cable(s) is enclosed by one thickness of the protective wrap. Metal clad cable, type MC, utilized in low energy, 120 V AC and 125 V DC nominal, circuits and in low density applications is considered adequate protection. As such, the minimum separation between these cables and other cables, or raceway (where required) is 1 inch. These cables are further described as follows: 1. Type MC cable is a factory assembly of conductors, each individually insulated, enclosed in a metallic sheath of interlocking tape or a smooth or corrugated tube. 2. Largest conductor size number 10 AWG. 3. No more than three number 10 AWG conductors with remaining conductors of smaller size. 4. Aluminum sheath cable (a Type MC cable in which the aluminum is continuously welded) may have an overall jacket of neoprene or hypalon. 2. Position C.1 The power circuits for the non-Class 1E pressurizer heaters, control rod drive mechanism cooling fans, and containment air recirculation fans connected to Class 1E power sources are provided with two separate Class 1E breakers connected in series. In addition, the interconnecting cables (i.e., from power source to load) are identified by the same color code as the Class 1E power source to which they are connected.	

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		Power circuits for other non-Class 1E equipment connected to Class 1E power sources are provided with two separate Class 1E breakers or fuses connected in series. In addition, the interconnecting cables are identified by the same color code as the Class 1E power source to which they are connected (i.e., from power source up to and including the second breaker). In general, cable from the second breaker to the load are routed in rigid conduit, or routed in raceway of the same color as the power source. Refer to Table 8.3-3 for description of the routing of the interconnecting cable. The controlled routing (i.e., continuation of the circuit with the same color code or continuation of the circuit in rigid conduit) ensures the physical and electrical independence of the power circuit beyond the Class 1E isolation device (i.e., batter charger, isolation transformer, two series connected interrupting devices circuit breakers, fuses) or circuit breakers that trip on accident or loss-of-power signals. For those cables not routed in rigid conduit or identified with the same color beyond the isolation device, the use of a Class 1E isolation device ensures the electrical independence of the Class 1E power source to the non-Class 1E equipment. Coordination between the two series connected Class 1E breakers or fuses is not required. Coordination between the two series connected Class 1E breakers or fuses and the Class 1E main supply breaker is provided. 3. Position C.4 (Clarification) Associated circuits are identified by the same color code as the Class 1E circuit with which they are associated. This color code exists up to and including an isolation device, except as discussed under Position C.1. Associated circuits meet all other requirements of Class 1E circuits up to and including the isolation device. 4. Position C.6 (Clarification)	

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		Analyses of potential hazards in Section 5.1.1.1 of IEEE 384 are accomplished as follows: 1. The high pressure piping and missile analyses are described in FSAR Sections 3.6 and 3.5, respectively. 2. The fire protection analyses are outlined in FSAR Section 9.5.1 and the Fire Protection Evaluation Report. 3. Cable that is not flame retardant is enclosed in a dedicated raceway for the entire length of the run. 4. The building design for flooding is described in FSAR Section 3.4. Analysis for establishing minimum separation distances as allowed for Section 5.1.1.2 of IEEE 384 for use in areas defined by Section 5.1.3 and 5.1.4 of IEEE 384 is accomplished by the following: Test and analysis performed to determine the separation requirements between Class 1E and non-Class 1E is presented in Wyle Test Report No. 47506-02. On a case-by-case basis, Wyle Test Report No. 47506-02 will be utilized to justify acceptable deviations to the electrical separation criteria between Class IE circuits as well as Class IE and non-Class IE circuits. These deviations located in areas outside of panels, will be listed in Specification SP-EE-076. 5. Position C.7 (Section 4.6 of IEEE 384)	

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		Where plant arrangements preclude maintaining the minimum separation distance between Class 1E and non-Class 1E circuits, either the Class 1E or the non-Class 1E circuit shall be run in an enclosed raceway or a barrier shall be provided between the circuits. Other barriers may be installed as illustrated in Figures 2 through 5 of IEEE 384-1974. The minimum distance from barriers or between enclosed raceway and exposed circuits is 1 inch. (Exception: The copper feeders to the reactor coolant pumps and circulating water pumps will meet the separation requirements of R.G. 1.75 and IEEE 384-1974.) The minimum separation between Class 1E and non-Class 1E enclosed raceways of "X," "C," and "K" service is 1/8 inch. Position C.9 Proposed splices in raceways will be evaluated on a case-by case basis and documented in Specification SP-EE-076. 6. Position C.10 Class 1E cable and raceways shall be marked at intervals not exceeding 15 feet. The 5-ft. requirement is a typographical error which has been confirmed by the NRC. 7. Position C.12 1. Power cables that supply power to instrument rack room and control room distribution panels, limited to 120 V AC and/or 125 V DC, are: • Enclosed in rigid conduit in the cable spreading room. The rigid conduit is either aluminum or steel. • Enclosed in rigid conduit with flexible conduit at entrance to the panels in the instrument rack room and control room.	

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		2. Power cables (from the above distribution panels) to facilities serving the control room and instrument rack room, limited to 120 V AC and/or 125 V DC, are enclosed in rigid conduit except at entrance/exit to floor sleeves in the cable spreading room, instrument rack room and control room, and at entrance to equipment in the instrument rack room and control room. 3. Other power cable (4,160 V, 480 V, and 120 V AC service) that traverses the cable spreading room are enclosed in rigid steel conduit. 4. The loss of the above cables or the control room, instrument rack room, or the cable spreading room due to the design basis event fire will not compromise the capability to achieve cold shutdown as outlined in the Fire Protection Evaluation Report. 5. The Millstone 3 design utilizes a single cable spreading room. 8. Position C.16 (Section 5.6.2 of IEEE 384) 1. The minimum 6 inch separation (or a barrier) applies to spacing between exposed terminals, contacts, and equipment of redundant Class 1E circuits for testing and maintenance purposes. A minimum of 1 inch separation (or a barrier) is required between redundant wire bundles or Class 1E and non-Class 1E wire bundles. The minimum of 1 inch separation is sufficient since the control boards are protected from and/or are not subject to hazards such as external fire, flooding, high energy piping, and missiles. Internal electrical fires are not considered a hazard due to fire retardant materials and low energy application. 2. For internal to control room panels and cabinets (specifically 3CES*MCB-MB1 through MB8 and 3HVS*PNLVP1), the minimum separation distance between redundant Class 1E and non-Class 1E circuits is as follows: A minimum 1 inch separation (or a barrier) between exposed contacts or terminals.	

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		A minimum 1 inch separation (or a barrier) is required between redundant wire bundles or Class 1E and non-Class 1E wire bundles. Test and Analysis performed to determine the reduced separation requirements within control room panels (specifically 3CES*MCB-MB1 through MB8 and 3HVS*PNLVP1) for redundant Class 1E or Class 1E and non-Class 1E is presented in Wyle Test Report 46317-1.	
1.76	Design Basis Tornado for Nuclear Power Plants (Rev. 0, April 1974)	Comply	3.3.2.1
1.77	Assumptions Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors (Rev. 0, May 1974)	(1) Comply with the following exception. Regulatory Position C.3 no longer applies and has been replaced by R.G. 1.183.	15.4.2 15.4.7
1.78	Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release (Rev. 0, June 1974)	Comply, with the following clarifications:	6.4.4.2

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		The assumptions used for identifying chemicals potentially hazardous to the control room are in accordance with R.G. 1.78, dated June 1974. Chemicals not known or projected to be present within a 5 mile radius of the reactor facility are not considered in the evaluation, and no specific design features are provided for the chemicals listed in Table C-1 of the Regulatory Guide. Hazardous chemicals that are known or projected to be used, transported, or stored within 5 miles of the reactor facility are considered in the evaluation of control room habitability. If the potential buildup of a specific hazardous chemical is slow, so that the time from detection to incapacitation is greater than 2 minutes, human detecting is used as appropriate. If the potential buildup of a specific hazardous chemical exceeds the toxic limit, automatic detection and isolation, low leakage design features, and pressurization, if necessary, are provided to ensure that the control room remains habitable. In this case, specific design features are included: Manual control room ventilation isolation, coating of concrete and concrete block surfaces of control room with a suitable surface treatment to reduce leakage due to porosity, cracks, and construction joints in the control room. Sealing of all pipes, ducts, and electrical penetrations into the control room envelope. Compression seals for access doors and equipment removal hatches in the control room. In order to ensure control room habitability for design basis accidents, the following are provided: Maintenance of 0.125 inch wg positive pressure. Two tight butterfly dampers in series in each intake.	

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		These features provide the control room operator with the ability to isolate the control room. Hazards which could threaten control room habitability have been evaluated. The evaluation is consistent with the guide, and all hazards within 5 miles of the site are included. As a result of this evaluation, no commitment to provide hazardous chemical detectors has been made.	
	Preoperational Testing of Emergency Core Cooling Systems for Pressurized Water Reactors (Rev. 1, September 1975)	Millstone 3 initial startup test program complies with R.G. 1.79 Rev. 1 with exceptions as stated in FSAR Section 14.2.7.	14.2
1.80	Preoperational Testing of Instrument Air Systems	Withdrawn	
1.81	Shared Emergency and Shutdown Electric Systems for Multi-Unit Nuclear Power Plants (Rev. 1, January 1975) (3)	Not applicable. Applicable only to those plants which share emergency and shutdown electrical systems in a multi-unit plant, which Millstone 3 does not.	8.1
1.82	Sumps for Emergency Core Cooling and Containment Spray Systems (Rev. 0, June 1974)	(1) Comply, with the following clarifications: C.1 The recirculation spray pumps take suction from a single sump. The sump and strainer were designed to eliminate any credible failure mechanisms which would require installation of a redundant sump or strainer and is considered especially qualified for service and exempt from passive failure. (See Section 3.1.1.3) C.3 The strainer is made of hollow fins constructed of perforated plate which is structurally robust. The strainer fins are located above the floor, which protects them from large rolling debris. Therefore trash racks are not required. C.8 The strainer is fully submerged at the start of the spray pumps. There is no top deck, however, there is non QA cover plate over the strainer fins to protect the strainer from damage during outages.	6.2.2.2 6.2.2.4.2
1.83	Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes (Rev. 1, July 1975)	(1) Comply	5.4.2.2 16.3 / 4.4.5

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
1.84	Design and Fabrication Code Case Acceptability - ASME Section III, Division 1	(1) Comply	5.2.1.2
1.85	Materials Code Case Acceptability - ASME Section III, Division 1	(1) Comply	5.2.1.2
1.86	Termination of Operating Licenses for Nuclear Reactors (Rev. 0, June 1974)	Comply	5.8 (EROLS)
1.87	Guidance for Construction of Class 1 Components in Elevated-Temperature Reactors (Supplement to ASME Section III Code Cases 1592, 1593, 1594, 1595, and 1596) (Rev. 1, June 1975)	Not applicable. Applicable only to Elevated Temperature Reactors	
1.88	Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records (Rev. 2, October 1976) (3)	1. (1) Prior to 12/14/05, Millstone 3 complied as follows: <u>Construction</u> Millstone 3 complies with R.G. 1.88, Rev. 0, August 1974 as noted in Appendix VII of the Millstone 3 Quality Assurance Program Manual. <u>Operation</u> Millstone 3 complies with R.G. 1.88, Rev. 2, October 1976, with the exception that the records storage vault door and hardware have a 2 hour rating instead of the recommended 4 hour minimum stated in ANSI N45.2.9. This position is noted in the Quality Assurance Topical Report. 2. Current compliance is as described in the QAPD Topical Report.	17.1
1.89	Qualification of Class 1E Equipment for Nuclear Power Plants (Rev. 0, November 1974)	(1) Class 1E equipment other than that within the NSSS Scope of Supply complies with R.G. 1.89, dated November 1974, with the following clarifications:	3.11.2

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		1. Determination of the radiation dose used for qualification of Class 1E plant equipment will take into account design features, such as the location of equipment within or outside the containment, fission product cleanup of the containment atmosphere by the containment spray system, local shielding, the time period required for equipment operation and spatial location. These design features will be applied in a conservative manner to realistically determine the radiation doses to which the devices must be qualified in addition to the other environmental factors. The radiological source term as defined in R.G. 1.7 includes a conservative margin well in excess of that required to qualify Class 1E components. 2. Qualification testing of organic materials in beta radiation environments will not be required. The effect of beta radiation will be accounted for by a weighted addition of calculated gamma and beta doses and specifying a given does in rads with qualification testing in a gamma environment treated as sufficient qualification. The clarification as stated above is similar to the position taken by the IEEE.	
1.90	Inservice Inspection of Prestressed Concrete Containment Structures with Grouted Tendons (Rev. 1, August 1977)	Not applicable. Millstone 3 does not have a prestressed concrete containment structure.	
1.91	Evaluation of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plants (Rev. 1, February 1978)	Comply	2.2.3.1.1
1.92	Combination of Modes and Spatial Components in Seismic Response Analysis (Rev. 0, December 1974) (2)	(1) Comply, with the following clarification:	3.7.2

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		The combination of modes and spatial components in seismic response analysis satisfies the requirements of R.G. 1.92, Rev. 1, dated February 1976. The time history dynamic analysis uses three statistically independent (maximum correlation factor of 0.2) orthogonal ground accelerations (two horizontal and one vertical) of the prescribed earthquake input simultaneously as prescribed in R.G. 1.92, Rev. 1. Computation of structural responses due to input of three simultaneous earthquakes is more accurate and simpler than computing them for three separate earthquakes.	
1.93	Availability of Electric Power Sources (Rev. 0, December 1974)		8.1
		1. One emergency diesel generator may be inoperable for up to 14 days.	8.3
		2. Preventive, as well as corrective, maintenance will be performed during plant operation within the constraints of the appropriate Technical Specification allowed outage time.	
		3. Millstone Unit 3 station batteries No. 3 & 4 (301A-2 & 301B-2 respectively) have a 24 hour AOT (Allowable Out of Service Time).	
1.94	Quality Assurance Requirements for Installation, Inspection and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants (Rev. 1, April 1976) [3]	1. Prior to 12/14/05, Millstone 3 complied, with the following exceptions: a. Millstone 3 will comply with the requirements of ANSI N45.2.5-1974 except that correlation testing shall be performed in accordance with the applicable paragraphs of Section 6.11 of N45.2.5-1978. b. Admixture manufacturer shall submit certified test data confirming admixture complies with ASTM C260 when tested in accordance with ASTM C233. For each production lot shipped, the manufacturer shall certify that the admixture is similar to the material represented by the test data.	3.8.1.6
		2. No longer comply - 12/14/05 (QA standards are described in QAPD Topical Report.)	17.1

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
1.95	Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release (Rev. 1, January 1977)	Comply, with the following clarifications and exceptions: In accordance with R.G. 1.78, R.G. 1.95, and Section 2.2, no off site chlorine storage or transportation is close enough to the plant nor frequent enough to be considered a hazard. There is no on site chlorine that is considered a hazard under R.G. 1.95. A sodium hypochlorite biocide system is used and no specific control room design features are provided for chlorine.	6.4.4.2
1.96	Design of Main Steam Isolation Valve Leakage Control Systems for Boiling Water Reactor Nuclear Power Plants (Rev. 1, June 1976)	Not applicable. Applicable only to BWRs.	
1.97	Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident (Rev. 2, December 1980)	(1) Millstone 3 compliance with Regulatory Guide 1.97, Rev. 2, is found in specification SP-M3-IC-022 titled, Millstone 3 Design Basis to Respond to Regulatory Guide 1.97, Rev. 2.	7.4 7.5
1.98	Assumptions Used for Evaluating the Potential Radiological Consequences of a Radioactive Offgas System Failure in a Boiling Water Reactor (Rev. 0, March 1976)	Not applicable. Applicable only to BWRs.	
1.99	Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials (Rev. 1, April 1977)	(1) See Section 1.8N.	5.3
1.100	Seismic Qualification of Electric Equipment for Nuclear Power Plants (Rev. 1, August 1977)	(1) Replacement items meet the original criteria of IEEE-344-87 (Endorsed by Regulatory Guide 1.100, Rev. 2.)	
1.101	Emergency Planning for Nuclear Power Plants (Rev. 2, October 1981)	Comply	13.3
1.102	Flood Protection for Nuclear Power Plants (Rev. 1, September 1976)	Comply with the following clarification:	3.4.1

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		Position C.1.c for Incorporated Barriers states that the plant should be designed and operated to keep doors necessary for flood protection closed during normal operation. Incorporated Barriers for each service water pump cubicle includes a watertight door and an isolated sump drain line. During normal operations, the drain line of each cubicle is open and the door of one cubicle only is open. Plant procedures require isolating the sump drain line and watertight door of each cubicle on approaching severe weather. These actions are performed prior to sea level rising to the level which requires entry into the LCO for Technical Specification 3.7.6. A Technical Specification to ensure closing of the cubicle watertight doors in advance of a potential flooding event was required by Section 2.4 of the SER for MP3. Section 2.4 of the SER concluded that the guidelines of Regulatory Guide 1.102 had been met. The NRC therefore recognized that the cubicle watertight doors may be open during normal operations and found this to be acceptable.	
1.103	Post-Tensioned Prestressing Systems for Concrete Reactor Vessels and Containments (Rev. 1, October 1976)	Withdrawn	
1.104	Overhead Crane Handling Systems for Nuclear Power Plants	Withdrawn The withdrawal of Regulatory Guide 1.104 does not affect licensing commitments for the design of single-failure-proof cranes made on the basis of the Guide.	
1.105	Instrument Setpoints (Rev. 1, November 1976)	(1) Comply	7.1.2.1.9 7.2.2.2.1 7.3.1.2.6
1.106	Thermal Overload Protection for Electric Motor on Motor-Operated Valves (Rev. 1, March 1977) (3)	Comply	7.1

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
			8.3.1.1.5
1.107	Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures (Rev. 1, February 1977)	Not applicable.	
1.108	Periodic Testing of Diesel Generator Units Used as On site Electric Power Plants (Rev. 1, August 1977)	Comply, with the following clarifications and exceptions: Section C.2(a)2: Proper operation for design-accident-loading-sequence will be demonstrated under conditions as close to design as possible. Section C.2(a)9: Comply as stated in the ERRATA dated September 1977. Section C.2(d): If the number of failures in the last 20 valid tests is less than or equal to one, the test frequency is once per 31 days. If the number of failures in the last 20 valid tests is greater than or equal to two, the test frequency is once per seven days. If the number of failures in the last 100 valid tests is less than or equal to four, the test frequency is once per 31 days. If the number of failures in the last 100 valid tests is greater than or equal to five, the test frequency is once per 7 days. The above testing clarification was approved by the NRC in License Amendment 64, issued March 9, 1992. Section C2(a): These tests will be conducted at the frequency specified in the Surveillance Frequency Control Program.	8.3.1
1.109	Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10 CFR Part 50, Appendix I (Rev. 1, October 1977)	Comply	13.3.1
1.110	Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors (Rev. 0, March 1976)	Comply	

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
1.111	Methods for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water-Cooled Reactors (Rev. 1, July 1977)	Comply	2.3.5.2.3
1.112	Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water-Cooled Power Reactors (Rev. 0, April 1976)	Comply	11.2.1.3 11.3.3 15.7
1.113	Estimating Aquatic Dispersion of Effluents from Accidental and Routine Reactor Releases for the Purpose of Implementing Appendix I (Rev. 1, April 1977)	Comply	2.4.12
1.114	Guidance on Being Operator at the Controls of a Nuclear Power Plant (Rev. 1, November 1976)	Comply	13.1.3
1.115	Protection Against Low-Trajectory Turbine Missiles (Rev. 1, July 1977)	Comply, with the following clarification: The turbine missile analysis complies with the Guide to the extent that the calculated total probability hazard rate is less than 1.0E-7 per annum. This is achieved by (1) assigning a 1.0E-2 value to strike and damage probability product and (2) implement maintenances and testing program to maintain the probability of turbine failure less than 1.0E-5 per annum.	3.5.1.3.3
1.116	Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems (Rev. 0-R, May 1977)	1. (1) Prior t 12/14/05, Millstone 3 complied. 2. Current compliance is as described in the QAPD Topical Report.	17.1
1.117	Tornado Design Classification (Rev. 1, April 1978)	Comply, with the following clarification: 1. Paragraph 3 Appendix, “Structure, Systems, and Components of Light-Water-Cooled Reactors to be Protected Against Tornados.” The statement:	3.2.5

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		"3. The reactor core and individual fuel assemblies, at all times, including during refueling" Is clarified as: Protection during refueling is provided by the containment equipment hatch while the concrete missile shield blocks are removed. A probabilistic analysis demonstrated that the mean value of a tornado missile impacting the equipment hatch during refueling meets the NRC's acceptance criterion of less than or equal to 10^{-6} per year. 2. Paragraph 4.(4) Appendix, "Structure, Systems, and Components of Light-Water-Cooled Reactors to be Protected Against Tornados." The statement: "4. Systems or portions of systems that are required for...(4) mitigating the consequences of a tornado-caused PWR streamline break..." Is interpreted as: Protection of systems and components for which credit is taken in the analysis of PWR steamline break outside containment. Clarification of Paragraph 4.(4) results from telephone communication between L. P. Walker, SWEC, and G. Chipman, USNRC (AAB) June 19, 1978. Mr. Chipman, who is the author of Paragraph 4, stated that the intent of this requirement is to ensure tornado missile protection for valves, lines, water storage tanks (e.g., demineralized water storage, etc.) that are required to mitigate a steamline break outside of containment.	
1.118	Periodic Testing of Electric Power and Protection Systems (Rev. 2, June 1978) (3)	(1) Comply	7.1.2.4 8.1.6

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
1.119	Surveillance Program for New Fuel Assembly Designs	Withdrawn	
1.120	Fire Protection Guidelines for Nuclear Power Plants (Rev. 1, November 1977)	Comply, with the following clarification: Since R.G. 1.120 has been deleted from SRP 9.5.1, the provisions of BTP CMEB 9.5-1, Rev. 2, July 1981 are followed.	
1.121	Bases for Plugging Degraded PWR Steam Generator Tubes (Rev. 0, August 1976)	(1) See Section 1.8N.	5.4.2
1.122	Development of Floor Design Response Spectra for Seismic Design of Floor-Supported Equipment or Components (Rev. 1, February 1978)	Not applicable. The implementation section of R.G. 1.122 states that the guide will be used in the evaluation of construction permit applications. The mathematical methods suggested in the Guide were not applied to the original Millstone 3 design since the development of the floor design response spectra had been completed prior to the issuance of the guide in February 1978. A description of the Millstone 3 methodology, using techniques which were state-of-the-art at the time of the ARS Development follows: 1. Millstone 3 seismic design is based on three independent orthogonal components of earthquake motion as described in Section 3.7.1.1. 2. Millstone 3 ARS peaks are broadened plus and minus 15% in all cases, and the broadened peaks are bounded by vertical lines. 3. For both symmetrical and non symmetrical structures, the floor response spectra corresponding to the direction of input is used.	3.7.2.5

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
1.123	Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants (Rev. 1, July 1977)	1. (1) Prior to 12/14/05, Millstone 3 complied as follows: Construction Comply with the alternative that certain standard or non engineered items may be procured without seller qualification as described in Section 7 of the Millstone 3 Quality Assurance Program Manual. Operation Comply as described in the Quality Assurance Topical Report. 2. Current compliance is as described in the QAPD Topical Report.	17.1
1.124	Service Limits and Loading Combinations for Class 1 Linear-Type Component Supports (Rev. 1, January 1978)	Comply, with the following exceptions and additions: 1. The following paragraph should be added to Regulatory Position C.3: C.3.c. The bending stress limits F_b resulting tension and bending in structural members as specified in Appendix XVII 2214 of Section III, Div. 1, should be the smaller value of 0.66 S_y or 0.55 S_u for compact sections, 0.75 S_y or 0.63 S_u for doubly symmetrical members with bending about the minor axis, 0.6 S_y or 0.5 S_u for box-type flexural members and miscellaneous members. The paragraph added to Regulatory Position C.3 is necessary because of an apparent oversight in applying the 5/6 factor to bending stress allowables. 2. The second paragraph under Regulatory Position C.4 should be replaced with the following: However, all increases (i.e., those allowed by NF-3231.1 (a), XVII-2110 (a), and F-1370 (a) shall always be limited by XVII-2110 (b) of Section III. The critical buckling strengths defined by XVII-2110 (b) of Section III should be calculated using material properties at temperature.	3.6 5.4.14

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		The increased allowable permitted for tensile stress in bolts shall not exceed the lesser of 0.70 S_y or S_u at temperature. The increased allowable permitted for shear stress in bolts shall not exceed 0.42 S_u at temperature. The third sentence in the second paragraph of Regulatory Position C.4 prohibits the application of the increased allowables presently permitted by NF-3231.1 (a) and F-1370 (a) to Service Limits A or B for bolted connections. The danger of applying the increases presently allowed by Subsection NF has been pointed out at Subsection NF has been pointed out at Subsection NF Committee meetings. The Millstone 3 position asserts that maximum safe increased allowables are achieved by limiting bolting tensile stress to the less of 0.7 S_y or S_u at temperature and bolting shear stress to 0.42 S_u at temperature. The 0.7 S_u limit is well recognized in Section III of the Code. The average shear strength of bolting material is about 0.62 S_u according to test data, with a standard deviation of 0.033. Results indicate that the ratio of shear strength to tensile strength is independent of the bolt grade. Curves showing this appear on page 50 of Guide to Design for Bolted and Riveted Joints by J. W. Fisher. Test data are given in a paper by J. J. Wallaert and J. W. Fisher, Shear Strength of High-Strength Bolts, Journal of the Structural Division, ASCE, Volume 91, ST3, June 1965. 3. Paragraph C.5.a should be revised as follows: "The stress limits of XVII-2000 of Section III, and Regulatory Position 3 of this Guide, should not be exceeded for component supports designed by the linear elastic analysis method. These stress limits may be increased according to the provisions of NF-3231.1 (a) of Section III and Regulatory Position 4 of this Guide when effects resulting from constraint of free-end displacement and anchor motion are added to the loading combination."	

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		Loads developed by anchor motions are also deformation limited and, as such, are considered to be grouped in the same category as loads from restraint of free-end displacement. The resulting stresses are essentially of the secondary type. 4. Regulatory Position C.8 should read as follows: Supports for the "active" components that are required only during an emergency or faulted plant condition and that are subjected to loading combinations described in Regulatory Positions C.6 and C.7 should be designed within the design limits described in Regulatory Position C.5 or other justifiable design limits. These limits should be defined by the design specification and stated in the SAR, such that the function of the supported system will be maintained when they are subjected to the loading combinations described in Regulatory Positions 6 and 7. Regulatory Position C.8 is revised as shown because this section implies that the lower stress limits associated with the Design Levels A and B Service Limits must be used for any component support that serves a safety-related function during an Emergency or Faulted Plant condition. This would seem to imply that a main coolant pump support, which constitutes a passive element in the main coolant loop, would have to be designed to meet the Design, Level A and B Service Limits during an Emergency or Faulted (LOCA) plant condition. This would require that a snubber providing restraint on an RHR line would have to be designed to the Design, Levels A and B Service Limits during an Emergency or Faulted plant condition. If this is the intent, it is a severe departure from current practice. Only active components, such as valves, whose operation is required for safe shutdown during an Emergency or Faulted condition, have been required to meet design stress limits for these plant conditions. Levels C and D service limits have been considered adequate to assure pressure boundary integrity under the more severe operating conditions.	

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
1.125	Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants (Rev. 1, October 1978)	Comply	
1.126	An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification (Rev. 1, March 1978)	(1) See Section 1.8N.	4.2
1.127	Inspection of Water-Control Structures Associated with Nuclear Power Plants (Rev. 1, March 1978)	Not application. Applies to water control structures; i.e., dams, reservoirs, conveyance facilities, etc., specifically for use in conjunction with a nuclear power plant. Millstone 3 is located on Long Island Sound and, as such, does not require dams or reservoirs for water impoundment. The intake and discharge structures are Category 1 and, as such, are built to stringent design and construction requirements. Normal maintenance during the course of operation of the plant would detect any abnormal conditions in these structures.	9.2.1
1.128	Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plant (Rev. 1, October 1978) (3)	Comply	8.1.6 8.3.2
1.129	Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants (Rev. 1, February 1978) (3)	Comply, except the battery discharge test frequency, which is controlled by the Surveillance Frequency Control Program - Note: This Reg. Guide endorses IEEE 450-1975 with an additional requirement that the “Service Test” be performed in addition to the “Performance Discharge Test” during refueling outages. The Technical Specification basis references IEEE Std. 450-1980 for battery capacity test procedures and schedule. Sections 5 and 6 of 450-1980 replace 450-1975. Guidance on bypassing weak cells, if required, is in accordance with section 7.4 of IEEE 450-2002. The balance of 450-1975 applies to MP3.	8.1.6 8.3.2

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
1.130	Design Limits and Loading Combinations for Class 1 Plate-and-Shell-Type Component Supports (Rev. 1, October 1978)	Comply, with the following clarifications and exceptions: 1. Regulatory Position C.3 should be revised as follows: "Service limits for component supports designed by linear elastic analysis should always be limited by the critical buckling strength. The critical buckling strength should be calculated using material properties at temperature. Conservative factors of safety for flat plates and for shells should be maintained for each design and service limit. The allowable stress for Service Limit D should not exceed two-thirds of the critical buckling stress." 2. Regulatory Position C.7 should read as follows: "Support for 'active' components that are required only during an emergency or faulted plant condition and that are subjected to loading combinations described in Regulatory Position C.5 and C.6 should be designed within the design limits described in Regulatory Position C.4 or other justifiable design limits. These limits should be defined by the design specification and stated in SAR, such that the function of the supported system will be maintained when they are subjected to the loading combinations described in Regulatory Positions C.5 and C.6." Regulatory Position C.7 implies that the lower stress limits associated with the Levels A and B Service Limits must be used for any component support that serves a safety related function during an Emergency or Faulted (LOCA) plant condition. This would seem to imply that a main coolant pump support, which constitutes a passive element in the main coolant loop, would have to be designed to meet the Levels A and B Service Limits during an Emergency or Faulted plant condition. If this is the intent, it is a severe departure from current practice. Only active components, such as valves, whose operation is required for safe shutdown during an Emergency or Faulted condition, have been required to meet design stress limits for these plant conditions. Levels C and D Service Limits have been considered adequate to assure pressure boundary integrity under the more severe operating conditions.	

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
1.131	Qualification Tests of Electric Cables, Field Splices, and Connections for Light-Water-Cooled Nuclear Power Plants (Rev. 0, August 1977)	Comply	
1.132	Site Investigations for Foundations of Nuclear Power Plants (Rev. 1, March 1979)	Comply with the following exceptions and clarifications: 1. Paragraph B.5: Surveys of horizontal deviation are made in all boreholes that are used for cross-hold seismic tests. The requirements for surveys of vertical deviation appear to be in error. Horizontal deviations must be measured in borings used for cross-hole seismic surveys in order to determine the true distance between energy source and receiver. 2. Paragraph C.1, Item (5): Only typical time-distance plots should be included. Time-distance plots are not usually included on geologic profiles. A typical time-distance plot is more appropriate. 3. Paragraph C.2: Test results of field permeability tests and borehole logging should be tabulated and/or graphed. Field and laboratory test results are presented in tables and figures especially designed for these tests. Boring logs do not typically include field and laboratory test results. 4. Paragraph C.3: Measurement of water or drilling mud levels in borings not required in all cases. Water or drilling mud levels in some materials, such as clays, may give false information about groundwater levels. A sufficient number of observations wells monitor groundwater levels.	2.5

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		5. Paragraph C.6: Continuous undisturbed samples will be taken in compressible or normally consolidated clays only if required for geotechnical analysis. The need for continuous undisturbed samples is a matter of engineering judgment and is evaluated for each case. 6. Appendix C: Spacing and depth requirements for borings under structures and pipelines are not in conformance for all cases. Borings for structures and pipelines were performed prior to issuance of the regulatory guide. It is considered that sufficient data are available concerning foundation conditions, bedrock quality, and bedrock contours from existing borings and the extensive geologic bedrock mapping program. 7. Paragraph B.2s: No explorations have been conducted off site. The area surrounding the Millstone site has been mapped by the U.S. Geological Survey and the Connecticut State Geological and Natural History Survey. The closest fault shown on any of these maps is approximately 10 miles northeast of the site in the Uncasville quadrangle. Detailed mapping of the 5 mile radius could uncover faults similar to those uncovered during the excavations of Millstone 3. However, detailed investigations at the Millstone site have demonstrated the incapability of these faults. Studies performed at other sites have verified that no capable faults are known to exist in New England.	
1.133	Loose-Part Detection Program for the Primary System of Light-Water-Cooled Reactors (Rev. 1, May 1981)	Comply, with the following exceptions: 1. Position C.1.g	

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		The loose-part detection system need not be qualified to an OBE, however, the equipment inside containment has been demonstrated to be functional following an OBE. OBE qualification is in excess of the requirements placed on other alarms of equal importance in disclosing failures during plant operation. 2. Position C.5.a The location of the required sensors is not needed in the Technical Specifications since their locations are delineated in Section 4.4.6.4 of MNPS-3 FSAR. 3. Position C.5.c Calibration will be verified at least once per fuel cycle or every 18 months, whichever is greater. This calibration period is adequate because the loose-part detection system is not safety-related.	
1.134	Medical Evaluation of Nuclear Power Plant Personnel Requiring Operator Licenses (Rev. 1, March 1979)	Comply	
1.135	Normal Water Level and Discharge at Nuclear Power Plants (Rev. 0, September 1977)	Comply, with the following clarification: R.G. 1.59, Rev. 2, is used to determine initial water levels for design basis flood analysis.	
1.136	Material for Concrete Containments (Rev. 2, June 1981)	Not applicable. The following Regulatory Guides have been incorporated into R.G. 1.136, Rev. 2, and have been withdrawn: • 1.10 Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures. • 1.15 Testing of Reinforcing Bars for Category I Concrete Structures.	

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		• 1.18 Structural Acceptance Test for Concrete Primary Reactor Containments. • 1.19 Nondestructive Examination of Primary Containment Liner Welds. • 1.55 Concrete Placement in Category I Structures. • 1.103 Post-Tensioned Prestressing Systems for Concrete Reactor Vessels and Containments. Withdrawal of the above Guides does not alter any prior existing licensing commitments based on their use.	
1.137	Fuel-Oil Systems for Standby Diesel Generators (Rev. 1, October 1979)	Comply with the following clarifications and exceptions: The Millstone 3 design provides approximately 3-day diesel fuel storage tanks which are interconnected with normally closed valves. The interconnected tanks provide approximately a 6-day supply of fuel oil for either of the diesel generators. This was approved by the NRC in License Amendment No. 97 issued October 17, 1994. The EDG Fuel Oil System does not meet the recommended configuration in Section 6.3 of ANSI N159 concerning strainers. A single type strainer is in the discharge of the A and B pumps and no strainer is in the discharge of the C and D pumps. Procedural controls and sampling from the TK1A and TK1B tanks assure that fuel oil quality meets or exceeds the EDG manufacturer's acceptance criteria. In addition, the following exception is taken: Analysis of the fuel properties listed in the applicable specifications are completed within 30 days after fuel addition rather than 2 weeks. Section C.1.e(2): The requirement of “pressure testing of the fuel oil system to a pressure 1.10 times the system design pressure at 10 year intervals...” has been deleted. This was approved by the NRC in License Amendment 110 issued May 1, 1995.	9.5.4

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
1.138	Laboratory Investigations of Soils for Engineering Analysis and Design of Nuclear Power Plants (Rev. 0, April 1978)	Comply, with the following clarifications and exceptions: 1. Paragraph C.1.c: Standards used to calibrate laboratory test equipment are of a known higher accuracy than the test equipment, rather than four times more accurate than the working instrument. Standards used to calibrate geotechnical laboratory equipment need only be more accurate, rather than at least four times more accurate, than the working instrument. Soils and rocks are materials whose properties vary widely within the same deposit or formation. In addition, certain physical properties of soils and rocks are greatly affected by sampling and by preparation for testing in the laboratory. The geotechnical engineer takes these natural variations and sampling/preparation effects into account, and exercises considerable judgment in assigning the material properties to be used in an analysis. (This differs from manufactured materials, where dimensions and physical properties are maintained within a narrow range by the manufacturing process.) Calibration of geotechnical laboratory equipment to higher standards than currently in use is not justified. 2. Paragraph C.1.d: Index and classification tests are not performed on all soil and rock samples. Classification of soil and rock samples is performed by visual manual techniques. Index and classification tests are performed on representative samples to confirm the visual-manual classifications. 3. Paragraph C.2: Moisture seals are not periodically checked, but are renewed as needed.	2.5

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		Tube samples are inspected for obvious leakage when they are received. The moisture seals are inspected in detail when a tube has been selected for testing. Each sample is examined when it is extruded, and any evidence of drying in the tube is noted on the sample description log. This procedure is sufficient to evaluate whether drying has occurred. Samples that appear to have dried are not tested. Periodic inspection and replacement of moisture seals would be time consuming and would not provide any better protection against testing samples whose water content has changed. 4. Paragraph C.2: Duration of storage is not specifically reported for each test. The duration of storage can be calculated from the boring logs, where the sampling date is given, and the laboratory test data sheets, where the date of testing is reported. Therefore, it is unnecessary to make a separate listing of storage time. 5. Paragraph C.3.a: Classification tests are not performed on every undisturbed test specimen of soil or rock. Visual-manual techniques are the primary means of classifying soil and rock samples. Classification tests are performed on representative samples as necessary to confirm the results of the visual-manual classifications. Therefore, it is not necessary to perform classification tests on all undisturbed samples have changed during shipment, storage, and handling. 6. Paragraph C.3.2: Measurements and control tests are not performed to determine whether undisturbed samples have changed during shipment, storage, and handling.	

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		Undisturbed samples are visually inspected as they are opened and extruded, to determine whether there has been any change in sample length within the tube or if there are any signs of sample disturbance. Results of these inspections are reported on the sample description log. This procedure and examination of the laboratory test results for possible indications of sample disturbance, is sufficient to determine whether sample disturbance has occurred. Therefore, the sample does not have to be measured, weighted, or otherwise tested during shipment, storage, and handling. 7. Paragraph C.3.b: A discussion of the validity of results on scalped materials is not presented as part of the laboratory test data. The laboratory test results indicate which portion of the sample has been scalped. Reasons for expecting test results to be valid are not given as part of the laboratory test data. A competent reviewer of the test results would understand the effect of scalping on the results of specific tests. 8. Paragraph C.4.A (2): Results of tests with B-values less than 0.95 may be used for analysis in certain cases. For very stiff or hard clays, it may not be possible to achieve a B-value of 0.95. If results of such a test are used, the B-value is reported and the probable effect on results of the test could be evaluated by a competent reviewer. 9. Paragraph C.5.a: Soil and rock identifications and descriptions are not documented so that an independent review can be performed. The only way to provide for independent review of all soil and rock descriptions is to provide the samples to the reviewer, without testing them. This is obviously impossible. The current practice of recording sample descriptions and determining index properties of representative samples is consistent with good engineering practice. 10. Paragraph C.5.a: Anomalous test data are not reported if they are caused by sample disturbance or equipment malfunction.	

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		Anomalous test data caused by sample disturbance or equipment malfunction are not reported because the data do not reflect the true properties of the material in the field. However, records of such tests are maintained as part of the laboratory records. 11. Appendix B, Relative Density: The frequency of the vibratory table is not adjusted. The frequency of the vibratory table cannot be adjusted, because it depends upon the fixed frequency of the input current.	
1.139	Guidance for Residual Heat Removal (Rev. 0, May 1978)	(1) Comply	3.1.2.34
1.140	Design, Testing, and Maintenance Normal Ventilation Criteria for Exhaust System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants (Rev. 1, October 1979)	Comply, with the following clarifications and exceptions: Paragraph C.2.c The Technical Support Center filter train is instrumented to indicate, but not to alarm, filter flow within the habitability zone. The containment air filtration system is not instrumented to monitor or alarm filter flow. Surveillance procedures verify the flow rates of these two infrequently operated filter trains once per refueling interval. Paragraph C.2.f: Housing leak tests are performed in accordance with the provisions of Section 6 of ANSI N510-1980. A leak rate of 1.0% in accordance with Table 4-3 of ANSI N509-1980 is acceptable. However, ductwork leak tests shall be performed in accordance with the procedures delineated in Chapter 8, Leak Testing, of the Manual for the Balancing and Adjustment of Air Distribution System published by SMACNA (First Ed., dated 1967). Paragraph C.3.b (Clarification): The Technical Support Center filter has been purchased to ANSI N509-1980, while the containment filters have been purchased to ANSI N509-1976. Filter media will be subjected to velocities recommended by the HEPA filter manufacturer which exceeds ANSI N509-1976 or 1980 requirements, as applicable, given in Section 4.3.1.	9.4

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		The HEPA filter cell testing is conducted initially at the manufacturer's facilities and again after installation at the plant site. All HEPA filters furnished are equipped with the face guards in accordance with MIL-F-51068. When installed in the filter housing, the HEPA filters and housing are inspected for defects and tested for leak-tightness in accordance with ANSI N510-1980. Paragraph C.3.c: For HEPA filters and adsorber mountings, the requirements of ANSI N509-1976 or 1980, as applicable, Section 5.6.3, will be complied with except for the tolerance requirements. The tolerances for HEPA filters and adsorber mounting frames is sufficient to pass the bank leak tests of Paragraphs 5.c and 5.d of the Guide. Paragraph C.3.e: Exception is taken to the recommendations of Section 4.5.8 of ERDA 76-21 relative to drain sizes and arrangement. All drains are capped, and no permanent deluge system is provided. Millstone 3 complies with ANSI N509-1980 paragraph 4.6.2.2 with respect to designing inlet units and components which can be isolated from the fan to withstand a peak negative pressure by ensuring that such isolation is precluded via the design control logic between the fans and the inlet dampers. Compliance with designing inlet units and components, as noted in the same paragraph with respect to the plugging of such components, is demonstrated via routine surveillance and subsequent filter replacement as necessary. Paragraph C.3.f: Exception is taken to Section 5.10.3.5 of ANSI N509-1976 or 1980, as applicable; ductwork, as a structure, will have a resonant frequency above 25 Hz, but this may not be true for the unsupported plate or sheet sections. Ductwork testing shall be subject to the limitations of Paragraph C.2.f above. Paragraph C.3.g: The charcoal for the Technical Support Center filters is purchased in accordance with the requirements of Regulatory Guide 1.52, Paragraph C.3.i.	

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		Paragraph C.3.1 Exception is taken to the provisions in Section 5.9 of ANSI N509-1976 of designing dampers to ANSI B31.1 and to using butterfly valves. The non-ESF filtration systems are the containment air filtration system and the Technical Support Center filtration system. The first system is an internal recirculation system consisting of two 50% capacity air cleaning trains. Damper leakage will not impact on the air cleaning effectiveness of this system. The second system filters outside supply air before introducing it into the Technical Support Center. The intake damper will be open during system operation, and the discharge damper will leak filtered air into the room, which will not impact on air cleaning effectiveness of the system. Paragraph C.3.m The Technical Support Center filter is designed, constructed, and tested to Section 5.3 of ANSI N509-1980. Paragraph C.4.a Exception is taken to full compliance with Section 2.3.8 of ERDA 76-21; i.e., no communication system is used, decontamination areas and showers are not "nearby," filters are not used at duct inlets, and duct inspection hatches are not provided. Additionally, the Technical Support Center filter complies with Section 4.7 of ANSI N509-1980 rather than 1976. Paragraph C.4.b Partial compliance, with a minimum spacing between filter frame of 2 ft.-6 in. instead of a minimum of 3 feet. This is deemed adequate since replacement of containment filter elements would be minimal due to system function, use, and location. The Technical Support Center filter does not comply with this requirement since all components of the filter are easily accessible from outside the filter. Paragraph C.4.c	

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

<table>
<tr><th>R.G. No.</th><th>Title</th><th>Degree of Compliance</th><th>FSAR Section Reference</th></tr>
<tr><td></td><td></td><td>
The Technical Support Center complies with Section 4.11 of ANSI N509-1980.
Paragraph C.5.a
Visual inspection is performed in accordance with ANSI N510-1980.
Paragraph C.5.b
Test for air flow distribution to the HEPA filters and the adsorbers shall be conducted in accordance with Section 8 of ANSI N510-1980 with the following exceptions and clarifications.
Paragraph 8.3.1.6 - Test shall be conducted to verify the following pressure drop values across the combined HEPA filter and charcoal adsorber banks and across the entire filter housing.
<table>
<tr><th>System</th><th>ΔP Across HEPA & Charcoal (in. wg)</th><th>ΔP Across Housing (in. wg)</th></tr>
<tr><td>Containment Air Filters</td><td>5.94</td><td>7.54</td></tr>
<tr><td>Technical Support Center Air Filters</td><td>5.25</td><td>6.45</td></tr>
</table>
Paragraph 8.3.1.7 - Exception is taken to the requirement for testing at 50% of the pressure drop valve. Testing the filter at this condition will require the removal of certain filter components with potential damage due to mishandling. Above condition is not considered a design parameter of the system.
Paragraph 8.3.2.3 - Air distribution test across the prefilter and moisture separator is not required by the project specification.
Paragraph C.5.c
HEPA filter DOP testing is conducted in accordance with ANSI N510-1980.
Paragraph C.5.d
</td><td></td></tr>
</table>

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		Charcoal adsorber leak testing with refrigerant is conducted in accordance with ANSI N510-1980.	
1.141	Containment Isolation Provisions for Fluid Systems (Rev. 0, April 1978)	(1) Comply, with the following exception: On the containment recirculation system, the only closed system outside containment, vent/drain valves and branch connections are not lock closed.	5.4.5 5.4.6 6.2.4
1.142	Safety Related Concrete Structures for Nuclear Power Plants (Other than Reactor Vessels and Containments) (Rev. 0, April 1978)	Not applicable. The following Regulatory Guides have been incorporated into R.G. 1.142, Rev. 0, and have been withdrawn: • 1.10 Mechanical (Cadweld) Splices in Reinforcing Bars of Category 1 Concrete Structures. • 1.15 Testing of Reinforcing Bars for Category 1 Concrete Structures. • 1.55 Concrete Placement in Category 1 Structures. • 1.103 Post-Tensioned Prestressing Systems for Concrete Reactor Vessels and Containments.	
1.143	Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants (Rev. 1, October 1979)	1. Prior to 12/14/05, Millstone 3 complied, with the following clarifications and exceptions: Section C.1.1.1 - All demineralizers have been designed, procured, and manufactured in accordance with the ASME Section VIII Boiler and Pressure Vessel Code However, overpressurization protection was not provided in accordance with ASME VIII for 3LWS-DEMN1 and 3LWS-DEMN2. These demineralizers have been evaluated and determined that over pressurization is not considered a credible failure. Section C.1.1.2 - Pipe and pipe fittings (valves are excluded) can be procured to an ASTM specification with the basis of acceptance of each item as indicated below. The item shall be manufactured without welding.	3.1.2.63 11.2 11.3 11.4

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		1. Each item is marked with the applicable ASTM material specification by the original manufacturer. No material manufacturer's Certificate of Conformance (C of C), as defined in ANSI N45.2.10-1973, is required. Only markings from the original manufacturers are acceptable. 2. Unmarked items with a Certificate of Conformance to the ASTM material specification are acceptable if shipped directly from the original manufacturer. 3. Unmarked items received with no certification are non-destructively tested to ascertain the material of the item. Additional testing, if required, is specified by the project engineer. Section C.1.2.1 - Tank high level alarms are provided either locally or in the control room. Tank overflow is precluded by administrative controls where alarms are not provided in the control room, filling is monitored at local control which has high level alarms. Section C.1.2.5 - Rather than dikes or retention ponds to retain overflows, certain outside tanks are designed to have overflows piped directly to the waste disposal building sump and ultimately to the liquid radioactive waste system. Sections C.2.1.3 and C.5.1 - Portions of the gaseous waste system which store gaseous radioactive wastes have not been designed to seismic criteria. Rather, the building housing in the gaseous waste system is seismically designed in accordance with procedure set forth in FSAR Section 3.7B and releases from the buildings can be filtered by charcoal filters on a high radiation signal. Section C.4.3 - instead of butt-welded joints, belled end socket-welded fittings are used for some of the radioactive waste management system piping of nominal size 2.5 to 4 inches. Installation of this piping was limited to liquid waste management systems (LWS, LWC, BRS) piping located outside of the Waste Disposal Building.	

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		Section C.5.2 - The seismic analyses of the buildings housing the radwaste systems use the Millstone 3 method of seismic analysis of Category 1 buildings (FSAR Section 3.7B) and do not use the data in R.G. 1.60 and R.G. 1.61 (see compliance for these Guides in this section). 2. Current compliance is as described in the QAPD Topical Report	17.1
1.144	Auditing of Quality Assurance Programs for Nuclear Power Plants (Rev. 1, September 1980)	1. (1) Prior to 12/14/05, Millstone 3 complied as follows: <u>Construction</u> Millstone 3 complies with ANSI N45.2.12-1977, as implemented by Appendix VII of the Millstone 3 Quality Assurance Program Manual. <u>Operation</u> - Comply 2. Current compliance is as described in the QAPD Topical Report.	2.3 17.1
1.145	Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants (Rev. 0, August 1979)	Comply	13.3
1.146	Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants (Rev. 0, August 1980)	1. (1) Prior to 12/14/05, Millstone 3 complied. 2. Current compliance is as described in the QAPD Topical Report.	17.1
1.147	Inservice Inspection Code Case Acceptability ASME Section XI Division 1 (Rev. 0, February 1981)	Comply	
1.148	Functional Specification for Active Valve Assemblies in Systems Important to Safety in Nuclear Power Plants (Rev. 0, March 1981)	The majority of active valves used at Millstone 3 were purchased prior to the issuance of Regulatory Guide 1.148. However, Millstone 3 meets the intent of Regulatory Guide 1.148 in that all active valves relied upon to perform a safety function are designed and analyzed to ensure their structural integrity and operability during the transients or events considered in the respective operating condition categories.	

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		The overall design process includes systems design, valve specifications, and quality assurance procedures. Many of the functional requirements are included in the systems design which dictates the type of valve required for the system application. Additionally, the NRC conducted a Pump and Valve Operability Review Team Audit in March 1985 following which the NRC concluded that the Millstone 3 valve operability program met the intent of Regulatory Guide 1.148.	
1.149	Nuclear Power Plant Simulators for Use in Operator Training (Rev. 3, October 2001)	Millstone 3 complies with R.G. 1.149, Rev.3.	13.2
1.150	Ultrasonic Test of Reactor Vessel Welds During Preservice and Inservice Examinations (Rev. 1, February 1983)	Comply, per response to NRC Question Q250.3.	5.2.4
1.152	Criteria for Digital Computers in Safety Systems of Nuclear Power Plants (Rev. 1, January 1996)	Compliance is as described in the QAPD Topical Report.	1.7
1.155	Station Blackout	Comply	8.3.1.1.5
1.163	Performance-Based Containment Leak-Test Program	Comply	
1.183	Alternative Radiological Source Terms for Evaluating Design Basis Accidents at Nuclear Power Reactors (Rev. 0, July 2000)	Complies with the following exceptions: Section 3.2, Table 3, footnote 11 - Non-LOCA Fraction of Fission Product Inventory in Gap – for the fuel handling accident, the fraction of the fuel rods exceeding the criteria of footnote 11 that would bound projected loading plans and operating strategies were modeled with the gap fractions listed in Regulatory Guide 1.25 (as modified by the direction of NUREG/CR-5009) instead of values from Table 3. App. A, Section 3.7– for control room dose analysis, the Technical Specification Containment leak rate is reduced by 50% after 1 hour versus 24 hours listed in the Reg. Guide. 24 hours is used for off site dose analysis.	6.5.2 11.1.1 15

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		Appendix E and Appendix F, Sections 5.1 – Transport – for the main steam line break, steam generator tube rupture, locked rotor accident, and rod ejection accident Regulatory Guide 1.183 requires primary-to-secondary leakage to be modeled as the leak rate limiting condition for operation specified in the technical specifications. The Regulatory Guide 1.183 requirements predates the Steam Generator Program, which currently limits primary-to-secondary operational leakage to RCS LCO 3.4.6.2 and requires that accidents be modeled with that accident induced leakage (Technical Specification 6.8.g.b.2). The accident induced leakage is consistent with the leak rate limiting condition for operation specified in the technical specifications prior the advent of the Steam Generator Program.	
1.194	Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants (Rev. 0, June 2003)	Comply	2.3.4
1.196	Control Room Habitability at Light–Water Nuclear Power Reactors (Rev. 0, May 2003)	Comply, with the following exceptions and clarifications: 1. RG 1.196 calls for evaluating, testing and maintaining the Control Room Emergency Filtration System per RG 1.52 Rev. 3. Millstone 3 complies with RG 1.52 Rev. 2, with the clarifications and exceptions as noted in this table. 2. RG 1.196 calls for performing an evaluation, and also periodic assessments, of the impact of hazardous chemical release on control room habitability using the methodology of RG 1.78 Rev. 1 (which replaced the previous revisions of RG 1.78 for hazardous gas release and RG 1.95 for chlorine release). Millstone 3 complies with RG 1.78 Rev. 0 and RG 1.95 Rev. 1, with the clarifications and exceptions as noted in this table. 3. RG 1.196 calls for determining control room radiological dose per RG 1.183 Rev. 0 for plants employing alternative source term methodology. Millstone 3 complies with RG1.183 Rev. 0, with the exceptions as noted in this table.	6.4.2 6.4.2

TABLE 1.8-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
1.197	Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors (Rev. 0, May 2003)	Comply, with the following exceptions and clarifications: 1. With respect to Position C.1 for performance of an integrated Control Room Envelope (CRE) inleakage test: • Appropriate application of ASTM E741 shall include the ability to take minor exceptions to the test methodology. These exceptions shall be documented in the test report. 2. With respect to Position C.2 for establishing the licensing basis of the CRE and developing compensatory actions in the event of excessive CRE inleakage: • Vulnerability assessments for radiological, hazardous chemical and smoke, and emergency ventilation system testing were completed as documented in the UFSAR and other licensing basis documents. The exceptions to the Regulatory Guides referenced in RG 1.196 (i.e., RG 1.52, RG 1.78 and RG 1.183), which were considered in completing the vulnerability assessments, are documented in the UFSAR current licensing basis.	

NOTE:

1. See FSAR Section 1.8N For Nuclear Steam Supply System (NSSS) scope of compliance for Millstone 3.
2. Later revisions NA - Millstone 3 is exempted by the implementation sections of the later revisions of the Guide.
3. Millstone 3 addresses a later revision of the Guide than is required by the implementation section of the Guide.

1.8N NSSS CONFORMANCE TO NRC REGULATORY GUIDES

Table 1.8N-1 lists the NRC Division 1 Regulatory Guides that were in effect during the time of application for an Operating License. It identifies applicable FSAR sections, and indicates the NSSS scope of compliance for Millstone 3.

TABLE 1.8N-1 NRC REGULATORY GUIDES

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
1.1	Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps (Rev. 0, November 2, 1970	In predicting the NPSH available, Westinghouse assumes that the vapor pressure of the liquid in the sump is equal to the containment pressure, i.e., that the liquid is at a saturated condition, in calculating the available NPSH for the recirculation mode. This assumption meets the intent of Regulatory Guide 1.1, since no credit is taken for any increase in containment pressure. It is also a reasonable assumption since the containment is a closed thermodynamic system and will remain at an equilibrium condition.	6.2.2, 6.3
1.2	Thermal Shock to Reactor Pressure Vessels (Rev. 0, November 1970)	ISSUE: General Design Criterion 35 specifies design and operating conditions necessary to assure that the reactor coolant pressure boundary will behave in a nonbrittle manner. To provide protection against loss-of-coolant accidents, present designs provide for the injection of large quantities of cold emergency coolant into the reactor coolant system. The effect on the reactor pressure vessel of this cold water injection is of concern because the reactor vessel is subjected to greater irradiation than other components of the reactor coolant pressure boundary and, thus, has a greater potential for becoming brittle. Regulatory Guide 1.2 describes a suitable program which may be used to implement General Design Criterion 35 to assure that the reactor pressure vessel will behave in a nonbrittle manner under loss-of-coolant accident conditions. NUCLEAR SAFETY POSITION: Westinghouse follows all recommendations of the guide. The guide Position C.1 is followed by Westinghouse's own analytical and experimental programs, as well as by participation in the Heavy Section Steel Technology (HSST) Program at Oak Ridge National Laboratory.	5.2.3.3

TABLE 1.8N-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		Under the Heavy Section Steel Technology Program, a number of steel pressure vessels containing carefully prepared and sharpened surface cracks have been tested. Test specimens have been subjected to either hydraulic internal pressure loadings or thermal shock loadings. This program has validated analytical fracture mechanics techniques and has demonstrated quantitatively the margin of safety inherent in reactor pressure vessels. Additional testing will be done to demonstrate the behavior of vessels under combined pressure and thermal shock loadings. Details of progress and results obtained in the HSST program are available in the Heavy Section Steel Technology Program Semiannual (quarterly, beginning in 1974) Progress Reports issued by Oak Ridge National Laboratory. Westinghouse is continuing to obtain fracture toughness data for reactor pressure vessel steels through internally funded programs, as well as industry sponsored work. Regulatory Position C.2 is followed inasmuch as no adverse changes have been made in approved core or reactor designs. A change to low leakage loading pattern core designs has been made on some plants. This change reduces the rate irradiation of the pressure vessel and is, therefore, beneficial. Regulatory Position C.3 is followed since the vessel design does not preclude the use of an engineering solution to assure adequate recovery of the fracture toughness properties of the vessel material. If additional margin is needed, the reactor vessel can be annealed. This solution was shown to be feasible by EPRI program RP1021-1, “Feasibility and Methodology for Thermal Annealing an Embrittled Reactor Vessel.”	

TABLE 1.8N-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		Westinghouse is continuing to develop analytical capability for reactor vessel integrity evaluations. A conservative generic evaluation was performed in 1981 under the Westinghouse Owners Group for all domestic operating plants. This evaluation, summarized in WCAP-10019, "Summary Report on Reactor Vessel Integrity for Westinghouse Operating Plants," confirms that no near-term safety concern exists. Acceptability through end-of-life was not demonstrated by this report for all plants, and additional plant specific evaluations are being performed to remove some of the excessive conservatisms inherent in the generic report. Additional development work is also being performed in response to NRC questions arising from their review of this issue under Task Action Plan A-49.	
1.7	Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident (Rev. 2, November 1978)	Westinghouse will conform to all assumptions established in Regulatory Guide 1.7 and thereby comply with General Design Criterion 41.	6.2.5
1.13	Spent Fuel Storage Facility Design Basis (Rev. 1, December 1975)	Comply.	9.1.2
1.14	Reactor Coolant Pump Flywheel Integrity (Rev. 1, August 1975)	Comply, with the following exceptions: 1. <u>Post-Spin Inspection</u>	5.4.1.1

TABLE 1.8N-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		Westinghouse has shown in WCAP-8163, "Topical Report Reactor Coolant Pump Integrity in LOCA," that the flywheel would not fail at 290 percent of normal speed for a flywheel flaw of 1.15 inches or less in length. Results for a double ended guillotine break at the pump discharge with full separation of pipe ends assumed, show the maximum overspeed to be less than 110 percent of normal speed. Even with an assumed instantaneous loss of power to the reactor coolant pump, the maximum overspeed was calculated in WCAP-8163 to be about 280 percent of normal speed for the same postulated break. In comparison with the overspeed presented above, the flywheel is tested at 125 percent of normal speed. Thus, the flywheel could withstand a speed up to 2.3 times greater than the flywheel spin test speed of 125 percent provided that no flaws greater than 1.15 inches are present. If the maximum speed were 125 percent of normal speed or less, the critical flaw size for failure would exceed 6 inches in length. Nondestructive tests and critical dimension examinations are all performed before the spin tests. The inspection methods employed (described in WCAP-8163) provide assurance that flaws significantly smaller than the critical flaw size of 1.15 inches for 290 percent of normal speed would be detected. Flaws in the flywheel will be recorded in the prespin inspection program (see WCAP- 8163). Flaw growth attributable to the SPIN test (i.e., from a single reversal of stress, up to speed and back), under the most adverse conditions, is about three orders of magnitude smaller than what nondestructive inspection techniques are capable of detecting. For these reasons, Westinghouse performs no post-spin inspection and believes that prespin test inspections are adequate. 2. Interference Fit Stresses and Excessive Deformation	

TABLE 1.8N-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		Much of Revision 1 deals with stresses in the flywheel resulting from the interference fit between the flywheel and the shaft. Because Westinghouse's design specifies a light interference fit between the flywheel and the shaft, at zero speed, the hoop stresses and radial stresses at the flywheel bore are negligible. Centering of the flywheel relative to the shaft is accomplished by means of keys and/or centering devices attached to the shaft, and at normal speed, the flywheel is not in contact with the shaft in the sense intended by Revision 1. Hence, the definition of "Excessive Deformation," as defined in Revision 1 of Regulatory Guide 1.14, is not applicable to the Westinghouse design since the enlargement of the bore and subsequent partial separation of the flywheel from the shaft does not cause unbalance of the flywheel. Extensive Westinghouse experience with reactor coolant pump flywheels installed in this fashion has verified the adequacy of the design. Westinghouse's position is that combined primary stress levels, as defined in Revision 0 of Safety Guide 14 (C.2 (a) and (c)) are both conservative and proven and that no changes to these stress limits are necessary. Westinghouse designs to these stress limits and thus, does not have permanent distortion of the flywheel bore at normal or spin test conditions. 3. Section B, Discussion of Cross Rolling Ratio of 1 to 3 Cross Rolling Ratio - Westinghouse's position is that specification of a cross rolling ratio is unnecessary since past evaluations have shown that ASME SA-533-B Class 1 materials produced without this requirement have suitable toughness for typical flywheel applications. Proper material selection and specification of minimum material properties in the transverse direction adequately ensure flywheel integrity. An attempt to gain isotropy in the flywheel material by means of cross rolling is unnecessary since adequate margins of safety are provided by both flywheel material selection (ASME SA- 533-B Class 1) and by specifying minimum yield and tensile levels and toughness test values taken in the direction perpendicular to the maximum working direction of the material. 4. Section C, Item 1a Relative to Vacuum-Melting and Degassing Process or the Electroslag Process	

TABLE 1.8N-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		Vacuum Treatment - The requirements for vacuum melting and degassing process or the electroslag process are not essential in meeting the balance of the Regulatory Position nor do they, in themselves, ensure compliance with the overall Regulatory Position. The initial Safety Guide 14 stated that the "flywheel material should be produced by a process that minimized flaws in the material and improves its fracture toughness properties." This is accomplished by using SA-533 material including vacuum treatment. 5. Section C, Item 2b Westinghouse interprets this paragraph as follows: Design Speed Definition Design speed should be 125 percent of normal speed or the speed to which the pump motor might be electrically driven by station turbine generator during anticipated transients, whichever is greater. Normal speed is defined as the synchronous speed of the AC drive motor at 60 Hz. 6. Section C, Item 4b, Inservice Inspections (1) and (2) Instead of the flywheel inspections required at approximately 3-year and 10-year intervals, an inspection will be performed by either a qualified in-place UT examination over the volume from the inner bore of the flywheel to the circle of one-half the outer radius or a surface examination (magnetic particle testing and/or penetrant testing) of exposed surfaces defined by the volume of the disassembled flywheels at least once every 10 years.	
1.20	Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing (Rev. 2, May 1976)	For each prototype reactor internals design, a program of vibration analysis, measurement, and inspection has been developed and reviewed by the NRC. This is documented in WCAP-7879. The reactor internals similar to the prototype design will be subjected during hot functional testing to the same system flow conditions imposed on the prototype design applicable, and for the same duration. Pre- and post test inspections will be conducted to assure that the internals are well behaved and that no excessive motion or wear are experienced.	3.9.2

TABLE 1.8N-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
1.22	Periodic Testing of Protection System Actuation Functions (Rev. 0, February 17, 1972)	Periodic testing of the actuation equipment and actuated equipment of the reactor trip system and the engineered safety features actuation system is in agreement with the provisions of Regulatory Guide 1.22. Where the ability of a system to respond to a bona fide accident signal is intentionally bypassed for the purpose of performing a test during reactor operation, there are provisions so that the bypass condition may be automatically indicated to the reactor operator in the main control room by a separate indication for the train in test. Test circuitry does not allow two trains to be tested at the same time so that extension of the bypass condition to the redundant system is prevented. Actuation logic for the reactor trip system and for the engineered safety features actuation system is tested at power. Where actuated equipment is not tested during reactor operation, it has been determined that: 1. There is no practical system design that would permit operation of the equipment without adversely affecting the safety or operability of the plant 2. The probability that the protection system will fail to initiate the operation of the equipment is, and can be maintained, acceptably low without testing the equipment during reactor operation; and 3. The equipment can routinely be tested when the reactor is shut down. The list of equipment that cannot be tested at full power so as not to damage equipment or upset plant operation is: 1. Manual actuation switches 2. Turbine 3. Main steam line isolation valves (close) 4. Main feedwater isolation valves (close) 5. Feedwater control valves (close) 6. Main feedwater pump trip solenoids 7. Reactor coolant pump seal water return valves (close) 8. Charging header to cold leg isolation valves (close)	7.2 7.3.2.2

TABLE 1.8N-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		9. Charging and letdown isolation valves (close) 10. Spray header isolation valves (open) 11. CVCS suction valves - Normal (close) 12. Instrument air to containment isolation valves (close) 13. Chillwater supply and return containment isolation valves (close)	
1.26	Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants (Rev. 3, February 1976)	The definitions of safety classes found in ANSI N18.2a are utilized by Westinghouse.	3.2.2
1.28	Quality Assurance Program Requirements (Design and Construction) (Rev. 2, February 1979)	The Westinghouse Quality Assurance Plan is presented in WCAP-8370, Rev. 9A, "Westinghouse Quality Assurance Program" and WCAP-7800, "Nuclear Fuel Division Quality Assurance Program Plan."	
1.29	Seismic Design Classification (Rev. 3, September 1978)	Westinghouse classifies each component important to safety as Safety Class 1, 2, or 3 and these classes are qualified to remain functional in the event of the Safe Shutdown Earthquake, except where exempted by meeting all of the below requirements. Portions of systems required to perform the same safety function as required of a safety class component which is part of that system shall be likewise qualified or granted exemption. Conditions to be met for exemption are: 1. Failure would not directly cause a Condition III or IV event (as defined in ANSI N18.2-1973), 2. There is no safety function to mitigate, nor could failure prevent mitigation of, the consequence of a Condition III or IV event, 3. Failure during or following any Condition II event would result in consequences no more severe than allowed for a Condition III event, and 4. Routine post seismic procedures would disclose loss of the safety function.	3.2.1

TABLE 1.8N-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		Westinghouse agrees with Position C2 that establishes a second category of earthquake-resistant equipment but, primarily this affects proper methods of installation and anchoring of certain equipment such that non-Seismic Category I components do not cause loss of function of Seismic Category I components in an earthquake.	
1.31	Control of Ferrite Content in Stainless Steel Welding (Rev. 3, April 1978)	The welding of austenitic stainless steel is controlled to mitigate the occurrence of microfissuring or hot cracking in the weld. Although published data and experience have not confirmed that fissuring is detrimental to the quality of the weld, it is recognized that such fissuring is undesirable in a general sense. Also, it has been well documented in the technical literature that the presence of delta ferrite is one of the mechanisms for reducing the susceptibility of stainless steel welds to hot cracking. However, there are insufficient data to specify a minimum delta ferrite level below which the material will be prone to hot cracking. It is assumed that such a minimum lies somewhere between 0 and 3 percent delta ferrite. The scope of these controls discussed herein encompasses welding processes used to join stainless steel parts in components designed, fabricated, or stamped in accordance with ASME B&PV Code, Section III, Class 1, 2, 3, and CS components. Delta ferrite control is appropriate for the above welding requirements, except in the following cases: where no filler metal is used (for example, in electron beam welding and in autogenous gas shielded tungsten arc welding), where stainless steel filler metal is used for weld metal cladding, explosive welding, and welding using fully austenitic welding materials.	4.5.1 4.5.2.4 5.2.3.4 5.3.1.4 10.3.6.1

TABLE 1.8N-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		The fabrication and installation specifications require welding procedure and welder qualification in accordance with Section III, and include the delta ferrite determinations for the austenitic stainless steel welding materials that are used for welding qualification testing and for production processing. Specifically, the undiluted weld deposits of the "starting" welding materials are required to contain a minimum of 5 percent delta ferrite (or the equivalent Ferrite Number) as determined by chemical analysis and calculation using the appropriate weld metal constitution diagrams in Section III. When new welding procedure qualification tests are evaluated for these applications, including repair welding or raw materials, they are performed in accordance with Section III and Section IX. The results of all the destructive and nondestructive tests are reported in the procedure qualification record in addition to the information required by Section III. The "starting" welding materials used for fabrication and installation welds of austenitic stainless steel materials and components meet the requirements of Section III. The austenitic stainless steel welding material conforms to ASME weld metal analysis A-7 (designated A-8 in the 1974 Edition of the ASME code), type 308 or 308L for all applications. Bare weld filler metal, including consumable inserts, used in inert gas welding processes conform to ASME SFA-5.9, and are procured to contain not less than 5 percent delta ferrite in the deposit according to Section III. Weld filler metal materials used in flux shielded welding processes conform to ASME SFA-5.4 or SFA-5.9 and are procured in a wire-flux combination to be capable of providing not less than 5 percent delta ferrite in the deposit according to Section III. Welding materials are tested using the welding energy inputs to be employed in production welding.	

TABLE 1.8N-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		Combinations of approved heats and lots of "starting" welding materials are used for all welding processes. The welding quality assurance program includes identification and control of welding material by lots and heats, as appropriate. All of the weld processing is monitored according to approved inspection programs which include review of "starting" materials, qualification records, and welding parameters. Welding systems are also subject to quality assurance audits, including calibration of gages and instruments, identification of "starting" and completed materials, welder and procedure qualifications, availability and use of approved welding and heat treating procedures, and documentary evidence of compliance with materials, welding parameters, and inspection requirements. Fabrication and installation welds are inspected using nondestructive examination methods according to Section III rules. To assure the reliability of these controls, Westinghouse has performed a delta ferrite verification program, described in WCAP-8324, "Control of Delta Ferrite in Austenitic Stainless Steel Weldments," June 1974. The verification program has been approved as a valid approach to verify the Westinghouse hypothesis and is considered an acceptable alternative for conformance with the Interim Position on Regulatory Guide 1.31. The Regulatory Staff's acceptance letter and topical report evaluation were received on December 30, 1974. The program results, which support the hypothesis presented in WCAP-8324, are summarized in WCAP-8693, "Delta Ferrite in Production Austenitic Stainless Steel Weldments," January 1976. Welds made in accordance with the criteria discussed herein have continually resulted in sound production welds, which are free from detrimental fissuring and consistently conform to Section III nondestructive acceptance standards.	
1.36	Nonmetallic Thermal Insulation for Austenitic Stainless Steel (Rev. 0, February 1973)	The Westinghouse practice follows the recommendations of Regulatory Guide 1.36 but is more stringent in several respects as discussed below.	5.3.2

TABLE 1.8N-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		The nonmetallic thermal insulation used on the reactor coolant pressure boundary is specified to be made of compounded materials which yield low leachable chloride and/or fluoride concentrations. The compounded materials, in the form of blocks, boards, cloths, tapes, adhesives, cements, etc, are silicated to provide protection of austenitic stainless steels against stress corrosion which may result from accidental wetting of the insulation by spillage, minor leakage, or other contamination from the environmental atmosphere. Each lot of insulation materials is qualified and analyzed to assure that all of the materials provide a compatible combination for the reactor coolant pressure boundary. The tests for qualification specified by the guide (ASTM C692-71 or RDT M12-1T) allow use of the tested insulation material if no more than one of the metallic test samples crack. Westinghouse rejects the tested insulation material if any of the test samples crack. The Westinghouse procedure is more specific than the procedures suggested by the guide, in that the Westinghouse specification requires determination of leachable chloride and fluoride ions from a sample of the insulating material. The procedures in the guide (ASTM D512 and ASTM D1179) do not differentiate between leachable and unleachable halogen ions. In addition, Westinghouse experience indicates that only one of the three methods allowed under ASTM D512 and ASTM D1179 for chloride and fluoride analysis is sufficiently accurate for reactor applications. This is the “referee” method, which is used by Westinghouse.	6.1.1
1.37	Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants (Rev. 0, March 1973)	The Westinghouse position for all Water Reactors Division (WRD) on this Regulatory Guide is documented in WCAP-8370.	4.5.1 17.1.2 17.1.3 17.2

TABLE 1.8N-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
1.38	Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants (Rev. 2, May 1977)	The Westinghouse position for the WRD NSSS scope of supply on Regulatory Guide 1.38 is presented in WCAP-8370, "WRD Quality Assurance Plan." The Nuclear Fuel Division (NFD) position on this Regulatory Guide is presented in WCAP-7800, "NFD Quality Assurance Program Plan."	
1.40	Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants (Rev. 0, March 1973)	There are no Class 1 continuous duty motors installed inside containment in the NSSS scope of supply.	
1.43	Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components (Rev. 0, May 1973)	Westinghouse performs a qualification test on any high heat input welding process (such as the submerged arc wide-strip welding process or the submerged arc 6-wire process) used to clad coarse or fine grained SA-508 Class 2 material. This test follows the recommendations of Regulatory Position C.2 of Regulatory Guide 1.43. Production welding is monitored by the fabricator to ensure that essential variables remain within the limits established by the qualification. If the essential variables exceed the qualification limits, an evaluation will be performed to determined if the cladding is acceptable for use. Where Westinghouse permits the use of the submerged arc strip process on SA-508 Class 2 material, a two-layer technique is used to minimize intergranular cracking.	5.3.1.4
1.44	Control of the Use of Sensitized Stainless Steel (Rev. 0, May 1973)	It has been and continues to be Westinghouse practice to use processing, preoperational cleaning, packaging, and shipping controls to preclude adverse effects of exposure to contaminants on austenitic stainless steel materials. Furthermore, Westinghouse strongly 6.1.1.1 recommends rigorous control of reactor coolant system water chemistry to prevent the intrusion of aggressive species. Austenitic stainless steel materials are utilized in the final heat treated condition required by the respective ASME Section II material specification for the particular type of grade or alloy. More specifically, the austenitic stainless steel materials are utilized in one of the following conditions:	4.5.1 4.5.2.4 5.2.3.4 5.3.1.4 6.1.1.1 10.3.6.2

TABLE 1.8N-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		1. Solution annealed or water quenched, or 2. Solution annealed and cooled by other means through the sensitization temperature range within approximately five minutes. It is generally accepted that these practices will prevent sensitization; Westinghouse has verified this by performing corrosion tests (ASTM 393) on as-received wrought material. The Westinghouse practice is that austenitic stainless steel materials of product forms with simple shapes need not be corrosion tested provided that the solution heat treatment is followed by water quenching. Simple shapes are defined as all plates, sheets, bars, pipe, and tubes, as well as forgings, fittings, and other shaped products which do not have inaccessible cavities or chambers that would preclude rapid cooling when water quenched. Stainless steel cast metal and weld deposits (including weld deposited safe ends), which contain a minimum of 5 percent ferrite, are not considered to be susceptible to sensitization and, therefore, are not corrosion tested. When testing is required, the tests are performed in accordance with ASTM A 262-70, Practice A or E, as amended by Westinghouse Process Specification 84201 MW. This process specification supplements the A262 specification since the latter does not define specimen removal location and does not adequately define bend testing criteria for thick and complex stainless steel raw material. The Westinghouse specification requires that: 1. Specimens be removed from the same location from which mechanical test specimens are removed, and 2. The bend test diameter must be 4X material thickness instead of 1X (Paragraph 36.1, ASTM A 262-70). This second modification is based on the fact that almost all stainless steel materials procured by Westinghouse are eventually welded, and the 4X thickness bend test diameter is required for weldments.	

TABLE 1.8N-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		The heat affected zones of welded components must, of necessity, be heated into the sensitization temperature. However, severe sensitization can be avoided by control of welding parameters and welding processes. Westinghouse controls the heat input in all austenitic pressure boundary weldments by: 1. Prohibiting the use of block welding. 2. Limiting the maximum temperature to 350°F. 3. Exercising approval rights on all welding procedures. Westinghouse has demonstrated the importance of heat input and cooling rate by corrosion testing a number of production and qualification weldments. The tested weldments represented all major welding processes and included a variety of components and base metal thicknesses from 0.10 to 4.0 inches. Portions of only 2 out of 25 weldments exhibited sensitization; in both cases, sensitization was caused primarily by high heat inputs relative to the section thickness. If it becomes necessary to further assure that the present controls are effective in preventing sensitization, Westinghouse will conduct additional corrosion tests on qualification weldments. It is not normal Westinghouse practice to expose wrought unstabilized austenitic stainless steel materials to the sensitization range of 800 to 1,500°F during fabrication other than welding. If, during the course of fabrication, the steel is inadvertently exposed to the sensitization temperature range, the material may be tested (as described in Regulatory Position 5 of the Guide) in accordance with ASTM A 262-70, as amended by Westinghouse Process Specification 84201 MW, or the material will be resolution annealed and water quenched or rejected.	
1.46	Protection Against Pipe Whip Inside Containment (Rev. 0, May 1973)	The criteria implemented in the evaluation of the main reactor coolant loop is based on draft ANS Standard 20.2, “Design Basis for Protection Against Pipe Whip,” and is documented in WCAP-8172-A, “Pipe Breaks for the LOCA Analysis of the Westinghouse Primary Coolant Loop.” WCAP-8172-A has received NRC approval as providing an equivalent degree of protection as would be obtained by applying the criteria of Regulatory Guide 1.46.	3.6

TABLE 1.8N-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
1.48	Design Limits and Loading Combination for Seismic Category 1 Fluid System Components (Rev. 0, May 1973)	Westinghouse meets and will continue to meet the requirements of General Design Criterion 2 and will thereby meet the intent of Regulatory Guide 1.48. The analytical and experimental procedures which Westinghouse will use to demonstrate structural integrity of fluid system components and operability of active components are discussed below. These procedures are intended to provide an alternate acceptable basis for the demonstration of compliance with General Design Criterion 2. Structural Integrity To ensure the structural integrity of fluid systems components, the limits given in RESAR-3, Amendment 5, Section 5.2.1 will be used in the design and analysis of ASME Code Class 1 components. For ASME Code Class 2 and 3 components, the limits given in RESAR-3, Amendment 6, Section 3.9.2 will be used. The conservatism in the above limits and the associated ASME design requirements precludes any component structural failure. For a discussion on operability of active pumps and valves, see Section 3.9.	3.9.3
1.50	Control of Preheat Temperature for Welding of Low-Alloy Steel (Rev. 0, May 1973)	Westinghouse considers that this Guide applies to ASME Section III, Class 1 components. The Westinghouse practice for Class 1 components is in agreement with the requirements of Regulatory Guide 1.50, except for Regulatory Positions 1(b) and 2. For Class 2 and 3 components, Westinghouse does not apply Regulatory Guide 1.50 recommendations. 1. Regulatory Position 1(b) The welding procedures are qualified within the preheat temperature ranges required by Section IX of the ASME Code. Westinghouse experience has shown excellent quality of welds using the ASME qualification procedures. 2. Regulatory Position 2	5.2.3.3

TABLE 1.8N-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		The Westinghouse position is that this guide requirement is both unnecessary and impractical. Code acceptance low-alloy steel welds have been and are being made under present Westinghouse specified procedures. It is not necessary to maintain the preheat temperature until a post-weld heat treatment has been performed by the guide, in the case of large components. In the case of reactor vessel main structural welds, the practice of maintaining preheat until the intermediate or final post-weld heat treatment has been followed by Westinghouse. In either case, the welds have shown high integrity. Westinghouse practices are documented in WCAP-8577, "The Application of Preheat Temperature After Welding of Pressure Vessel Steel," which has been accepted by the NRC.	
1.53	Application of the Single Failure Criterion to Nuclear Power Plant Protection Systems (Rev. 0, June 1973)	Performance of a Single Failure Analysis - The principles described in the IEEE Standard were used in the Power Plant Protection Systems design of the protection systems on Westinghouse plants. For documentation of applicable analysis, refer to WCAP-8584, Revision 1, which is a FMEA for the ESFAS and to WCAP-7706 for Reactor Trip System. The latter topical is in the format of a fault tree analysis rather than a failure mode and effects analysis (FMEA). Even though Regulatory Guide 1.53 proposes a FMEA as an acceptable format, the fault tree analysis format is considered equally acceptable and more useful for arriving at quantitative results. Subjects which are covered in the standard include: 1. Identification of undetectable failures, 2. Analyses of channel interconnections for failures which could compromise independence, 3. Testing to determine independence between redundant parts of the protection system, and 4. Analysis to show that no single failure can cause loss of function due to improper connection of actuators to a power source. The intent of the guide is met in these areas through existing design requirements.	3.1.1

TABLE 1.8N-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		Scope of Analysis - The regulatory guide requires that the single failure analysis extend beyond the scope of IEEE-Standard 279-1971 and include actuation and actuated equipment. WCAP-8584 does not extend to actuation and actuated equipment, but contains interface criteria in Appendix B that, when it is incorporated in the BOP design, assures that the FMEA results are equally applicable to actuation and actuated equipment.	
1.54	Quality Assurance Requirements for Protective Coatings Applied to Water Cooled Nuclear Power Plants (Rev. 0, June 1973)	The Westinghouse NSSS equipment located in the for containment building is separated into four categories to identify the applicability of this Regulatory Guide to various types of equipment. These categories of equipment are as follows: Category 1 - Large equipment Category 2 - Intermediate equipment Category 3 - Small equipment Category 4 - Insulated/stainless steel equipment Category 1 - Large Equipment The Category 1 equipment consists of the following: 1. Reactor Coolant System Supports 2. Reactor Coolant Pumps (motor and motor stand) 3. Accumulator Tanks 4. Manipulator Crane The total exposed surface area for these items is approximately 20,830 (sq ft) for a four loop plant. Since this equipment occupies a large surface area and is procured from only a few vendors, it is possible to implement tight controls over these items. Westinghouse specifies stringent requirements for protective coatings on this equipment through the use of a painting specification in its procurement documents. This specification defines requirements for:	6.1.2 17.1

TABLE 1.8N-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		1. Preparation of vendor procedures 2. Use of specific coatings systems which are qualified to ANSI N101.2 3. Surface preparation 4. Application of the coating systems in accordance with the paint manufacturer's instructions 5. Inspections and nondestructive examinations 6. Exclusion of certain materials 7. Identification of all nonconformances 8. Certifications of compliance The vendor's procedures are subject to review by WRD Engineering personnel, and the vendor's implementation of the specification requirements is monitored during the Westinghouse QA Surveillance activities. This system of controls provides assurance that the protective coatings will properly adhere to the base metal during prolonged exposure to a post-accident environment present within the containment building. No loss of paint is anticipated. <u>Category 2 - Intermediate Equipment</u> The Category 2 equipment consists of the following: 1. Seismic platform and tie rods 2. Reactor internals lifting rig 3. Head lifting rig 4. Electrical cabinets The total exposed surface area of these items is approximately 3,450 sq ft. Since these items are procured from a large number of vendors, and individually occupy very small surface areas, it is not practical to enforce the complete set of stringent requirements which are applied to Category I items. However, Westinghouse does implement another specification in its procurement documents. This specification defines to the vendors the requirements for:	

TABLE 1.8N-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		1. Use of specific coatings systems which are qualified to ANSI N101.2 2. Surface preparation 3. Application of the coating systems in accordance with the paint manufacturer's instructions The vendor's compliance with the requirements is also checked during the Westinghouse QA Surveillance activities in the vendor's plant. Westinghouse believes that these measures of control provide a high degree of assurance that the protective coatings will adhere properly to the base metal and withstand the postulated accident environment within the containment building. However, to be conservative, Westinghouse has not taken credit for this in calculating the amount of paint which might peel or flake off in the post-accident environment. Category 3 - Small Equipment Category 3 equipment consists of the following: 1. Transmitters 2. Alarm horns 3. Small instruments 4. Valves 5. Heat exchanger supports These items are procured from several different vendors and are painted by the vendor in accordance with conventional industry practices. Because the total exposed surface area is only 900 sq ft, Westinghouse does not believe it is necessary to specify further requirements. For purposes of estimating the amount of paint that might peel or flake off, Westinghouse has assumed that all of this material might come off.	

TABLE 1.8N-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		Category 4 - Insulated or Stainless Steel Equipment Category 4 equipment consists of the following: 1. Steam generators - covered with wrapped insulation 2. Pressurizer - covered with wrapped insulation 3. Reactor pressure vessel - covered with rigid reflective insulation 4. Reactor coolant piping - stainless steel 5. Reactor coolant pump casings - stainless steel The wrapped or rigid insulation captures and retains any paint which might come off the equipment surfaces, thereby preventing the paint from blocking the sump drains or interrupting the water flow in the containment spray system.	
1.58	Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel (Rev. 1, September 1980)	The Westinghouse portion for the WRD NSSS scope of supply on Regulatory Guide 1.58 is presented in WCAP-8370, "WRD Quality Assurance Plan." The Nuclear Fuel Division position on this Regulatory Guide is presented in WCAP-7800, "NFD Quality Assurance Program Plan."	17.1.2 17.1.3 17.2
1.60	Design Response Spectra for Seismic Design of Nuclear Power Plants (Rev. 1, December 1973)	The design response spectra of Regulatory Guide 1.60, Revision 1, are acceptable to Westinghouse with the following exception: The damping values recommended and approved by the Staff in WCAP-7921-AR, "Damping Values of Nuclear Power Plant Components," are used in dynamic analysis of Westinghouse supplied equipment.	3.7N
1.61	Damping Values for Seismic Design of Nuclear Power Plants (Rev. 0, October 1973)	The damping values listed in Regulatory Guide 1.61 Design of Nuclear Power are acceptable to Westinghouse for plants using 3D seismic analysis. However, one exception is that of the large piping systems faulted conditions value of 3 percent critical. Higher damping values, when justified by documented test data, have been provided for in Regulatory Position C.2. A conservative value of 4 percent critical has therefore been justified by testing for the Westinghouse reactor coolant loop configuration in WCAP-7921, "Damping Values of Nuclear Power Plant Components," and has been approved by the Staff.	3.7N

TABLE 1.8N-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
1.62	Manual Initiation of Protective Actions (Rev. 0, October 1973)	There are four individual main stop valve momentary control switches (one per loop) mounted on the control board. Each switch, when actuated, will isolate one of the main steam lines. In addition, there will be two system level switches. Each switch will actuate all four main steam line isolation and bypass valves of the system level. Manual initiation of switchover to recirculation is in compliance with Section 4.17 of IEEE Standard 279-1971 with the following comment. Manual initiation of either one of two redundant safety injection actuation main control board mounted switches provides for actuation of the components required for reactor protection and mitigation of adverse consequences of the postulated accident. Manual safety injection actuation will initiate delayed actuation of sequenced started emergency electrical loads if a LOP signal is also present. The safety injection mode is completed when the residual heat removal (RHR) pumps automatically stop on receipt of a low-low RWST level signal. Refer to Section 6.3 for a discussion of the manual switchover from injection mode to cold leg recirculation mode. Manual operation of other components or manual verification of proper position as part of emergency procedures is not precluded nor otherwise in conflict with the above described compliance to paragraph 4.17 of IEEE Standard 279-1971 of the semiautomatic switchover circuits.	7.3.2.2.7

TABLE 1.8N-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		No exception to the requirements of IEEE Standard 279-1971 has been taken in the manual initiation circuit of safety injection. Although paragraph 4.17 of IEEE Standard 279-1971 requires that a single failure within common portions of the protective system shall not defeat the protective action by manual or automatic means, the standard does not specifically preclude the sharing of initiated circuitry logic between automatic and manual functions. It is true that the manual safety injection initiation functions associated with one actuation train (e.g., train A) shares portions of the automatic initiation circuitry logic of the same logic train; however, a single failure in shared functions does not defeat the protective action of the redundant actuation train (e.g., train B). A single failure in shared functions does not defeat the protective action of the safety function. It is further noted that the sharing of the logic by manual and automatic initiation is consistent with the system level action requirements of the IEEE Standard 279- 1971, paragraph 4.17, and consistent with the minimization of complexity. Although manual actuation of main steamline isolation (all valves), containment isolation (Phase A), and containment spray actuation is not within the NSSS scope, the same criteria herein described for the manual safety injection also applies to these aforementioned manual actuation functions in the balance of plant scope.	
1.64	Quality Assurance Requirements for the Design of Nuclear Power Plants (Rev. 2, June 1976)	The Westinghouse position for the WRD NSSS scope of supply on Regulatory Guide 1.64 is presented in WCAP-17.1.3 8370 “WRD Quality Assurance Plan.” The Nuclear Fuel Division position on this Regulatory Guide is presented in WCAP-7800, “NFD Quality Assurance Program Plan.”	17.1.2 17.1.3 17.2
1.65	Materials and Inspections for Reactor Vessel Closure Studs (Rev. 0, October 1973)	Westinghouse is in agreement with Regulatory Guide 1.65 with the following exception for the material and tensile strength guidelines:	5.3.1

TABLE 1.8N-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		1. Westinghouse has specified both 45 ft lb and 25 mils lateral expansion for control of fracture toughness determined by Charpy-V testing, required by the ASME Boiler and Pressure Vessel Code, Section III, Summer 1973 Addenda and 10 CFR Part 50, Appendix G (July 17, 1973, Paragraph IV.A.4). These toughness requirements assure optimization of the stud bolt material tempering operation with the accompanying reduction of the tensile strength level when compared with previous ASME Boiler and Pressure Vessel Code requirements. The specification of both impact and maximum tensile strength as stated in the guide results in unnecessary hardship in procurement of material without any additional improvement in quality. The closure stud bolting material is procured to a minimum yield strength of 130,000 psi and a minimum tensile strength of 145,000 psi. This strength level is compatible with the fracture toughness requirements of 10 CFR 50, Appendix G (July 1973, Paragraph 1.C), although higher strength level bolting materials are permitted by the code. Stress corrosion has not been observed in reactor vessel closure stud bolting manufactured from material of this strength level. Accelerated stress corrosion test data do exist for materials of 170,000 psi minimum yield strength exposed to marine water environments stressed to 75 percent of the yield strength (given in Reference 2 of the Guide). These data are not considered applicable to Westinghouse reactor vessel closure stud bolting because of the specified yield strength differences and a less severe environment; this has been demonstrated by years of satisfactory service experience.	

TABLE 1.8N-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		The ASME Boiler and Pressure Vessel Code requirement for toughness for reactor vessel bolting has precluded the guide's additional recommendation for tensile strength limitation, since to obtain the required toughness levels, the tensile strength levels are reduced. Prior to 1972, the Code required a 35 ft lb toughness level which provided maximum tensile strength levels ranging from approximately 155 to 178 kpsi (Westinghouse review of limited data - 25 heats). After publication of the Summer 1973 Addenda to the Code and 10 CFR Part 50, Appendix G, wherein the toughness requirements were modified to 45 ft lb with 25 mils lateral expansion, all bolt material data reviewed on Westinghouse plants showed tensile strengths of less than 170 kpsi. Additional protection against the possibility of incurring corrosion effects is assured by: 1. Decrease in level of tensile strength comparable with the requirements of fracture toughness as described above. 2. Design of the reactor vessel studs, nuts, and washers allowing them to be completely removed during each refueling permitting visual and/or nondestructive inspection in parallel with refueling operations to assess protection against corrosion, as part of the inservice inspection program described in Chapter 5. 3. Design of the reactor vessel studs, nuts, and washers, providing protection against corrosion by allowing them to be completely removed during each refueling. The bolting materials are discussed in Chapter 5. 4. Use of manganese phosphate or a vapor phase plating process. a. Use of Code Case 1605 does not constitute an issue between the NRC and Westinghouse inasmuch as use of this code case has been approved by the NRC via the guideline of Regulatory Guide 1.85 (see Revision 6, May 1976).	
1.67	Installation of Overpressure Protection Devices (Rev. 0, October 1973)	The scope of Regulatory Guide 1.67 is limited to the design of open discharge systems. Installation of overpressure protection devices is in the balance of plant (BOP) and therefore this guide is not in the Westinghouse NSSS scope of supply.	5.4.11.3

TABLE 1.8N-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
1.71	Welder Qualification for Areas of Limited Accessibility (Rev. 0, December 1973)	Westinghouse practice does not require qualification or requalification of welders for areas of limited accessibility as described by the Guide and has provided welds of high quality. Westinghouse believes that limited accessibility qualification or requalification, which are additional to ASME Section III and IX requirements, is an unduly restrictive requirement for shop fabrication, where the welders' physical position relative to the welds is controlled and does not present any significant problems. In addition, shop welds of limited accessibility are repetitive due to multiple production of similar components, and such welding closely supervised. For field application, the type of qualification should be considered on a case-by-case basis due to the great variety of circumstances encountered.	5.2.3 5.3.1.4 10.3.6.2
1.73	Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants (Rev. 0, January 1974)	The qualification programs for Westinghouse WRD supplied Class IE electric motor operators, solenoid valves, and limit switches described in WCAP-8587 and WCAP-9688 meet the requirements of Regulatory Guide 1.73.	3.11N 8.3.1
1.74	Quality Assurance Terms and Definitions (Rev. 0, February 1974)	The Westinghouse position for the WRD NSSS scope of supply on Regulatory Guide 1.74 is presented in WCAP-8370, “WRD Quality Assurance Plant.” The Nuclear Fuel Division position on this Regulatory Guide is presented in WCAP-7800, “NFD Quality Assurance Program Plan.”	17.1.2 17.2
1.75	Physical Independence of Electric Systems (Rev. 2, September 1978)	Westinghouse takes exception to the Regulatory Guide 1.75 in several areas as discussed below. These issues have been presented to the Regulatory Staff and are not resolved at this time. 1. <u>Isolation Devices (Paragraph 3.8)</u> Regulatory Position: Interrupting devices actuated by fault current are not isolation devices. Westinghouse Position: Interrupting devices actuated by fault current are isolation devices when justified by test or analysis.	7.1 8.3.1.4

TABLE 1.8N-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		2. Cable Spreading Area and Main Control Room (Paragraph 5.1.3) Regulatory Position: Places additional severe restrictions on equipment in area. Westinghouse Position: The IEEE draft criteria are adequate. 3. Instrument Cabinets (Paragraph 5.7) Regulatory Position: Separation requirements for instrument cabinets are the same as those for control boards. Westinghouse Position: Separation requirements should not be the same for instrumentation racks and control boards because functional requirements are different. The IEEE draft criteria are adequate. Refer to WCAP-8892-A and FSAR Section 7.1.2.2.1 for further information.	
1.77	Assumption Used for Evaluating a Control Rod Ejection Accident for Pressurized Water Reactors (Rev. 0, May 1974)	The result of the Westinghouse analysis shows compliance with the Regulatory Position given in Section C.1 of Regulatory Guide 1.77. In addition, Westinghouse complies with the intent of the assumptions given in Appendix A of the Regulatory Guide. However, Westinghouse takes exception to Position C.2, which implies that the Rod Ejection Accident should be considered as an emergency condition. Westinghouse considers this a faulted condition as stated in ANSI N18.2. Faulted condition stress limits will be applied for this accident. Westinghouse also complies with Position C.3 for dose calculations and uses the assumptions in Appendix B.	15.4.2 15.4.7
1.82	Sumps for Emergency Core Cooling and Containment Spray Systems (Rev. 0, June 1974)	The Robust Fuel Assembly (RFA) implemented in Cycle 7 (Region 9) includes the debris resistant bottom nozzle (DRBN) and the protective bottom grid (P-Grid) fuel features (see Section 4.2). Due to these Region 9 fuel features, the minimum restriction at the fuel assembly inlet of approximately 0.075 in. is larger than the fine mesh screening for the sump (1/16 in. = 0.0625 in).	6.2.2.2

TABLE 1.8N-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
1.83	Inservice Inspection of Pressurized Water Reactor Steam Generator Tubes (Rev. 1, July 1975)	Access The Westinghouse steam generator design permits access to steam generator tubes for inspection, plugging, or other repair. Baseline Inspection Westinghouse concurs with the option of the last paragraph of Section B, which permits the shop examination of tubing to serve as an adequate baseline inspection, provided that the examination is done in accordance with the requirements of the ASME Code, Section III, Subsection NB, Article 2550. The owner may, at his option, perform the inspection prior to operation of the plant in accordance with paragraph C.3.a. If the shop examination is chosen to serve as a baseline inspection, the technical details of the procedure should be presented to assure that the shop examination is no less sensitive than the succeeding inservice inspection technique. Sample Selection, Supplementary Sampling, Testing, and Acceptance Limits The detailed requirements for inservice inspections are delineated in the Applicant's Technical Specifications.	5.4.2.2 16.3 / 4.4.5
1.84	Design and Fabrication Code Case Acceptability - ASME Section III, Division 1 (Rev. 15, May 1979)	Westinghouse believes that code cases should be treated as any other part of the code and therefore accepted as approved by the ASME council. For Class 1 components, Footnote 6 to 10 CFR 50.55a applies to Millstone 3. Code Case 1528 was used in the procurement of components. Westinghouse has applied for generic approval for the use of this code case via letter NS-CE-1228 of October 4, 1976. Uses of any other code case applicable to Code Class 1 components were identified in applicable safety reports for U.S.A. projects and concurrence of the Commission was obtained by its approval of Applicant's documents and issuance of Construction Permits.	5.2.1.2

TABLE 1.8N-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		For Class 2, 3, and CS components, only code cases approved by the ASME council have been used.	
1.85	Materials Code Case Acceptability - ASME Section III, Division 1 (Rev. 15, May 1979)	See compliance for Regulatory Guide 1.84 above.	5.2.1.2
1.88	Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records (Rev. 2, October 1976)	The Westinghouse position for the WRD NSSS scope of supply on Regulatory Guide 1.88 is presented in WCAP-8370, "WRD Quality Assurance Plan." The Nuclear Fuel Division position on this Regulatory Guide is presented in WCAP-7800, "NFD Quality Assurance Program Plan."	17.1.2 17.1.3 17.2
1.89	Qualification of Class IE Equipment of Nuclear Power Plants (Rev. 0, November 1974)	For Westinghouse NSSS Class IE equipment, Westinghouse meets IEEE Standard 323-1974 (including IEEE 323a-1975 position statement of July 24, 1975) and Regulatory Guide 1.89 by an appropriate combination of any or all of the following: Type testing, operating experience, qualification, by analysis, and ongoing qualification. This commitment was satisfied by implementation of WCAP-8587, "Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety Related Electrical Equipment," as discussed in Section 3.11N.	3.11N
1.92	Combination of Modes and Spatial Components in Seismic Response Analysis (Rev. 0, December 1974)	Westinghouse will use the alternate method described in Section 3.7 to combine modal responses in the evaluation of seismic responses for Millstone 3.	3.7.2
1.97	Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident (Rev. 2, December 1980)	Westinghouse does not believe that adoption of the full scope of instrumentation proposed by Regulatory Guide 1.97, Revision 2, is necessary for safe operation of a Westinghouse pressurized water reactor. Nevertheless, Westinghouse has established a design basis with respect to Regulatory Guide 1.97, Revision 2, in order to support its customers in responding to this licensing recommendation. The position stated below indicates where the Westinghouse design basis takes exception to the regulatory positions contained in Section C and the implementation criteria contained in Section D of the guide:	7.4 7.5

TABLE 1.8N-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		1. Regulatory Guide Position C.1 a. The selection criteria specified by the NRC in Table 2 imply that all Type A variables should be Category I. Westinghouse permits Category 2 and 3 under Type A where a variable is employed in the Westinghouse Reference Emergency Operating Instructions for the sole purpose of providing preferred backup information. b. The guide contains (Sections 1.3.1b, 1.3.2.e, f, etc) recommendations concerning the methods of displaying information. Westinghouse has retained the intent of the Regulatory Guide display requirements (i.e., information should be immediately available, versus continuously displayed, etc) in the design basis, but has permitted the method of display to be optimally selected elsewhere in conjunction with the implementation of requirements contained in NUREG-0700, "Human Engineering Design Guidelines." c. The guide designates Type D and E key variables as Category 2. In addition, Westinghouse has designated as Category 2 those Type A, B, and C variables which provide preferred backup information and whose instrumentation will be subject to a high energy line break environment, when required, to provide the backup information. d. The qualification conditions specified for Category 2 contain the definitions for two subcategories by indicating that seismic qualification may be required for only those instruments associated with non safety systems need not be seismically qualified. In both cases, however, environmental qualification to Regulatory Guide 1.89 is specified. The logic for requiring environmental qualification, but not seismic, in certain cases is not explained and furthermore: - Regulatory Guide 1.89 itself requires seismic qualification as an essential part of the qualification sequence.	

TABLE 1.8N-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		- The basis for identifying what instrumentation is part of a safety-related system is open to interpretation. For example, this could mean that only instrumentation that is an essential part of the safety system performance is included (i.e., actuation of safeguards). Alternatively, this definition could be interpreted to include instrumentation that merely monitors the performance of the safety-related system (i.e., status lights). As a consequence, Westinghouse does not support this subdivision of Category 2 and recommends that Category 2 instrumentation be qualified for at least the environment (seismic and/or environmental) in which it must operate to service its intended post-accident function. e. The NRC has indicated (Section 1.3.3.a) that Category 3 instrumentation should be of high quality commercial grade and should be selected to withstand the specified service environment. In addition, it is Westinghouse's position that Category 3 instrumentation should not be required to provide information to the operator when exposed to a hostile environment resulting from a high energy line break. f. Westinghouse does not believe it appropriate to specify periodic checking, testing, calibration, and calibration verification in accordance with Regulatory Guide 1.118 for commercial grade instrumentation and has deleted this requirement. The scope of Regulatory Guide 1.118 is restricted to periodic testing of the protection system and electric power systems for systems important to safety. As such, it should not be applied in relation to non safety equipment. 2. Regulatory Position C.2	

TABLE 1.8N-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		The NRC definition of Type D employs the terminology; plant safety systems and other systems important to safety. Westinghouse does not employ this terminology since it has no strict definition. Westinghouse has defined the scope of Type D to include safety systems employed for mitigating the consequences of an accident and for achieving subsequent plant recovery to a safe shutdown condition and other systems normally employed for attaining safe shutdown. 3. Implementation - Section D The implementation section of the guide recognizes, in some part, the need to modify some of the requirements when considering backfit to operating plants or plants under construction. However, many of the regulatory guides referenced in the Regulatory Position (Section C) are not superseded by any statement under Section D. Westinghouse believes Regulatory Guide 1.97, Revision 2, should be revised to rectify this omission. It is the Westinghouse position that the criteria to be specified for Category I post-accident monitoring instrumentation for operating or licensed plants should be at least equivalent to those criteria originally specified for the plant's Class IE systems.	
1.99	Effects of Residual Elements on Predicted Radiation Damage to Reactor Vessel Materials (Rev. 1, April 1977)	Justification of the Westinghouse position on Revision 0 and Revision 1 of the Guide is detailed in References 1 and 2, respectively. In summary, Revision 1 of the guide is substantially identical with Revision 0, with minor clarifications and inclusion of a new position C.2, which had previously been included in the Discussion section of Revision 0. The Westinghouse letter of comment on Revision 1 reiterates the comments of Revision 0 and includes further clarification of vessel material hardship imposed by the guide. The Westinghouse position, with respect to each of the guide positions, is as follows:	5.3

TABLE 1.8N-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		1. Regulatory Position C.1 The basis, as well as the scope, of the guide for predicting adjustment of reference temperature as given in Regulatory Position C.1 are inappropriate since the data base used was incomplete and included some data which were not applicable. 2. Regulatory Position C.2 Westinghouse is in agreement with the Guide Position C.2a. However, with respect to Guide Position C.2b, Westinghouse believes that Figure 2 of the Guide is incorrect since the upper shelf energy for the 6-inch thick ASTM A302B reference correlation monitor material reported by Hawthorne indicates essentially a constant upper shelf at fluences above 1 x 10 n/cm (Hawthorne). 3. Regulatory Position C.3 The Westinghouse position, with reference to the Guide Position C.3 controlling residual elements to levels that result in a predicted adjusted reference temperature of less than 200°F at end-of-life, is that the stresses in the vessel can be limited during operation in order to comply with the requirements of Appendix G to 10 CFR Part 50, even though the end-of-life adjusted reference temperature may exceed 200°F. By applying the procedures of Appendix G to ASME Section III, the stress limits, including appropriate Code safety margin, can be met. References Letter of Comment on Revision 0 of the Guide to the Secretary of the Commission by C.E. Eicheldinger, NS-CE-784, September 22, 1975. Hawthorne, J.R., "Radiation Effects Information Generated on the ASTM Reference Correlation - Monitor Steels," to be published.	
1.100	Seismic Qualification of Electric Equipment for Nuclear Power Plants (Rev. 1, August 1977)	Westinghouse qualifies equipment to the requirements of IEEE 344-1971. Westinghouse has performed extensive testing to demonstrate the adequacy of seismic qualification to the requirements of IEEE 344-1971. See Section 3.10 for a further discussion of the seismic qualification of electrical equipment. Replacement items meet the original criteria or either IEEE 344-75 or IEEE-344-87 (endorsed by Reg. Guide 1.100, Rev. 2).	

TABLE 1.8N-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
1.105	Instrument Setpoints (Rev. 1, November 1976)	Technical Specifications provide the margin from the nominal setpoint to the technical specification limit. The allowances between the technical specification limit and the safety limit include the following items: a. The inaccuracy of the instrument b. Process measurement accuracy c. Uncertainties in the calibration d. The potential transient overshoot determined in the accident analyses (this may include compensation for the dynamic effect), and e. Environmental effects on equipment accuracy caused by postulated or limiting postulated events (only those systems required to mitigate consequences of an accident). Westinghouse designers choose setpoints such that the accuracy of the instrument is adequate to meet the assumptions of the safety analysis. The range of instruments is chosen based on the span necessary for the instrument's function. Narrow range instruments will be used where necessary. Instruments will be selected based on expected environmental and accident conditions. The need for qualification testing will be evaluated and justified on a case basis. Administrative procedures coupled with the present cabinet alarms and/or locks provide sufficient control over the setpoint adjustment mechanism such that no integral setpoint securing device is required. Integral setpoint locking devices will not be supplied. The assumptions used in selecting the setpoint values in Regulatory Position C.1 and the minimum margin, with respect to the technical specification limit and calibration uncertainty, will be documented by Westinghouse. Drift rates and their relationship to testing intervals will not be documented by Westinghouse.	7.1.2.1.9 7.2.2.2.1 16.3/ 16.4.3

TABLE 1.8N-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
1.116	Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems (Rev. O-R, May 1977)	The subject of the Regulatory Guide is not in the Westinghouse NSSS scope of supply. The Westinghouse position for the WRD NSSS scope of supply on Regulatory Guide 1.116 is presented in WCAP-8370, "WRD Quality Assurance Plan." This Regulatory Guide is not applicable within NFD.	17.1.2 17.2
1.118	Periodic Testing of Electric Power and Protection Systems (Rev. 2, June 1978)	Westinghouse will make clear the distinction between "recommendations" and "requirements" when addressing criteria. The position is as follows: Westinghouse defines "Protective Action Systems" to mean electric, instrumentation, and controls portions of those protection systems and equipment actuated and controlled by the protection system. Equipment performing control functions, but actuated from protection system sensors is not part of the safety system and will not be tested for time response. Status, annunciating, display, and monitoring functions, except those related to the Post Accident Monitoring Systems (PAMS) are considered by Westinghouse to be control functions. Reasonability checks, i.e., comparison between or among similar such display functions, will be made. Response time testing for control functions operated from protection system sensors will not be performed. Moreover, Nuclear Instrumentation sensors are exempt from testing since "their worst case response time is not a significant fraction of the total overall system response (i.e., less than 5 percent)." This exemption is permitted by IEEE-338. The standard Westinghouse protection system design does not include provisions which permit in-situ testing of process sensors.	7.1.2.4 8.1.6
1.121	Bases for Plugging Degraded PWR Steam Generator Tubes (Rev. 0, August 1976)	Millstone 3 complies with Regulatory Guide 1.21, Rev. 0, August 1976 with the following clarifications: 1. Regulatory Position C.1 Westinghouse interprets the term "Unacceptable defects" to apply to those imperfections resulting from service induced mechanical or chemical degradation of the tube walls which have penetrated to a depth in excess of the Plugging Limit.	5.4.2

TABLE 1.8N-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		2. Regulatory Position C.2a(2) and C.2a(4) Westinghouse will use a 200-percent margin of safety based on the following definition of tube failure. Westinghouse defines tube failure as plastic deformation of a crack to the extent that the sides of the crack open to a nonparallel, elliptical configuration. This 200-percent margin of safety compares favorable with the 300-percent margin requested by the NRC against gross failure. 3. Regulatory Position C.2.b a. In cases where sufficient data exist to establish degradation allowance, the rate used will be an average time-rate determined from the mean of the test data. b. Where requirements for minimum wall are markedly different for different areas of the tube bundle, e.g., U-bend area versus straight length in Westinghouse designs, two plugging limits may be established to address the varying requirements in a manner which will not require unnecessary plugging of tubes. 4. Regulatory Position C.3.d(1) and C.3.d(3) The combined effect of these requirements would be to establish a maximum permissible primary-to-secondary leak rate which may be below the threshold of detection with current methods of measurement. Westinghouse has determined the maximum acceptable length of a through-wall-crack based on secondary pipe break accident loadings which are typically twice the magnitude of normal operating pressure loads. Westinghouse will use a leak rate associated with the crack and size determined on the basis of accident loadings. 5. Regulatory Position C.3.e(6) Westinghouse will supply computer code names and references rather than the actual codes.	

TABLE 1.8N-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		6. Regulatory Position C.3.f(1) Westinghouse will establish a minimum acceptable tube wall thickness (Plugging Limit) based on structural requirements and consideration of loadings, measurement accuracy, and, where applicable, a degradation allowance as discussed in this position and in accordance with the general intent of this guide. Analyses to determine the maximum acceptable number of tube failures during a postulated condition are normally done to entirely different bases and criteria are not within the scope of this guide.	
1.123	Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants (Rev. 1, July 1977)	The Westinghouse position for the WRD NSSS scope of supply in Regulatory Guide 1.123 is presented in WCAP-8370, "WRD Quality Assurance Plan." The Nuclear Fuel Division position on this Regulatory Guide is presented in WCAP-7800, "NFD Quality Assurance Program Plan."	17.1.3 17.2
1.126	An Acceptable Model and Related Statistical Methods for the Analysis of Fuel Densification (Rev. 1, March 1978)	Millstone 3 does not use the model and related statistical method for the analysis of fuel densification that is presented in Regulatory Guide 1.126. The Regulatory Guide clearly states that, "The model presented in...this Guide is not intended to supersede NRC approved vendor models." Millstone 3 uses the Westinghouse fuel densification model presented in WCAP-8218 (Proprietary) which has been approved by the NRC. WCAP-8219 (Nonproprietary) and WCAP-8264 (Customer Version) are companions to the approved versions.	4.2
1.139	Guidance for Residual Heat Removal (Rev. 0, May 1978)	Millstone 3 meets the requirements of BTP RSB 5-1 and SRP 5.4.7 except as indicated in FSAR Sections 5.4.7.1 and 5.4.7.2.4.	
1.141	Containment Isolation Provisions for Fluid Systems (Rev. 0, April 1978)	Westinghouse's containment isolation philosophy for fluid systems complies with the guidance provided by ANSI N271-1976 and/or Regulatory Guide 1.141 with the following exceptions and/or clarifications:	5.4.5 5.4.6 6.2.4

TABLE 1.8N-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		1. The standard in Section 3.6.3 states that remote manual closure of isolation valves on ESF or ESF related systems is acceptable when provisions are made to detect possible failure of the fluid lines inside and outside containment. Although such provisions are outside Westinghouse scope of supply, Westinghouse is of the opinion that provisions to detect failure of fluid lines inside containment are unnecessary. Since redundant ESF capacity is provided and off site doses due to leakage inside containment are not a concern, Westinghouse does not require or provide for detection of failures in fluid lines inside containment. 2. Section 3.6.4 states that a single valve and closed system outside containment is acceptable if the closed system is treated as an extension of the containment. Further, the standard requires that the valve and the piping between the valve and the containment be enclosed in a protective leak tight or controlled leakage compartment. The closed system is also required to be leak tested in accordance with 10 CFR 50 Appendix J unless it can be shown by inspection that system integrity is being maintained for those systems operating during normal plant operation at a pressure equal to or above the containment design pressure. Sections 4.2.5 and 4.4.6 of the standard as implemented by item C.3 of the Regulatory Guide are interpreted by Westinghouse to state that to preclude common mode failures, diversity is required in the parameters sensed from which isolation signals are generated. Westinghouse design criteria for the initiation of containment isolation do not include a requirement for diversity in the primary system variables for any given design condition or event.	

TABLE 1.8N-1 NRC REGULATORY GUIDES (CONTINUED)

R.G. No.	Title	Degree of Compliance	FSAR Section Reference
		Westinghouse, however, utilizes different primary system variables to derive and generate the protection function for the first phase (A) of containment isolation. Some diversity is therefore available for this phase for a given event. The second phase of containment isolation (B), which isolates only component cooling water to the reactor coolant pump, is initiated only by a high containment pressure signal. Diversity is therefore not available for this containment isolation function. Westinghouse believes that restricting the standards recommendation (i.e., for diverse means of actuation) to diversity in parameters sensed that the Regulatory Guide unjustifiably ignores the benefits of alternate methods of obtaining different means of actuation. Westinghouse believes the standard gives appropriate guidance and the Regulatory Guide restriction should be deleted. The standard states in Section 1 that "If an accident occurred, fluid systems penetrating the containment would be isolated except those which are engineered safety related." With respect to this recommendation, the reactor coolant pump seal injection line is either isolated or not isolated, depending on whether the charging pump is used for safety injection and the value of the pump header discharge pressure. On plants which utilize the charging pumps for safety injection, flow will be provided by the pumps through the seal injection lines following an accident. Each line is, however, equipped with a remote manual containment isolation valve which the operator can close when the charging pumps have completed their safeguard function.	
1.144	Auditing of Quality Assurance Program for Nuclear Power Plants (Rev. 0, January 1979)	The Westinghouse position for the WRD NSSS scope of supply on Regulatory Guide 1.144 is presented in WCAP-8370, "WRD Quality Assurance Plan." The Nuclear Fuel Division position on this Regulatory Guide is presented in WCAP-7800, "NFD Quality Assurance Program Plan."	17.1.2 17.1.3 17.2
1.146	Qualification of Quality Assurance Program Audit Personnel for Nuclear Power (Rev. 0, August 1980)	The Westinghouse position for the WRD NSSS scope of supply on Regulatory Guide 1.146 is presented in WCAP-8370, "WRD Quality Assurance Plan." The Regulatory Guide is not applicable within NFD.	

1.9 STANDARD REVIEW PLAN DOCUMENTATION OF DIFFERENCES

The Millstone 3 FSAR was reviewed against NUREG-0800 at the time of application for an operating license to satisfy the requirements of 10 CFR 50.34(g). The following two tables were developed to identify deviations from SRP acceptance criteria and provided a justification for those deviations. The differences noted were not construed as variances from regulation, rather, they documented the deviations from acceptance criteria as stated within the NRC's internal review guide (SRP) for safety analysis reports.

Table 1.9-1 summarized the differences between the Millstone 3 FSAR and NUREG-0800.

Table 1.9-2 presented the FSAR differences from NUREG-0800 and their justifications.

Information contained in this section (Tables 1.9-1 and 1.9-2) has been retained for historical purposes.

TABLE 1.9-1 SUMMARY OF DIFFERENCES FROM SRP

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Corresponding FSAR Section
2.1.3 (Rev. 2)	II.6 - Population density within 30-mi radius.	A distance of 60 km (37 mi) used for checking density.	2.1.3.6
2.5.1 (Rev. 2)	II. 4.b,c, and d - Zones of alterations, irregular weathering, structural weakness; unrelieved residual stresses in rock; unstable material or areas considered unstable.	These features are not discussed.	2.5.1.2
2.5.2 (Rev. 1)	II (2.5.2.1) - Magnitude designations.	Earthquake magnitudes not identified as M_b, M_L, or M_s.	Tables 2.5.2–3 and 2.5.2–4
	II (2.5.2.4) - Earthquake return period.	There is no probabilistic determination of earthquake return period for the largest earthquakes in each province.	2.5.2.4
	II (2.5.2.7) - Probability of exceeding acceleration of OBE.	No estimate given for probability of exceeding acceleration of OBE.	2.5.2.7
2.5.3 (Rev. 2)	II (2.5.3.1) - Offshore geologic investigation.	No detailed offshore geologic investigation of 5-mi radius has been done.	2.5.3.1
	II (2.5.3.4) - Fault location.	Location and investigation of all faults within 5 mi has not been done.	2.5.3.4
	II (2.5.3.6) - Age of faults.	Age documentation of all faults within 5 mi has not been done.	2.5.3.6
2.5.4 (Rev. 2)	II (2.5.4.2) - Analyses for saturated soils and clays underlying the site.	No FSAR table lists values of parameters used in analyses of liquefaction potential, behavior, or static and dynamic behavior.	2.5.4.2
3.2.2 (Rev. 1)	II - Use of RG 1.26 as the acceptance criteria for defining Quality Groups for components important to safety.	Millstone 3 uses the classification system provided in ANS 18.2.	3.2.2
3.4.1 (Rev. 2)	III.3 - Postulated failures of nonseismic Category I and non-tornado protected tanks.	FSAR does not directly address these postulated failures.	3.4.1, 3.6
3.4.2 (Rev. 2)	II. 1 - Use of DBF or highest groundwater level in design.	DBF or normal groundwater level used.	3.4.2

TABLE 1.9-1 SUMMARY OF DIFFERENCES FROM SRP (CONTINUED)

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Corresponding FSAR Section
3.5.1.3 (Rev. 1)	II - Missile protection for cold shutdown maintenance components.	Cold shutdown maintenance components not specified as targets.	3.5.1.3
3.5.1.5 (Rev.1)	III.3 - Definition of P_P.	Missiles that produce secondary missiles which could damage vital equipment not considered.	3.5.1.5, 2.2.3.2.1
3.5.1.6 (Rev. 2)	III.2 - Inflight crash rate of $4x10^{-10}$.	NUREG-75/087 crash rate of $3x10^{-9}$ used.	3.5.1.6
3.6.1 (Rev. 1)	BTP ASB 3-1, B.1.a(1) -Requires an arbitrary split be postulated on the main steam and the feedwater systems at a location proximate to essential systems.	FSAR does not commit to postulate this arbitrary split.	3.6.1.3.3
	BTP ASB 3-1, B.1.a(2) -Suggests main steam and feedwater pipes not be routed in the vicinity of the control room.	Main steam and feedwater pipes are routed in the vicinity of the control room.	3.6.1.3.3
	BTP ASB 3-1, B.2.a -States that essential systems and components should be designed to meet the seismic design criteria of R.G. 1.29.	In the aux steam and hot water heating systems, only the electrical detection and actuation devices for the isolation valves are qualified to Class 1E requirements and located in a seismic Cat. I Building.	3.6.1.3.1
3.6.2 (Rev. 1)	III.2.a - Requires use pressure and temperature values corresponding to the greater contained energy at hot standby or at 102 per cent power.	FSAR uses internal pressure, and temperature conditions in the piping system during reactor operation at 100 percent power.	3.6.2.2.1
	III.2.a - Requires that the allowable capacity for crushable material shall be limited to 80 percent of its rated energy absorbing capacity as determined by dynamic testing.	FSAR uses an allowable of 80 percent of energy absorbing capacity based on static testing.	3.6.2.2.1
	BTP MEB 3-1, B.1.e -States particular criteria for postulating through wall leak age cracks in high energy Class 1, 2, 3, and non-nuclear piping.	FSAR does not postulate cracks in high energy piping.	3.6.2.1.2
3.7.2 (Rev. 1)	II.4 - Finite element and half space representation of subgrade soil stiffness.	FSAR does not address half space representation.	3.7.2

TABLE 1.9-1 SUMMARY OF DIFFERENCES FROM SRP (CONTINUED)

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Corresponding FSAR Section
	II.11 - Consideration of accidental torsion by assuming additional seismicity of ±5 per cent of maximum building dimension.	Additional seismicity not addressed.	3.7B.2.11
3.7.3 (Rev. 1) (BOP Scope)	II.2.l (1) - Interaction of Category I piping extended to address on a "system basis."	FSAR discussion is restricted to piping.	3.7B.3.13
3.7.3 (Rev. 1) (NSSS Scope)	II.2.g - Combination of closely spaced modes should be in accordance with Regulatory Guide 1.92.	Westinghouse combines closely spaced modes as described in FSAR Section 3.7N.3.7.	3.7N.3.7
3.8.1 (Rev. 1)	II.2 - Use of Regulatory Guide 1.136.	FSAR does not reference Regulatory Guide 1.136.	3.8.1
	11.4.f - Use of ASME III, Division 2, Article CC-3000 for design of containment structure tangential shear.	Article CC-3000 of ASME III, Division 2 was not used.	3.8.1.4.1
	II.4.j - Ultimate capacity of reactor containment.	Ultimate capacity of the reactor containment is not discussed.	3.8.1
	II.5 - Article 3000 of ASME III, Division 2 for loads, load combinations, and stress allowables.	Article 3000 of ASME III, Division 2 not used.	3.8.1.2
3.8.3 (Rev. 1)	II.2 - ACI 349-76.	ACI 349-76 was not used.	3.8.3.2, 3.8.1.2
3.8.4 (Rev. 1)	II.2 - ACI 349-76.	ACI 349-76 was not used.	3.8.4.2, 3.8.1.2
	II.4.d - Design report format.	FSAR does not use this format.	3.8.4
3.8.5 (Rev. 1)	II.4.b - ACI 349-76. rather than ACI349-76.	ACI 318-71 was used	3.8.5.2, 3.8.1.2
3.9.1 (Rev. 2) (BOP Scope)	III.1 - Plant conditions identified as design Levels A,B,C,D.	FSAR identifies plant conditions as normal, upset, emergency, and faulted.	3.9B.1.1
	III.4 - Methods used in stress analysis of components.	FSAR contains no justification for methods used	3.9B.1.4
3.9.1 (Rev. 2) (NSSS Scope)	II.2 - Computer codes used in design and analysis of seismic Category I components.	Only a brief description of computer codes used by Westinghouse is given.	3.9N.1.2

TABLE 1.9-1 SUMMARY OF DIFFERENCES FROM SRP (CONTINUED)

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Corresponding FSAR Section
3.9.2 (Rev. 2) (BOP Scope)	II.1.d - List snubbers on systems which experience sufficient thermal expansion.	FSAR does not provide a list of snubbers.	3.9B.2
	II.1.e and f - Tests to verify thermal expansion/ vibration measurements.	FSAR does not provide a description of tests.	3.9B.2
	II.2 - Seismic sub- system analysis.	Information is not contained in FSAR Section 3.9B.2.	3.9B.3
3.9.2 (Rev. 2) (NSSS Scope)	II.2.e - Criteria for combining closely spaced modes.	Westinghouse method is provided in FSAR Section 3.7N.3.7.	3.7N.3.7
3.9.3 (Rev. 1) (BOP Scope)	II.1 - Stress limit criteria.	FSAR does not reflect the stress limit criteria.	3.9B.3.1
	II.2 - Information on Class 3 safety/relief devices.	FSAR does not address Class 3 safety/relief devices.	3.9B.3
	II.3 - Information on snubbers.	Requirements not addressed in FSAR.	3.9B.3
3.9.3 (Rev. 1) (NSSS Scope)	II.1 - Design criteria for internal parts of components such as valve discs and pump shafts.	Westinghouse does not provide criteria for the nonpressure boundary portions of ASME Code Class 1, 2, and 3 components in the FSAR.	3.9N.3
	Appendix A, 1.3.3 -Design basis pipe break (DBPB)	Westinghouse defines DBPB as a faulted, not emergency condition.	3.9N
	Appendix A, 3.1 - Stress limits and loading combinations for core support structures.	FSAR does not provide a table defining stress limits and loading combinations for core support systems.	3.9N
3.10 (Rev. 2) (BOP Scope)	II.1.a (14) (b) iii and iv - LOCA induced hydraulic forcing functions or differential pressures upon valve discs or pump rotors.	There is no analysis of LOCA induced hydraulic functions or differential pressures upon valve discs or pump rotors.	3.10B.1
	II.5.c - Seismic Qualification Report format.	Seismic Qualification Report format was not used to document seismic qualification.	3.10B.1
	II - Mechanical equipment seismic and operability qualification.	FSAR does not address mechanical equipment seismic and operability qualification.	3.10B.1
	II.1 - Verify operability of pumps and valves during all operational conditions by test and analysis.	Pump operability has only been performed by analysis.	3.10B.1

TABLE 1.9-1 SUMMARY OF DIFFERENCES FROM SRP (CONTINUED)

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Corresponding FSAR Section
	II.5.b - Listing of systems necessary to perform functions out lined in SRP 3.10.	This list is not included in FSAR Section 3.10.	3.10B.1
3.10 (Rev. 2) (NSSS Scope)	II - Mechanical equipment seismic and operability qualification.	FSAR does not address mechanical equipment seismic and operability qualification.	3.10N
	II.1.a(2) - Testing of equipment in the operational condition.	Flow loads are not superimposed on seismic loads for valve operability tests.	3.10N
	II.1.a(8) - Fixture design for seismic tests.	The seismic qualification testing configurations are designed to represent the typical plant installation.	3.10N
	II.1.a(10) - Static testing of pump or valve assemblies.	End loadings are not applied and all dynamic amplification effects are not included in the static deflection test for active valves.	3.10N
	II.1.a(14)(a) - Operability of active pumps and valves.	Operability of active pumps and valves are not covered in FSAR Section 3.10N.	3.7N, 3.9N
		The Millstone program utilizes a combination of test and analysis to demonstrate operability for active valves.	3.9N
		No specific tests are done on valve gate, disc assemblies, motor, etc.	3.9N
	II.1.a(14)(b) viii - Use of R.G. 1.92 for combination of multimodal and multidirectional responses.	Westinghouse utilizes the methods defined in FSAR Section 3.7 for combining closely spaced modes.	3.7N
	II.1.b(3) - Testing of supports.	Seismic testing of all supports is not conducted.	3.10N
	II.1.c - Seismic and dynamic testing according to IEEE 323-1974.	For some mechanical components, aging and sequence testing was not included.	3.10N
	II.3 - Requirements for central files.	Requirements for central files are not addressed in FSAR Section 3.10N.	3.10N
	II.5.b(1) - Requirement for a list of systems necessary to perform the functions outlined in SRP 3.10.	This list is not included in FSAR Section 3.10N.	3.10N, Chapters 6 and 7

TABLE 1.9-1 SUMMARY OF DIFFERENCES FROM SRP (CONTINUED)

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Corresponding FSAR Section
	II.5.b(2) - Description of the results of any in-plant tests.	Actual test results are not included in the FSAR.	3.10N
	II.5.c - Contents of Seismic Qualification Report (SQR).	Westinghouse does not maintain such a report for Millstone 3.	3.10N
3.11 (Rev. 2) (BOP Scope)	II - Mechanical equipment qualification.	FSAR does not address mechanical equipment qualification.	3.11B.1
	II - NUREG-0588 methodologies.	NUREG-0588 methodologies are not strictly followed.	3.11B.1
3.11 (Rev. 2) (NSSS Scope)	I.1 - Requirement for a list of systems necessary to perform functions outlined in SRP 3.11.	This list is not included in FSAR Section 3.11N.	Chapters 6 and 7, 3.7N, 3.9N, 3.11N
	II - NUREG-0588.	No reference is made in FSAR Section 3.11N to the results of NUREG-0588 study.	3.11N
	Scope of SRP 3.11 -Environmental qualification of mechanical and electrical equipment.	Environmental qualification of mechanical electrical equipment is not addressed in FSAR Section 3.11N.	3.9N
4.5.1 (Rev. 2)	II.4 - On site cleaning requirements.	On site cleaning requirements are not addressed in FSAR Section 4.5.1.	1.8, 17.1.2
5.2.1.1 (Rev. 2)	II - Compliance with 10 CFR 50.55a.	FSAR used ASME III 1971 Ed. through Summer 1972 addenda instead of Winter 1972 addenda for design and fabrication of loop bypass valves.	5.2.2.1
		Westinghouse uses ANS standards rather than R.G. 1.26.	1.8N, 3.2.2
5.2.5 (Rev. 1)	III.7 - Leakage detection testing requirements.	FSAR does not completely address testing of the unidentified leakage sump system.	5.2.5.3.7, 9.3.3
5.3.1 (Rev. 1) (NSSS Scope)	II.6.c(3) - Capsule removal schedule.	The tentative capsule removal schedule is not identical to the removal schedule described in 10 CFR 50, Appendix H, II.C.3.b.	5.3.1.6
5.4.1.1 (Rev. 1) (NSSS Scope)	II.2.b - Normal operating temperature is at least 100°F above the RT_{NDT}.	The FSAR states that the RT_{NDT} is no higher than 10°F.	5.4.1.1.3

TABLE 1.9-1 SUMMARY OF DIFFERENCES FROM SRP (CONTINUED)

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Corresponding FSAR Section
5.4.2.1 (Rev. 2) (NSSS Scope)	BTP MTEB 5-3 -Secondary side chemistry program.	Free hydroxide concentration is not measured. Procedure number and basis not supplied for chemical analysis.	5.4.2.1
	BTP MTEB 5-3, II.2 -Discussion of "clean metal" condition prior to startup.	FSAR does not address this concern.	5.4.2.1
	II.B.2 - Access to remove sludge by lancing from tube support plates.	Tube lancing from tube support plates is not discussed in the FSAR.	5.4.2.1
6.2.1 (Rev. 2) (BOP Scope)	II.f (6.2.1.1A) - Adequate margin above the maximum expected external pressure.	Actual margin of external pressure analysis not specifically addressed.	6.2.1.1.3.5
	II.B.2 (6.2.1.2) -NUREG-0609.	NUREG-0609 was not addressed.	6.2.1.2
6.2.1 (Rev. 2) (NSSS Scope)	II.2 (6.2.1.5) - BTP CSB 6-1 Section B.3b recommends conservative condensing heat transfer coefficients.	Westinghouse values for the long-term post-blowdown condensing heat transfer coefficients are nonconservative.	6.2.1.5
	II.B.3.e (6.2.1.3) - Models for calculating mass and energy releases for containment design basis calculations.	Model described in FSAR differs from Westinghouse model referenced in the SRP.	6.2.1.3.3, 6.2.1.3.4, 6.2.1.3.5, 6.2.1.3.6
6.2.2 (Rev. 3)	II.1.a (6.5.2) - Automatic switchover to recirculation mode.	Millstone 3 switchover is manual.	6.2.2
	II.1.e (6.5.2) - pH between 8.5 and 10.5 for fission product control.	Millstone 3 is designed to a minimum pH of 7.0	6.2.2
6.2.3 (Rev. 2)	II.D.1 and II.D.2 -Discussion heat transfer analysis and high energy line considerations.	FSAR does not provide a discussion of these subjects.	6.2.3
6.2.5 (Rev. 2)	II.3 - Mixing characteristics of the containment (plant specific analyses).	Millstone 3 references the analyses of plants with similar designs.	6.2.5.3
	II.11 - Containment hydrogen monitor requirements.	Hydrogen monitors will comply with these requirements by core load.	6.2.5

TABLE 1.9-1 SUMMARY OF DIFFERENCES FROM SRP (CONTINUED)

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Corresponding FSAR Section
6.3 (Rev. 1)	II - Discussion of non-safety grade interactions with ECCS.	FSAR Section 6.3 does not discuss this subject.	3.6
6.4 (Rev. 2)	II.5.b - Compliance to Regulatory Guide 1.95.	The chlorine detectors are not Seismic Category I.	6.4.3
6.4 Appendix A (Rev. 2)	Single failure (active) criteria.	Valves isolating air inlet ducting located in series.	6.4.3
	Item 6 - No manual action credit allowed for repairs until two hours.	Isolation valves can be manually opened within 10 minutes, thus, credit for manually opening the valves within one hour should be acceptable.	6.4.3
6.5.1 (Rev. 2)	II - Compliance to Regulatory Guide 1.52.	See exceptions listed in Section 1.8.	6.5.1
	II - Continuous indication and recording of air flow for ESF filtration units.	This is not provided.	6.5.1
	II - Flow sensors in ESF filtration units.	These are not provided.	6.5.1
	II - Compliance to Section 8.3.1.6 of ANSI N510-1980.	Three ESF filter train systems do not comply.	6.5.1
	II - Compliance to Sections 8.3.1.5, 8.3.1.6, and 8.3.1.7 of ANSI N510-1980.	Three ESF filter train systems do not comply.	6.5.1
6.5.2 (Rev. 1)	Containment spray as a fission product cleanup system.	Containment spray system requirements are not discussed in Section 6.5.2.	6.5.2
7. 2 (Rev. 2)	BTP ICSB 26 - Sensor qualification.	Sensors for reactor trip on turbine trip when power level is 50% or more are not seismically qualified.	7.2.1.1.2
7.5 (Rev. 2)	III.6 - NUREG-0696 compliance.	The Safety Parameter Display System, and the Emergency Response Facilities are not discussed.	7.5.3
8.3.1 (Rev. 2)	II.4.f - Compliance to NUREG/CR-0660.	NUREG/CR-0660 is not addressed. See Table 1.9-2 for details.	8.3.1
9.1.2 (Rev. 3)	III.2.e - Evaluation of lighter load drops at maximum heights.	This evaluation has not been performed.	9.1.2.3

TABLE 1.9-1 SUMMARY OF DIFFERENCES FROM SRP (CONTINUED)

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Corresponding FSAR Section
9.1.3 (Rev. 1)	II.1.d (4) - BTP ASB 9-2 decay heat removal.	Decay heat removal is based on DECOR (based on ORIGEN2) computer code and credit for evaporative cooling instead of BTP ASB 9-2.	9.1.3.2
9.1.3 (Rev. 1)	III.1.d - For maximum normal heat load, the pool temperature should be kept at or below 140°F	The maximum temperature for a normal heat load is 150°F.	9.1.3.3
9.1.3 (Rev. 1)	III.1.h (ii) - Maximum heat load is after 150 hours of decay	The decay time for the maximum heat load is based on the heat removal capacity of the spent fuel pool heat exchangers and varies from 165 hours to 349 hours.	9.1.3.3
9.1.4 (Rev. 2)	III.6 - Evaluation of lighter load drops at maximum heights.	This evaluation has not been performed.	9.1.4.1
9.2.1 (Rev. 2)	III.3.d - Location of radiation monitors.	No manual valve in series with motor operated valve.	9.2.1
9.2.2 (Rev. 1)	II.3.e - Loss-of-coolant test for reactor coolant-pumps.	Reactor-coolant-pumps have not been tested for the 20 minute time requirement.	9.2.2
9.4.1 (Rev. 2)	II.4 - Compliance to Regulatory Guide 1.95.	The chlorine detectors are not Seismic Category I.	9.4.1.3
9.4.5 (Rev. 2)	II.4 - Protection from	The bottoms of the fresh air intakes are not all located at least 20 feet above grade elevation.	9.4.4
	II.5 - Detection and Control of airborne contamination leakage from the system.	Only normal building ventilation is monitored.	9.4.4.2
	III.3.b - Tornado protection.	No protection of ductwork from negative pressure due to tornado.	9.4.4.4
BTP CMEB 9.5-1 (Rev 2)	Refer to the Fire Protection Evaluation Report, Appendix B, for a comparison of Millstone 3 design to BTP CMEB 9.5-1 guidelines.		Fire Protection Evaluation Report
9.5.4 (Rev. 2)	II.4.b - Regulatory Guide 1.137.	Millstone 3 has two 3.5-day capacity fuel oil tanks.	9.5.4.2
	III.5 - Turbulence of sediments.	There are no tank design features which minimize turbulence of sediments.	9.5.4.2

TABLE 1.9-1 SUMMARY OF DIFFERENCES FROM SRP (CONTINUED)

SRP Section	Specific SRP Acceptance Criteria	Summary Description of Difference	Corresponding FSAR Section
	II.2 (III.6.a) - Missile protection.	The fill lines for the diesel generator fuel oil vaults are protected from missiles.	9.5.4.2
9.5.8 (Rev. 2)	II.4.g (III.8) - Reducing airborne particulate material.	The bottoms of the fresh air intakes are all located at least 20 feet above grade elevation.	9.5.8
10.2.3 (Rev. 1)	II.1 - FATT and Charpy V-notch energies.	GE does not provide data for FATT and Charpy V-notch energies to compare with SRP.	10.2.3
10.3 (Rev. 2)	III.5.d - Tabulation of all flow paths.	FSAR does not tabulate this information.	10.3
11.5 (Rev. 3)	Table 1, Item 6 - Fuel storage area ventilation.	No automatic termination of effluents.	11.5.2.2.9
	Table 2, Item 5 - Spent fuel pool treating system.	No automatic termination of effluents.	9.3.2, 9.4.1
	Table 2, Items 16 and 17 -Steam generator blowdown system.	No automatic termination of effluents.	11.5.2.3.3
12.2 (Rev. 2)	I.2 - Tabulation of concentrations of airborne radioactive materials.	Only normal operation and anticipated operational occurrences are addressed.	12.2.2
13.5.2	II.C.2 - ANSI/ANS 3.2-1981, Section 5.3	FSAR uses ANSI N18.7-1976/ANS 3.2, Section 5.3	13.5.2
14.2 (Rev. 2)	II.4 - Categories of reportable occurrences that are repeatedly being experienced at other facilities.	FSAR does not provide categories of occurrences.	14.2
15.4.6	Entire SRP.	FSAR does not address this accident scenario.	15.4.6
15.4.8 (Rev. 1)	III - Stresses should be evaluated to emergency conditions for these accidents.	Westinghouse considers a faulted condition as stated in ANSI N18.2.	15.4.8
15.6.5 (Rev. 2)	II.3 - TMI Action Plan, II.K.3.30 and II.K.3.31.	No modifications have been made to the small break LOCA model.	15.6.5.3
15.7.3	III.1.a - Radionuclide inventory in failed components	FSAR analyzed postulated tank failure using 1% fuel defects.	2.4.13.3, 15.7.3.2

TABLE 1.9-2 SRP DIFFERENCES AND JUSTIFICATIONS

SRP 2.1.3

SRP TITLE: POPULATION DISTRIBUTION

A. Actual difference between FSAR and SRP

Population distribution wheels conform to the metric radius distances defined in the Environmental Standard Review Plans (ESRPs) and were used to assure conformance with the EROLS. Consequently, a distance of 60 km (37 miles) is used for checking density in FSAR Section 2.1.3.6 instead of a distance of 30 miles specified in SRP 2.1.3, Paragraph II.6.

B. Justification for difference from SRP

Distances were defined to correspond to those being used in the EROLS per the instructions of the ESRP (NUREG-0555) which, at the time of the EROLS preparation, was the most current document offering guidance for EROLS preparation. To assure consistency between the EROLS and FSAR, metric distances were used in both documents.

SRP 2.5.1

SRP TITLE: BASIC GEOLOGIC AND SEISMIC INFORMATION

A. Actual differences between FSAR and SRP

1. SRP 2.5.1, Paragraph II.4.b, requires a discussion on zones of alterations, irregular weathering, and structural weakness. These are not discussed in FSAR 2.5.1.
2. SRP 2.5.1, Paragraph II.4.c, requires unrelieved residual stresses in the rock to be addressed. This is not addressed in FSAR 2.5.1.
3. SRP 2.5.1, Paragraph II.4.d, requires a discussion on unstable materials or areas considered unstable due to physical properties. These are not discussed in FSAR 2.5.1.

B. Justification for differences from SRP

1. Major alteration zones, irregular weathering, and structural weakness do not exist at the site.
2. There is no history of stress relief problems in the area and there were none evident during excavations.
3. Unstable materials and areas considered unstable due to physical properties are discussed in FSAR Sections 2.5.4 and 2.5.5.

SRP 2.5.2

SRP TITLE: VIBRATORY GROUND MOTION

A. Actual differences between FSAR and SRP

1. Magnitudes of earthquakes shown in FSAR Tables 2.5.2–3 and 2.5.2–4 are not identified as M_b, M_L, or M_s as specified in SRP 2.5.2, Paragraph II(2.5.2.1).
2. FSAR Section 2.5.2.4 does not give a probabilistic determination of earthquake return period for the largest earthquakes in each province as specified in SRP 2.5.2, Paragraph II(2.5.2.4).
3. FSAR Section 2.5.2.7 does not give an estimate for the probability of exceeding the acceleration level of the 0BE during the 40-year operating life of the plant as specified in SRP 2.5.2, Paragraph II(2.5.2.7).

B. Justification for differences from SRP

TABLE 1.9-2 SRP DIFFERENCES AND JUSTIFICATIONS (CONTINUED)

1. The magnitude of most earthquakes is based on the relationship between empirical intensity and magnitude. It is not possible to be precise in terms of M_b or M_L for historical earthquakes. None of the magnitudes listed in the FSAR text are surface wave magnitudes (M_S).
2. Refer to the Applicant's response to the NRC Acceptance Review Request Number 230.1.
3. Refer to the Applicant's response to the NRC Acceptance Review Request Number 230.2.

SRP 2.5.3

SRP TITLE: SURFACE FAULTING

A. Actual differences between FSAR and SRP

1. FSAR Section 2.5.3.1: A detailed offshore geologic investigation of 5-mile radius has not been attempted as required in SRP 2.5.3, Paragraph II(2.5.3.1).
2. FSAR Section 2.5.3.4: The location and investigation of every fault within 5 miles of the site has not been performed as required in SRP 2.5.3, Paragraph II(2.5.3.4).
3. FSAR Section 2.5.3.6: Age documentation of every fault within the 5-mile radius has not been performed as required in SRP 2.5.3, Paragraph II(2.5.3.6).

B. Justification for differences from SRP

1. A study of the regional and site geology (on land) was performed, as discussed in FSAR Sections 2.5.1 and 2.5.3. This included extensive on site geologic mapping and age determinations of faults on site. Because the on site geologic investigation did not reveal any recent faulting or unusual features, an offshore geologic investigation was not considered necessary.
2. FSAR Section 2.5.3.4: The Millstone geologic study shows that the last period of faulting at the site occurred approximately 142 million years ago and was related to Triassic-Jurassic rifting or older events. According to the United States Geologic Survey (USGS) geological maps of the area, faults outside the site but within the 5 mile radius, would also be associated with these periods of tectonism. Therefore, locating and identifying every fault within a 5-mile radius of the site was not considered necessary.
3. Because of the above justifications, no detailed offshore geologic investigation within a 5 mile radius was considered necessary, nor was it considered necessary to investigate and determine the age of every fault within 5 miles of the site.

SRP 2.5.4

SRP TITLE: STABILITY OF SUBSURFACE MATERIALS AND FOUNDATIONS

A. Actual difference between FSAR and SRP

SRP 2.5.4, Paragraph II(2.5.4.2), asks for presentation of a table listing the values of parameters used in the analyses of the following properties for saturated soils and clays that underlie the site:

- Liquification potential
- Consolidation behavior
- Static and dynamic behavior

TABLE 1.9-2 SRP DIFFERENCES AND JUSTIFICATIONS (CONTINUED)

FSAR Section 2.5.4.2 does not contain a specific table listing values of parameters used to perform the analyses mentioned above.

B. Justification for difference from SRP

Although the FSAR does not contain a specific table listing values of parameters used to perform the required analyses, the data is available in applicable subsections of Section 2.5.4 as listed below:

- Liquification potential - FSAR Section 2.5.4.8 and its subsections contain values of parameters used in evaluating liquification potential of the underlying soils beneath Category I structures.
- Consolidation behavior - No clay soils are present at the site. The method of analysis to determine elastic settlement of Category I structures on bedrock, glacial till and structural backfill is contained in FSAR Section 2.5.4.10.2. Values used in the analysis to determine settlement of underlying soils are contained in FSAR Section 2.5.4.4.3.
- Static and dynamic behavior - Values of parameters used for evaluating static and dynamic behavior of underlying soils beneath Category I structures are contained in FSAR Sections 2.5.4.2.5, 2.5.4.2.6, and 2.5.4.5.2.

SRP 3.2.2

SRP TITLE: SYSTEM QUALITY GROUP CLASSIFICATION

A. Actual difference between FSAR and SRP

SRP 3.2.2, Subsection II, utilizes Regulatory Guide 1.26 as the acceptance criteria for defining the Quality Groups for components important to safety. In lieu of this Regulatory Guide, Millstone 3 utilizes the classification system provided in ANS 18.2. FSAR Section 3.2.2 provides a cross reference between the ANS safety classifications and the Quality Groups defined in Regulatory Guide 1.26.

B. Justification for difference from SRP

The ANS classification system implemented for Millstone 3 has been endorsed by industry as an acceptable alternative to Regulatory Guide 1.26. This classification system has been used on many other plants and has been accepted by the NRC's Mechanical Engineering Branch. Additionally, SRP 3.2.2, Subsection III, indicates that the NRC will accept alternatives to the Regulatory Guide 1.26 Quality Group classification system provided a correlation between Quality Groups and the classification system used by the applicant is provided in the FSAR. As noted above, such a correlation has been provided in Section 3.2.2 of the FSAR.

SRP 3.4.1

SRP TITLE: FLOOD PROTECTION

A. Actual difference between FSAR and SRP

SRP 3.4.1, Paragraph III.3, discusses the review of postulated failure of nonseismic Category I and nontornado protected tanks. FSAR Section 3.4.1 does not address the postulated rupture effects of these tanks.

TABLE 1.9-2 SRP DIFFERENCES AND JUSTIFICATIONS (CONTINUED)

B. Justification for difference from SRP

Millstone 3 does not have QA Category I tanks that are nonseismic. Postulated failure of nontornado protected tanks has been considered during the review of moderate energy lines as described in FSAR Section 3.6. Items in this category are located outside safety related structures in areas that would preclude flooding of safety related equipment.

Non-QA Category I tanks within safety related structures are not considered to contain sufficient inventory to cause flooding of safety related equipment. In addition, safety related equipment required for safe shutdown of the plant is located in cubicles, or on elevated platforms which would preclude damage due to potential flooding that would result if the non-QA Category I tanks were postulated to fail in a seismic event.

SRP 3.4.2

SRP TITLE: ANALYSIS PROCEDURES

A. Actual difference between FSAR and SRP

SRP 3.4.2, Paragraph II.1, requires that the design basis flood (DBF) or the highest groundwater level and the associated dynamic effects, if any, used in the design shall be the most severe ones that have been historically reported for the site. FSAR 3.4.2 states that structures located above the DBF level are designed for the hydrostatic effects of uplift and water pressure resulting from the DBF or normal groundwater, whichever is most severe.

B. Justification for difference from SRP

FSAR Section 2.5.4.6 describes the groundwater conditions for the Millstone 3 site and includes a description of the low permeability of the bedrock as well as the overlying glacial till. Because of the low permeability of these materials at the site the groundwater level would not significantly change. Therefore, the normal groundwater level and its associated dynamic effects are sufficient for the design of the foundations of the site structures.

Figure 2.5.4–37 shows the map of the stabilized groundwater level contours that were used as the basis for determining the hydrostatic loading on the structure foundations.

SRP 3.5.1.3

SRP TITLE: TURBINE MISSILES

A. Actual difference between FSAR and SRP

SRP 3.5.1.3, Subsection II, requires missile protection for components needed to maintain the reactor in cold shutdown. Cold shutdown maintenance components are not specified as targets in the FSAR.

TABLE 1.9-2 SRP DIFFERENCES AND JUSTIFICATIONS (CONTINUED)

B. Justification for difference from SRP

Evaluation of turbine missiles, as described in FSAR Section 3.5.1.3, has concluded that the probability of a turbine missile being generated and causing damage to a safety related system or component is lower than what is recommended in Regulatory Guide 1.115.

SRP 3.5.1.5

SRP TITLE: SITE PROXIMITY MISSILES (EXCEPT AIRCRAFT)

A. Actual difference between FSAR and SRP

SRP 3.5.1.5, Paragraph III.3, states that the definition of P_p includes probability of missiles that produce secondary missiles which could damage vital equipment. This was not considered in the FSAR analysis.

B. Justification for difference from SRP

FSAR Section 2.2.3 provides the analysis of site proximity missiles (except aircraft). Since the probability stated in this section is sufficiently below the acceptance criteria, consideration of secondary missiles was not deemed necessary. Inclusion of secondary missiles ($P_p = 1$) provides a total probability which is still within the acceptance criteria provided in the SRP.

SRP 3.5.1.6

SRP TITLE: AIRCRAFT HAZARDS

A. Actual difference between FSAR and SRP

SRP 3.5.1.6, Paragraph III.2, indicates an inflight crash rate of 4×10^{-10} per year. FSAR analysis uses the old SRP crash rate of 3×10^{-9}.

B. Justification for differences from SRP

A response to this difference has been provided in the Applicant's response to NRC Acceptance Review Request Number 311.4.

SRP 3.6.1

SRP TITLE:PLANT DESIGN FOR PROTECTION AGAINST POSTULATED PIPING FAILURES IN FLUID SYSTEMS OUTSIDE CONTAINMENT

A. Actual differences between FSAR and SRP

1. BTP ASB 3-1, B.1.a(1), requires an arbitrary split be postulated on the main steam and the feedwater systems at a location proximate to essential systems. The split must be postulated regardless of whether the break exclusion requirements of BTP MEB 3-1, Item B.1.6 are met. The FSAR does not commit to postulate this split.
2. BTP ASB 3-1, B.1.a(2), states that main steam or feedwater piping should not be routed in the vicinity of the control room. The FSAR states that the main steam and feedwater pipes are routed in the vicinity of the control room.
3. BTP ASB 3.1, B.2.a states that essential systems and components should be designed to meet the seismic design criteria of Regulatory Guide 1.29. The BTB defines essential systems and components as those required to shut down the reactor and mitigate the consequences of a postulated piping failure without off site power. FSAR Section 3.6.1.3.1 identifies two high-energy line break isolation systems for the auxiliary steam and hot water heating systems where the isolation valves are in a nonseismic piping system in a nonseismic area.

TABLE 1.9-2 SRP DIFFERENCES AND JUSTIFICATIONS (CONTINUED)

B. Justification for differences from SRP

1. Essential systems, components, or structures are not located within the main steam or feedwater containment penetration area. Environmental effects are of no consequence. The design basis for environmental effects in these areas is given in FSAR Section 3.11, Appendix B.
2. Pipe rupture restraints are provided to prevent main steam pipe whip into the control building wall. The feedwater pipe does not impact the control building in accordance with the discussion in FSAR Section 3.6.1.3.3.
3. Redundant isolation capability is provided for both systems with Category 1E qualified detection and actuation devices located in the Auxiliary Building. In order to provide optimal isolation capability, the isolation valves are located in nonseismic buildings. These valves are not fully seismically qualified. For a nonseismic system in a nonseismic building, seismic qualification of a component in that system is not feasible. However, location of the isolation valves in the nonseismic building is the only practical manner in which to provide the isolation function.

Given a pipe break of the auxiliary steam or hot water heating systems in a safety related building, isolation of the affected system is capable assuming the most limiting single failure. The isolation valves are normally open and fail in the closed position. To ensure continued isolation capability following the postulated pipe break, plant operating procedures require manual valves to be closed to isolate the affected piping.

SRP 3.6.2

SRP TITLE:DETERMINATION OF RUPTURE LOCATIONS AND DYNAMIC EFFECTS ASSOCIATED WITH THE POSTULATED RUPTURE OF PIPING

A. Actual differences between FSAR and SRP

1. SRP 3.6.2, Paragraph III.2.a, states that pressure and temperature values should correspond to the greater contained energy at hot standby or at 102 percent power. FSAR Section 3.6.2.2.1 states that pressure and temperature values associated with reactor operation at 100 percent power are used.
2. SRP 3.6.2, Paragraph III.2.a, states that the allowable capacity of crushable material shall be limited to 80 percent of its rated energy dissipating capacity as determined by dynamic testing at loading rates within ±50 percent of the specified design loading rate. FSAR Section 3.6.2.2.1 commits to 80 percent of energy absorbing capacity but does not commit to dynamic testing to determine energy absorbing capacity.
3. BTP MEB 3-1, B.1.e, states particular criteria for postulating through wall leakage cracks in high energy piping. FSAR Section 3.6.2.1.2 does not commit to postulate through wall leakage cracks in high energy piping.

B. Justification for differences from SRP

1. The FSAR commitment includes the hot standby mode of normal operation. The differences in internal pressure and temperature between 102 percent power and 100 percent power are not significant.

TABLE 1.9-2 SRP DIFFERENCES AND JUSTIFICATIONS (CONTINUED)

2. Dynamic effects on crushable energy absorbing material are not significant. The application is absorbing kinetic energy from pipe whip through relatively small distances. The impact velocities are small so that energy absorbing capacity based on static test data is acceptable.

3. High energy line pipe breaks are more limiting environmentally than high energy through wall leakage cracks in any area where essential systems are located.

SRP 3.7.2

SRP TITLE: SEISMIC SYSTEM ANALYSIS

A. Actual differences between FSAR and SRP

1. SRP 3.7.2, Paragraph II.4, requires consideration of and an envelope of responses from finite element and half space representations of subgrade soil stiffness. FSAR Section 3.7.2 does not address half space representation.

2. SRP 3.7.2, Paragraph II.11, requires consideration of accidental torsion by assuming an additional seismicity of 5 percent of the maximum building dimension at the level under consideration. This is not addressed in the FSAR.

B. Justification for differences from SRP

1. Millstone 3 has committed to use finite element representation of soil stiffness at the CP Stage as described in PSAR Section 3.7.1. Studies that have been conducted on the only structure which is completely soil-founded, the emergency generator enclosure, indicate that the finite element results provided more severe results than the half space representation.

2. Millstone 3 designs were finalized prior to the development of this SRP criteria.

Guidance found in 10 CFR 50 regulation does not require continuous upgrading of the codes and standards used in the design.

SRP 3.7.3

SRP TITLE: SEISMIC SUBSYSTEM ANALYSIS

A. Actual difference between FSAR and SRP (BOP Scope)

FSAR Section 3.7B.3 does not describe the seismic analysis procedures used to account for the seismic motion of non-Category I systems in the seismic design of Category I systems as specified in SRP 3.7.3, Paragraph II.2.l(1). The FSAR currently describes only the seismic analysis procedures used to account for the seismic motion of non-Category I piping in the seismic design of Category I piping.

B. Justification for difference from SRP

Additional information will be provided in an amendment to the FSAR.

TABLE 1.9-2 SRP DIFFERENCES AND JUSTIFICATIONS (CONTINUED)

SRP 3.7.3

SRP TITLE: SEISMIC SUBSYSTEM ANALYSIS

A. Actual difference between FSAR and SRP (NSSS Scope)

SRP 3.7.3, Paragraph II.2.g, requires the closely spaced modes be combined in accordance with Regulatory Guide 1.92. Westinghouse combines closely spaced modes in accordance with the methods described in FSAR Section 3.7N.3.7.

B. Justification for difference from SRP

The Westinghouse methods for combining closely spaced modes represent an alternative to Regulatory Guide 1.92 which has been accepted by the NRC's Structural Engineering Branch and Mechanical Engineering Branch on specific plant dockets. The Westinghouse position on combining closely spaced modes has been accepted on the Seabrook, Catawaba, SNUPPS, Byron, and Comanche Peak dockets.

SRP 3.8.1

SRP TITLE: CONCRETE CONTAINMENT

A. Actual differences between FSAR and SRP

1. FSAR Section 3.8.1 does not reference Regulatory Guide 1.136 as specified in SRP 3.8.1, Paragraph II.2.
2. FSAR Section 3.8.1 does not discuss the ultimate capacity of the reactor containment with respect to failure modes as described in SRP 3.8.1, Paragraph II.4.j.
3. Millstone 3 did not use Article 3000 of ASME III, Division 2, for loads, load combinations, and stress allowables as described in SRP 3.8.1, Paragraph II.5.
4. Millstone 3 did not use ASME III, Division 2, Article CC-3000 for the analysis and design of the containment structure tangential shear as described in SRP 3.8.1, Paragraph II.4.f.

B. Justification for differences from SRP

1. Regulatory Guide 1.136 does not apply to Millstone 3. See FSAR Section 1.8 for position on Regulatory Guide 1.136.
2. The ultimate capacity of the reactor containment with respect to failure modes has been considered in the PRA study, which will be submitted as a separate report.
3. ASME III, Division 2, was not available at the time of the Millstone 3 Construction Permit. ACI 318 and AISC-1969 Ed. were the codes used. ASME III, 1971 Ed., with Addenda through Summer 1973, Subsections NC and NE were used as a guide. Guidance found in 10 CFR 50 regulation does not require continuous upgrading of the codes and standards used in the design.
4. ASME III, Division 2, was not available at the time of the Millstone 3 Construction Permit. The procedure used for analysis and design of the containment structure tangential shear, as described in FSAR Section 3.8.1.4.1, meets the intent of SRP Section 3.8.1, Paragraph II.4.f.

SRP 3.8.3

SRP TITLE:CONCRETE AND STEEL INTERNAL STRUCTURES OF STEEL OR CONCRETE CONTAINMENTS

TABLE 1.9-2 SRP DIFFERENCES AND JUSTIFICATIONS (CONTINUED)

A. Actual difference between FSAR and SRP

ACI 349-76 was not used as described in SRP 3.8.3, Paragraph II.2.

B. Justification for difference from SRP

This code was not in effect at the time of the Construction Permit. ACI 318, AISC-1969 Ed. and ASME III 1971 Ed. through Summer 1973 addenda were the codes used. Guidance found in 10 CFR 50 regulation does not require continuous upgrading of the codes and standards used in the design.

SRP 3.8.4

SRP TITLE: OTHER SEISMIC CATEGORY I STRUCTURES

A. Actual differences between FSAR and SRP

1. ACI 349-76 was not used during the design stage of Millstone 3 as described in SRP 3.8.4, Paragraph II.2.
2. SRP 3.8.4, Paragraph II.4.d, addresses the use of the design report format presented in Appendix C to this SRP. The Applicant's design information is not in this format.

B. Justification for differences from SRP

1. The ASME III 1971 Ed. through Summer 1973 addenda and AISC-1969 Ed. codes were in effect during the design stage of Millstone 3. Guidance found in 10 CFR 50 regulation does not require continuous upgrading of the codes and standards used in the design.
2. The material described in Appendix C of this SRP can be found in the design criteria and design calculations which are contained in an auditable file located at the Millstone 3 site.

SRP 3.8.5

SRP TITLE: FOUNDATIONS

A. Actual difference between FSAR and SRP

ACI 318-71 was used rather than ACI 349-76 as specified in SRP 3.8.5, Paragraph II.4.b.

B. Justification for difference from SRP

ACI 349-76 was not in effect at the time the construction permit was issued. Guidance found in 10 CFR 50 regulation does not require continuous upgrading of the codes and standards used in the design.

SRP 3.9.1

SRP TITLE: SPECIAL TOPICS FOR MECHANICAL COMPONENTS

A. Actual differences between FSAR and SRP (BOP Scope)

1. FSAR Section 3.9B.1.1 identifies plant conditions as normal, upset, emergency, and faulted, whereas SRP 3.9.1, Paragraph III.1, requires them to be identified as Design Level A, B, C, and D.

Also, allowables used in stress analysis are not based on service limits.

2. SRP 3.9.1, Paragraph III.4, requires the FSAR to include justifications as well as the demonstration of acceptability of stress strain curves employed.

However, FSAR Section 3.9B.1.4 only describes the methods and the extent to which these methods are employed in the stress analysis of components, and references ASME Section III provisions.

B. Justification for differences from SRP

TABLE 1.9-2 SRP DIFFERENCES AND JUSTIFICATIONS (CONTINUED)

1. Millstone 3 design is based on ASME III 1971 Ed., which defined plant conditions as normal, upset, emergency and faulted as opposed to Level A, B, C, and D.
2. The piping and components associated with reactor coolant pressure boundary are designed, analyzed, and built to the requirements of ASME Section III. The rigorous requirements of ASME Section III to be satisfied for Code Class 1 systems and components ensure that the requirements of SRP 3.9.1, Paragraph III.4, are met. For noncode components, FSAR Section 3.9B.1.4.5 refers to the analytical procedures used for ASME code components noted in FSAR Section 3.7B.3.1.1.

SRP 3.9.1

SRP TITLE: SPECIAL TOPICS FOR MECHANICAL COMPONENTS

A. Actual difference between FSAR and SRP (NSSS Scope)

SRP 3.9.1, Paragraph II.2, requires a considerable amount of information for all the computer codes used in the design and analysis of Seismic Category I components. Westinghouse only provides a brief description of the computer codes used by Westinghouse for component design and analysis in FSAR Section 3.9N.1.2. Additional information required by the SRP for the computer codes referred to in FSAR Section 3.9N.1.2 is provided by reference to WCAPs-8252 and 8929. Computer codes used by Westinghouse vendors are not included in the FSAR.

B. Justification for difference from SRP

The information requested by the SRP for referenced Westinghouse computer codes is provided in WCAPs-8252 and 8929. Both of these documents have been submitted to the NRC for review. WCAP-8252 has been approved by the NRC, and WCAP-8929 is currently under review by Oak Ridge National Laboratory.

Vendor computer codes are not included in the FSAR because of the large number of codes used and the proprietary nature of this vendor information. Westinghouse assures the acceptability of vendor computer codes through quality assurance audits at vendor facilities (as described in Chapter 17 of the FSAR) and the technical review of various design documents submitted by vendors to Westinghouse.

The NRC's Mechanical Engineering Branch has interpreted this NRC guideline to be applicable only to computer codes used by the major contractors (i.e., NSSS supplier/AE).

SRP 3.9.2

SRP TITLE:DYNAMIC TESTING AND ANALYSIS OF SYSTEMS, COMPONENTS, AND EQUIPMENT

A. Actual differences between FSAR and SRP (BOP Scope)

1. FSAR Section 3.9B.2 does not provide a list of snubbers on systems which experience significant thermal expansion, as required by SRP 3.9.2, Paragraph II.1.d.

FSAR does not provide a description of the tests to be conducted to verify thermal expansion/ vibration measurements as described in SRP 3.9.2, Paragraph II.1.e and f.

2. Information required by SRP 3.9.2, Paragraph II.2, is not contained in FSAR Section 3.9B.2

TABLE 1.9-2 SRP DIFFERENCES AND JUSTIFICATIONS (CONTINUED)

B. Justification for differences from SRP

1. The Technical Specifications will provide details on snubber testing and a list of safety related snubbers.
2. Information requested is provided in FSAR Section 3.9B.3.

SRP 3.9.2

SRP TITLE:DYNAMIC ANALYSIS AND TESTING OF STRUCTURES, SYSTEMS, AND EQUIPMENT

A. Actual difference between FSAR and SRP (NSSS Scope)

SRP 3.9.2, Paragraph II.2.e, defines criteria for combining closely spaced modes. The Westinghouse method for combining closely spaced modes is provided in FSAR Section 3.7N.3.7.

B. Justification for difference from SRP

The Westinghouse methods for combining closely spaced modes represent an alternative to Regulatory Guide 1.92 which has been accepted by the NRC's Structural Engineering Branch and Mechanical Engineering Branch on specific plant dockets. The Westinghouse position on combining closely spaced modes has been accepted on the Seabrook, Catawaba, SNUPPS, Byron, and Comanche Peak dockets.

SRP 3.9.3

SRP TITLE:ASME CODE CLASS 1, 2, AND 3 COMPONENTS, COMPONENT SUPPORTS, AND CORE SUPPORT STRUCTURES

A. Actual differences between FSAR and SRP (BOP Scope)

1. FSAR Section 3.9B.3.1 does not reflect the stress limit criteria of SRP 3.9.3, Paragraph II.1, and Appendix A.
2. SRP 3.9.3, Paragraph II.2, requires information on Class 3 safety/relief devices along with Classes 1 and 2. As written, the FSAR does not specifically address Class 3 safety and relief devices.
3. The requirements of SRP 3.9.3, Paragraph II.3, for snubbers have been fulfilled but have not been included here.

B. Justification for differences from SRP

1. The allowable values utilized by Millstone 3 for piping systems meet the requirements stipulated in ASME Section III, 1971 Ed. through Summer 1972 addenda.

   Components, except piping, use stress criteria of ASME Section III, 1974 Ed., and Code Cases 1606, 1607, 1635, and 1636. These code cases were approved for use in Regulatory Guide 1.84.
2. The design and analysis requirements of Class 3 safety/pressure relief devices are the same as those of Class 2 as described in FSAR Section 3.9B.3.3.
3. Information on snubbers will be discussed in FSAR Chapter 16, Technical Specifications.

SRP 3.9.3

SRP TITLE:ASME CODE CLASS 1, 2, AND 3 COMPONENTS, COMPONENT SUPPORTS, AND CORE SUPPORT STRUCTURE

A. Actual differences between FSAR and SRP (NSSS Scope)

TABLE 1.9-2 SRP DIFFERENCES AND JUSTIFICATIONS (CONTINUED)

1. SRP 3.9.3, Paragraph II.1, requires that design criteria for internal parts of components, such as valve discs and pump shafts, be provided. Westinghouse does not provide criteria for the nonpressure boundary portions of ASME Code Class 1, 2, and 3 components in the FSAR.
2. SRP 3.9.3, Appendix A, Paragraph 1.3.3, defines the design basis pipe break (DBPB) as an emergency condition. For ASME Code Class 1, 2, and 3 components and component supports, Westinghouse defines the DBPB as a faulted condition (see loading combination tables in Section 3.9N).
3. SRP 3.9.3, Appendix A, Paragraph 3.1, requires that stress limits and loading combinations be provided for core support structures. The FSAR does not currently provide tables defining load combinations and stress limits for core support structures.

B. Justification of differences from SRP

1. Westinghouse does not consider it appropriate to provide this type of detail in the FSAR. Westinghouse employs good engineering practice in defining design criteria for critical internal parts. Generally, for critical internal parts of components, equipment specifications limit stresses to the criteria defined in the ASME Code or to the yield strength of the material.
2. Westinghouse defines the DBPB as a faulted condition event consistent with the criteria defined in ANS 18.2. Additionally, Westinghouse considers the stress limits and analysis methods for faulted conditions defined in the ASME Code and FSAR Section 3.9N to be sufficiently conservative to assure the structural integrity and operability of components when subjected to faulted condition loads including the DBPB.
3. A response to this difference will be provided in an amendment to the FSAR.

SRP 3.10

SRP TITLE:SEISMIC AND DYNAMIC QUALIFICATION OF MECHANICAL AND ELECTRICAL EQUIPMENT

A. Actual differences between FSAR and SRP (BOP Scope)

1. SRP 3.10, Paragraph II.1.a(14)(b)iii and iv, requires an analysis of LOCA-induced hydraulic forcing functions or differential pressures upon valve discs or pump rotors. FSAR Section 3.10 does not contain this analysis.
2. Seismic Qualification Report format, as specified in SRP 3.10, Paragraph II.5.c, was not used to document seismic qualification.
3. Mechanical equipment seismic and operability qualification is not addressed in FSAR Section 3.10B.1, as specified in SRP 3.10, Subsection II.
4. SRP 3.10, Paragraph II.1, requires a combination of test and analysis to verify the operability of pumps and valves during all plant operational conditions. Pump operability has only been performed by analysis.
5. SRP 3.10, Paragraph II.5.b, requires a list of systems necessary to perform the functions outlined in SRP 3.10. This list is not included in FSAR Section 3.10B.1.

B. Justification of differences from SRP

TABLE 1.9-2 SRP DIFFERENCES AND JUSTIFICATIONS (CONTINUED)

1. Components and equipment within the system where a LOCA occurs are considered to be rendered inoperable after this event. Valves adjacent to break exclusion areas are supported and/or restrained to maintain stresses within allowable limits to assure operability.
2. Controlled seismic qualification files containing equivalent information are maintained.
3. Seismic qualification of mechanical equipment is described in FSAR Sections 3.7B.1 and 3.9B.1. Pump and valve operability qualification is discussed in FSAR Section 3.9B.3.2.
4. The seismic and operability qualification programs described in FSAR Section 3.9B.3.2 provide adequate assurance of proper equipment performance under all required conditions.
5. Safety related mechanical and electric systems are listed in FSAR Table 3.2–1 and described in FSAR Chapters 6 and7.

SRP 3.10

SRP TITLE:SEISMIC AND DYNAMIC QUALIFICATION OF MECHANICAL AND ELECTRICAL EQUIPMENT

A. Actual differences between FSAR and SRP (NSSS Scope)

1. FSAR Section 3.10N currently addresses only Category I instrumentation and electrical equipment. Mechanical equipment seismic and operability qualification is not discussed in Section 3.10N as specified in SRP 3.10, Subsection II.
2. SRP 3.10, Paragraph II.1.a(2), requires that equipment should be tested in the operational condition and that loadings simulating normal plant conditions should be superimposed on seismic and dynamic loads. This includes flow induced loads and degraded flow conditions. For the tests performed by Westinghouse, operational conditions are included where practical, simulated in some manner, or addressed by analysis. Flow loads are not superimposed on seismic loads for valve operability tests.
3. SRP 3.10, Paragraph II.1.a(8), requires that fixture design for seismic tests should simulate actual service mounting and should not cause any extraneous dynamic coupling to the test item. Westinghouse seismic qualification testing configurations are designed to represent the typical plant installation for the tested component.
4. If the dynamic testing of a pump or valve is impractical, static testing of the assembly is acceptable if conducted in accordance with SRP 3.10, Paragraph II.1.a(10). However, end loadings are not applied and all dynamic amplification effects are not included in the static deflection tests for active valves.
5. FSAR Section 3.10N does not cover operability of active pumps and valves as specified in SRP 3.10, Paragraph II.1.a(14)(a).
6. FSAR Section 3.9N discusses the approach for seismic and operability qualification of safety related mechanical equipment. SRP 3.10, Paragraph II.1.a(14)(a), only allows analysis to be used for demonstrating structural integrity. It further states that operability of active pumps and valves must be demonstrated by test or a combination of test and analysis. The Millstone program utilizes a combination of test and analysis to demonstrate operability for active valves.

TABLE 1.9-2 SRP DIFFERENCES AND JUSTIFICATIONS (CONTINUED)

7. SRP 3.10, Paragraph II.1.a(14)(a), requires that all complex active components other than pump and valve bodies (simple and passive elements) should be tested for operability. The Millstone 3 valve operability program utilizes tests or a combination of test and analysis to demonstrate operability, but no specific tests are done on valve gate, disc assemblies, motors, etc.
8. SRP 3.10, Paragraph II.1.a(14)(b) viii, requires the use of Regulatory Guide 1.92 for combination of multimodal and multidirectional responses in analyses. Westinghouse utilizes the methods defined in FSAR Section 3.7 for combining closely spaced modes.
9. SRP 3.10, Paragraph II.1.b(3), states that supports should be tested with equipment installed or with dummy weights installed. Where seismic testing is conducted, the equipment is mounted as it is installed in the plant (this includes supports). However, seismic testing of all supports is not conducted.
10. SRP 3.10, Paragraph II.1.c, requires that seismic and dynamic testing be performed in sequence in accordance with IEEE 323-1974. For some mechanical components, aging and sequence testing was not included as part of the seismic and operability testing.
11. SRP 3.10, Paragraph II.3, spells out requirements for central files that are not addressed in FSAR Section 3.10.
12. SRP 3.10, Paragraph II.5.b(1), requires a list of systems necessary to perform the functions outlined in SRP 3.10, Subsection I, be included in FSAR Section 3.10. This list is not included in FSAR Section 3.10N.
13. SRP 3.10, Paragraph II.5.b(2), requires that a description of the results of any in-plant tests used to confirm qualification of equipment be included in the FSAR. Actual test results are not included in the FSAR.
14. SRP 3.10, Paragraph II.5.c, requires a seismic qualification report. Westinghouse does not maintain such a report for Millstone 3.

B. Justification for differences from SRP

1. The guidance provided in NUREG-0800 does not conform to the guidelines of Regulatory Guide 1.70 (Standard Format and Content Guide). The Millstone 3 FSAR covers seismic qualification for safety related mechanical equipment in Sections 3.9N and 3.7N. Valve operability is addressed in FSAR Section 3.9N.3.2.
2. Full operational testing conditions are not included in testing because performing such a test is impractical. For example, when static deflection tests on valves are performed, the P across the valve disc is simulated. However, the test is not performed with the valve in a flow loop. As stated above, Westinghouse addresses operational conditions other than by test.

   The active valve operability program defined in FSAR Section 3.9N.3.2 outlines the program for demonstrating operability under all required plant conditions. This program of conservative design, analysis, and test provides adequate assurance that safety related equipment will perform the required safety functions under the appropriate plant conditions.

TABLE 1.9-2 SRP DIFFERENCES AND JUSTIFICATIONS (CONTINUED)

3.	Interface requirements are defined based on the test configuration and other design requirements. Installation is then completed in accordance with the component interface requirements. Any dynamic coupling effects that result from mounting the component in accordance with these interface criteria would have been adequately considered during the test program.
4.	Westinghouse places conservative restrictions on the allowable piping loads transmitted to the valve or pump body such that these loads cannot cause detrimental deflections of the active components. This restriction of allowable piping loads combined with the static deflection testing performed on active valves provides adequate assurance of valve performance and obviates the need to apply end loadings during the static deflection tests for active valves. The Westinghouse operability program for active valves addresses dynamic amplification effects by increasing the g loadings utilized in static deflection tests and analyses when dynamic equipment response is a concern. In most cases, the equipment is rigid and does not display dynamic amplification characteristics.
5.	The Millstone 3 operability program is covered in FSAR Section 3.9N.3.2. The latest version of the SRP has included operability under Section 3.10 and has deleted it from Section 3.9.
6.	Programs currently in effect for Millstone 3 utilize analysis for demonstrating operability of active mechanical equipment such as check valves. For some components (valves with extended structures, etc.), static deflection test programs are utilized in combination with analysis to demonstrate operability. The programs defined for active electrical equipment comply with the guidelines outlined in IEEE 344-197 Regulatory Guide 1.100 and provide adequate assurance of operability under all required conditions. For active valves, the present operability programs confirm the conservative design of these components and provide adequate assurance that these devices can perform their safety function under all required conditions.
7.	The program defined in FSAR Section 3.9N.3.2 provides adequate assurance through the procedures employed that active valves will perform their required safety function under all required conditions.
8.	The Westinghouse methods for combining closely spaced modes has been previously justified and accepted by the NRC.
9.	The supports of safety related equipment are adequately qualified utilizing test or analysis procedures.
10.	For electrical equipment discussed in WCAP-8587, the guidelines provided in IEEE 344-1975 and 323-1974 were followed during the qualification program. For mechanical equipment, there are currently no official guidelines that dictate requirements for aging or sequence testing. The seismic and operability qualification programs implemented for Millstone 3 provide adequate assurance of proper equipment performance under all required conditions.

TABLE 1.9-2 SRP DIFFERENCES AND JUSTIFICATIONS (CONTINUED)

11. Westinghouse qualification documentation is maintained for the 40-year design life in engineering files at Westinghouse. These records are filed and stored in accordance with 10 CFR 50 Appendix B and Regulatory Guide 1.88 as defined in FSAR Chapter 17. Seismic information for the NSSS supplied Class 1E equipment qualified under the WCAP-8587 program is contained the Equipment Qualification Data Packages and Test Reports which Millstone 3 maintains in their central file.
12. The information requested on safety related mechanical and electrical systems is included in FSAR Chapters 6 and7.
13. The test results for Westinghouse supplied equipment are referenced in the FSAR.
14. Seismic qualification of equipment is documented in test reports, analysis reports, calculations, etc., contained in Westinghouse files. The documentation maintained by Westinghouse satisfies existing regulatory requirements and, therefore, it is not considered necessary to prepare an additional Seismic Qualification Report.

SRP 3.11

SRP TITLE:ENVIRONMENTAL QUALIFICATION OF MECHANICAL AND ELECTRICAL EQUIPMENT

A. Actual differences between FSAR and SRP (BOP Scope)
   1. Millstone 3 FSAR does not address mechanical qualification as required in SRP 3.11, Subsection II.
   2. NUREG-0588 methodologies are not strictly followed as required in SRP 3.11, Subsection II.

B. Justification for differences from SRP
   1. Preparation and submittal of information pertaining to environmental qualification of mechanical equipment is pending NRC rulemaking.
   2. A summary comparison of NUREG-0588 will be provided with the Electrical Equipment Qualification Data Packages as a separate report.

SRP 3.11

SRP TITLE:ENVIRONMENTAL QUALIFICATION OF MECHANICAL AND ELECTRICAL EQUIPMENT

A. Actual differences between FSAR and SRP (NSSS Scope)
   1. SRP 3.11, Paragraph I.1, states that all mechanical and electrical systems and equipment necessary to perform the functions listed in SRP 3.11, Subsection I, should be listed in FSAR Section 3.11. FSAR Section 3.11N does not include this list.
   2. No reference is made in FSAR Section 3.11N to the results of the NUREG-0588 study as specified in SRP 3.11, Subsection II.
   3. FSAR Section 3.11N is restricted to electrical equipment only. The environmental qualification of mechanical equipment is not addressed in this section as specified in SRP 3.11.

B. Justification for differences from SRP
   1. The information requested in the SRP is located in different parts of the FSAR as listed below:

      a. Safety Related Mechanical and Electrical Systems - FSAR Chapters 6 and7

TABLE 1.9-2 SRP DIFFERENCES AND JUSTIFICATIONS (CONTINUED)

b. Active Pumps and Valves - FSAR Tables 3.9N–11 and 3.9N–12

c. Class 1E Generic Components - FSAR Table 3.11N–1

d. Plant-Specific Class 1E Components - NUREG-0588 study for Millstone 3

e. Other safety related mechanical components - FSAR Sections 3.7N and 3.9N

2. NUREG-0588 study is not complete.

3. Mechanical equipment qualification for seismic and operability requirements are discussed in FSAR Section 3.9N.

In addition to the tests and analyses discussed in Section 3.9N, Westinghouse designs safety grade mechanical components to accommodate environmental effects through the stringent selection of materials utilized in safety grade mechanical components (e.g., stainless steel, etc.).

Soft parts or consumables such as gaskets, seals, and O-rings are selected for use based on their capability to perform in a nuclear application and are maintained through inservice inspection and maintenance programs. In some cases, partial type tests or separate effects tests have been performed to demonstrate adequacy of selected materials or components for use under adverse environments.

This program for mechanical equipment is based on a combination of design qualification tests and analyses and periodic in plant test and maintenance/surveillance procedures. This program for qualification of safety related mechanical equipment provides adequate assurance that safety grade mechanical components will perform their required functions under all normal, abnormal, accident, and post accident conditions.

SRP 4.5.1

SRP TITLE: CONTROL ROD DRIVE STRUCTURAL MATERIALS

A. Actual difference between FSAR and SRP

SRP 4.5.1, Paragraph II.4, addresses on site cleaning requirements which are not directly referenced in FSAR Section 4.5.1.

B. Justification for difference from SRP

Compliance with the cleanliness requirements of Regulatory Guide 1.37 is described in FSAR Section 1.8 and Appendix VII of the SWEC Topical Report for the construction phase referenced in FSAR Section 17.1.2.

TABLE 1.9-2 SRP DIFFERENCES AND JUSTIFICATIONS (CONTINUED)

SRP 5.2.1.1

SRP TITLE: COMPLIANCE WITH THE CODES AND STANDARDS RULE, 10 CFR 50.55a

A. Actual differences between FSAR and SRP

1. The loop bypass valves are designed and fabricated to ASME III1 971 Ed. through Summer 1972 addenda, whereas 10 CFR 50.55a requires the Winter 1972 addenda.

2. SRP 5.2.1.1, Subsection II, indicates the use of Regulatory Guide 1.26 to meet GDC 1 and 10 CFR 50.55a. However, the NSSS (Westinghouse) uses ANS standards rather than Regulatory Guide 1.26.

B. Justification for differences from SRP

1. Updating the loop bypass valves to a later ASME code addendum would require additional cost and administrative cost burden without a compensating increase in the level of quality or safety. In addition, the actual hardware configuration would not be changed by upgrading to a later code addendum.

2. Components are classified commensurate with the safety function to be performed. FSAR Sections 1.8N and 3.2.2 discuss the Millstone 3 position on Regulatory Guide 1.26.

SRP 5.2.5

SRP TITLE: REACTOR COOLANT PRESSURE BOUNDARY LEAKAGE DETECTION

A. Actual difference between FSAR and SRP

1. FSAR Section 5.2.5 does not address the frequency of testing of the unidentified leakage sump system as required in SRP 5.2.5, Paragraph III.7.

B. Justification for difference from SRP

1. Testing requirements for the unidentified leakage sump system, will be addressed in the Technical Specifications.

SRP 5.3.1

SRP TITLE: REACTOR VESSEL MATERIALS

A. Actual difference between FSAR and SRP (NSSS Scope)

The tentative capsule removal schedule is not identical to the removal schedule described in 10 CFR 50, Appendix H, II.C.3.b, as required in SRP 5.3.1, Paragraph II.6.c(3).

B. Justification for difference from SRP

The schedule described in the FSAR defines more frequent capsule removal in the early years than required by Appendix H and provides adequate standby capsules to meet the Appendix H requirements in later years. This schedule is more conservative than that which is required.

SRP 5.4.1.1

SRP TITLE: PUMP FLYWHEEL INTEGRITY (PWR)

A. Actual difference between FSAR and SRP (NSSS Scope)

SRP 5.4.1.1, Paragraph II.2.b, states that pump flywheel fracture toughness properties are acceptable if the normal operating temperature is at least 100°F above the RT_{NDT}. FSAR Section 5.4.1.1.3 states that the RT_{NDT} is no higher than 10°F.

B. Justification for difference from SRP

TABLE 1.9-2 SRP DIFFERENCES AND JUSTIFICATIONS (CONTINUED)

The pump flywheel will see an operating temperature of at least 110°F once steady state operating conditions have been achieved. In the actual plant environment, the temperature would likely be higher because of the proximity of heat sources such as the reactor coolant circulated through the pump and attached piping.

SRP 5.4.2.1

SRP TITLE: STEAM GENERATOR MATERIALS

A. Actual differences between FSAR and SRP (NSSS Scope)

1. SRP 5.4.2.1 (BTP MTEB 5-3) requires analysis of free hydroxide. Also required are the reference procedures for chemical analysis.
2. SRP 5.4.2.1 (BTP MTEB 5-3, Paragraph II.2) discusses a “clean metal” conditions prior to startup. FSAR Section 5.4.2.1 does not discuss this concern.
3. SRP 5.4.2.1, Paragraph II.B.2, states that access for tooling to remove sludge by lancing from the tube support plates should be provided. This is not discussed in FSAR Section 5.4.2.1.

B. Justification for differences from SRP

1. Free hydroxide will not be analyzed as no additional information on the condition of secondary water chemistry is gained by performing this analysis. Reference procedures for chemical analysis are contained in the Millstone 3 Chemistry Manual.
2. A small amount of corrosion product (oxide) retards further run-away corrosion by acting as a barrier to continued corrosion attack. A “metal clean” condition is therefore unduly restrictive and no real benefit is derived from this requirement. Additionally, the interpretation and quality of visual techniques that may be used to identify a “metal clean” condition can be unreliable and, therefore, conflicting and/or inaccurate conclusions may be drawn about the surface condition of steam generator components.
3. Sludge lancing is performed on the top of the tubesheet to remove the accumulation of corrosion products from the bottom of the steam generator. The sludge accumulates in the low flow area on the top of the tubesheet. Sludge particles do not accumulate to such a great degree on the support plates because of support plate flow slots and, in the case of Model F steam generators, quatrefoil tube support plate holes. These openings allow sludge to filter down (by gravity) to the bottom of the steam generator and accumulate in the low flow area on top of the tubesheet. By selectively placing the blowdown lines in this low flow area of the tubesheet, a large amount of the accumulated sludge can be removed during normal operation of the blowdown system. Any additional sludge which is not removed from the tubesheet by blowdown can be removed during an outage by sludge lancing. Access for sludge lancing of the tubesheet is made possible by access ports in the steam generator shell. Sludge lancing from the steam generator shell access ports is considered more effective than sludge removal from the tube support plates would be.

SRP 6.2.1

SRP TITLE: CONTAINMENT FUNCTIONAL DESIGN

A. Actual differences between FSAR and SRP (BOP Scope)

1. Actual margin of external pressure analysis as required in SRP 6.2.1.1A, Paragraph II.f, is not specifically addressed in FSAR Section 6.2.1.1.

TABLE 1.9-2 SRP DIFFERENCES AND JUSTIFICATIONS (CONTINUED)

2. NUREG-0609 was not addressed in FSAR Section 6.2.1.2 as required in SRP 6.2.1.2, Paragraph II.B.2.

B. Justification for differences from SRP

1. Conservatism of the analysis provides for margin.

2. NUREG-0609 was issued subsequent to the analysis performed for Millstone 3, which used the NUREG-75/087 SRP criteria. FSAR Section 6.2.1.2 describes the methods used to perform the analysis.

SRP 6.2.1

SRP TITLE: CONTAINMENT FUNCTIONAL DESIGN

A. Actual differences between FSAR and SRP (NSSS Scope)

1. BTP 6-1, Section B.3b, recommends conservative condensing heat transfer coefficients which differ from the Westinghouse model.

2. SRP 6.2.1.3, Paragraph II.B.3.e, identifies the Westinghouse model cited in Reference 18 of SRP 6.2.1 as acceptable. This model differs from the Westinghouse model actually used in the Millstone 3 containment design.

B. Justification for differences from SRP

1. It has been determined that Westinghouse values for the long-term post-blowdown condensing heat transfer coefficients are nonconservative. However, the SRP guideline (BTP 6-1) for blowdown heat transfer is four times Tagami values. During blowdown, Westinghouse conservatively uses five times Tagami values. Consequently, the Westinghouse evaluation model for ECCS minimum containment pressure, as presented in Appendix A of WCAP-8339 (1974), has been approved by the NRC staff.

2. The Westinghouse mass and energy release model for containment design is described in FSAR Sections 6.2.1.3.3, 6.2.1.3.4, 6.2.1.3.5, and 6.2.1.3.6. The references are listed in FSAR Section 6.2.7. The current FSAR model is under review by the NRC staff.

SRP 6.2.2

SRP TITLE: CONTAINMENT HEAT REMOVAL SYSTEMS

A. Actual differences between FSAR and SRP

1. SRP 6.5.2, Paragraph II.1.a, requires automatic switchover to recirculation mode. Millstone 3 switchover is manual.

2. SRP 6.5.2, Paragraph II.1.e, requires a pH between 8.5 and 10.5 for fission product control. Millstone 3 is designed to a minimum pH of 7.0.

NOTE:

SRP 6.2.2 refers to SRP 6.5.2 for requirements for “Heat removal only” spray steam. Otherwise, SRP 6.5.2 (Fission Product Removal) would not apply to Millstone 3.

B. Justification for differences from SRP

1. Millstone 3 complies with BTP ICSB 20 which states that manual switchover from injection mode to recirculation mode is sufficient if adequate instrumentation and information display are available to the operator so he can make the correct decision at the correct time. Description of operator action considerations during the switchover from the injection phase to recirculation phase is discussed in FSAR Section 6.3.2.8. Instrumentation and controls available to the operator are provided in FSAR Section 7.5.

TABLE 1.9-2 SRP DIFFERENCES AND JUSTIFICATIONS (CONTINUED)

2. The pH value of 8.5 to 10.5 is related to fission product control. Since no credit is taken for the Millstone 3 spray system for this purpose, a design basis minimum pH of 7.0 was chosen based on material considerations.

SRP 6.2.3

SRP TITLE: SECONDARY CONTAINMENT FUNCTIONAL DESIGN

A. Actual difference between FSAR and SRP

FSAR Section 6.2.3 does not provide a discussion of heat transfer analysis and high energy line considerations as specified in SRP 6.2.3, Paragraphs II.D.1 and II.D.2, respectively.

B. Justification for difference from SRP

Refer to the Applicant's response to the NRC Acceptance Review Request Number 480.4.

SRP 6.2.5

SRP TITLE: COMBUSTIBLE GAS CONTROL IN CONTAINMENT

A. Actual differences between FSAR and SRP

1. SRP 6.2.5, Paragraph II.3, requires a plant specific analysis of the mixing characteristics of the containment. FSAR Section 6.2.5 references the analyses of plants with a similar containment design.
2. SRP 6.2.5, Paragraph II.11, states that the containment hydrogen monitor shall meet the requirements of NUREG-0737, Item II.F.1; NUREG-0718; and the Appendix of Regulatory Guide 1.97.

B. Justification for differences from SRP

1. FSAR Section 6.2.5.3 references the analyses of Surry Power Station, Units 1 and 2 which have a similar containment design (FSAR Section 1.3) and for which the USAEC concluded in the Surry 1 and 2 Safety Evaluation Report that there is adequate mixing of hydrogen in the post-LOCA environment.
2. By fuel load, Millstone 3 will have implemented hydrogen monitors which will comply with these requirements.

SRP 6.3

SRP TITLE: EMERGENCY CORE COOLING SYSTEM

A. Actual difference between FSAR and SRP

No reference is made in FSAR Section 6.3 to studies which demonstrate that nonsafety grade interactions cannot exist which could degrade the performance of the ECCS or its supporting systems as specified in SRP 6.3, Subsection II.

B. Justification for difference from SRP

The effects of failures in non safety related systems due to pipe whip, jet impingement, and/or adverse environment are provided in FSAR Section 3.6.

SRP 6.4

SRP TITLE: CONTROL ROOM HABITABILITY SYSTEMS

A. Actual difference between FSAR and SRP

The chlorine detectors are not Seismic Category I nor Electrical Class 1E (IEEE 323-1974 qualified) as required by SRP 6.4, Paragraph II.5.b. They are redundant and classified as non-seismic.

TABLE 1.9-2 SRP DIFFERENCES AND JUSTIFICATIONS (CONTINUED)

B. Justification for difference from SRP

The redundant chlorine detectors are located in a non-harsh environment (i.e., control equipment room, 50-104°F, 10-60 percent RH, 1.1 x 10^2 Rads). In the event of detector failure, the control room envelope is automatically isolated.

SRP 6.4 APPENDIX A

SRP TITLE:ACCEPTANCE CRITERIA FOR VALVE OR DAMPER REPAIR ALTERNATIVE

A. Actual differences between FSAR and SRP

1. The air inlet ducting is isolated by two low leakage air operated butterfly valves positioned in series. Since the valves are located in series, the arrangement does not meet single failure (active) criteria as described in SRP 6.4, Appendix A, first paragraph.
2. SRP 6.4, Appendix A, Item 6, indicates no manual action credit allowed for repairs until 2 hours.

B. Justification for differences from SRP

As described in FSAR Section 6.4.3, the primary function of the air inlet isolation valves is to isolate the control room, enabling the air bottle pressurization system to pressurize the control room envelope. Following 1 hour of air bottle pressurization, one air bottle train will be exhausted. At this point, either the standby air bottle system can be used or the outside air pressurization system. Should either air inlet isolation valve fail to open automatically at this time, they are within the control room habitability zone and can be manually opened within 10 minutes. Thus, credit for manually opening the isolation valves within 1 hour should be acceptable.

NOTE: The air bottle pressurization system is no longer credited in radiological accident analyses.

SRP 6.5.1

SRP TITLE: ESF ATMOSPHERE CLEANUP SYSTEMS

A. Actual differences between FSAR and SRP

1. SRP 6.5.1, Subsection II, lists Regulatory Guide 1.52 as part of its acceptance criteria. Exceptions have been taken to this Regulatory Guide. See FSAR Section 1.8 for the Millstone 3 compliance to Regulatory Guide 1.52.
2. Continuous indication and recording of air flow for individual ESF filtration units is not provided as described in SRP 6.5.1, Subsection II.
3. Flow sensors are not provided for annunciating high air flows through ESF filtration units as described in SRP 6.5.1, Subsection II.
4. SRP 6.5.1, Subsection II, lists ANSI N510-1980 as part of its acceptance criteria with respect to in-place testing. The three ESF filter train systems listed below will not comply during in-place testing with Section 8.3.1.6 of ANSI N510-1980 which requires the establishment of design flow within ±10 percent of system design flow with a system resistance corresponding to 1.25 times design dirty filter condition. The three noncomplying filter train systems are the SLCRS filter banks, the auxiliary building filter banks, and the control room emergency ventilation filter trains.

TABLE 1.9-2 SRP DIFFERENCES AND JUSTIFICATIONS (CONTINUED)

5. SRP 6.5.1, Subsection II, lists ANSI N510-1980 as part of its acceptance criteria with respect to in-place testing. The three ESF filter train systems listed below will not comply during in-place testing with Sections 8.3.1.5, 8.3.1.6, and 8.3.1.7 of ANSI N510-1980 which require system flow rates to remain constant while the system resistance varies from clean to 1.25 times dirty filter condition. The three noncomplying filter train systems are the SCLRS filter banks, the fuel building filter banks, and the control room emergency ventilation filter trains.

NOTE: The fuel building filter banks are no longer credited as an ESF System in the radiological accident analyses.

B. Justification for differences from SRP

1. Justifications for the exceptions taken to Regulatory Guide 1.52 are found in FSAR Section 1.8.

2. Continuous indication or recording of air flows through individual ESF filtration units is not necessary to ensure reliable system operation. Periodic surveillance tests ensure that system balancing is adequate to maintain operating flow rates through filtration units within design limitations. Additionally, dp alarm setpoints for each ESF filter train can be set to ensure that during all operating conditions flow is maintained within ±10 percent of design flow.

3. Failure of system fans to function is sensed by flow sensors which annunciate low air flow conditions in the control room and automatically start standby units. Sensors to detect high air flow conditions are not necessary since the system is balanced such that the flow is limited to ensure proper performance of the filtration units. Periodic surveillance tests ensure that system flow rates will not exceed unit design parameters.

4. During tests, fans cannot develop a performance greater than for what they were designed. Fan performance requirements were based on Regulatory Guide 1.52 as a design document and the referenced ANSI N509, which do not, and need not, require the development of design flow rate at 1.25 times dirty filter pressure drop conditions. Dirty filter conditions recommended by the manufacturer are factored into the setpoint calculations of the differential pressure switches for annunciating filter changeout requirements. Thus, fan performance within the 10 percent tolerance beyond the setpoint pressure drop is unnecessary.

5. In accordance with fan laws and the laws of fluid flow, as system resistance increases from clean to dirty filter conditions, the system flow rate decreases unless volumetric capacity controls are incorporated into the design of the system to provide for constant system flow. Regulatory Guide 1.52 as a basic design document and the referenced ANSI N509 do not, and need not, require such controls. The identification of system design conditions and the definition of acceptable tolerances of system variables need to be determined by system function on a case-by-case basis as described below for the three ESF systems.

SLCRS Filter Banks

TABLE 1.9-2 SRP DIFFERENCES AND JUSTIFICATIONS (CONTINUED)

The SLCRS is a standby ESF system designed to operate following a LOCA. Its function is to drawdown enclosures contiguous to the containment to a minimum negative pressure of 1/4 inch water gauge in 60 seconds after SIS, and maintain negative pressure conditions for a minimum of 30 days following a LOCA. Flow rates shown on P&IDs are allowances used in design for purposes of equipment and duct sizing; actual flow rates will be determined by test to demonstrate the pressure drawdown characteristics described above. Therefore, the design conditions of the SLCRS are a flow rate to be determined by test, and a system resistance based on dirty filter condition recommended by the filter manufacturer and incorporated into the setpoint calculations, the station operating procedures, and the technical specifications. Also, periodic surveillance testing will ensure that the system operating variables are verified, and if necessary, manually adjusted, to be within the specified limits of the technical specifications.

Fuel Building Filter Banks (This design feature is no longer credited in radiological accident analysis and not an ESF atmospheric cleanup system, the following information is for historical documentation only.)

The fuel building exhaust and filtration system is designed to draw and filter exhaust air during refueling. The function of the filter banks exhaust system is to mitigate the consequences of a fuel handling accident by filtering exhaust air and by preventing uncontrolled outleakage from the fuel building. The design condition of the system is an exhaust flow rate of 41,360 cfm at a system resistance based on dirty filter condition of 10 inches water gauge. With clean filters the variable inlet vanes on the fans suction will be manually adjusted to provide the design flow rate of 41,360 cfm. This provides an exhaust flow rate of 2,360 cfm in excess of the supply air flow rate.

The fuel building exhaust and filtration system is not credited in the radiological analysis for fuel handling accidents.

Control Room Emergency Ventilation Filter Trains

The control room emergency ventilation system is a standby ESF system designed to be manually started one hour after a LOCA. Its function is to continue to maintain the pressurization of the control room habitability zone at a positive pressure of.125 inch water gauge with filtered air after the compressed air bottles have been depleted.

The design condition of the system is 1,000 cfm filtered flow at the system resistance corresponding to dirty filter condition of approximately 10 inch water gauge with an outdoor air makeup to recirculation air ratio of 3:1. With clean filter conditions, the system flow rate increases to 1,225 cfm with the same 3:1 ratio of makeup air to recirculation air.

The control room radiological habitability and the.125 inch water gauge pressurization requirements are both satisfied throughout the entire range of filter train performance from clean to dirty filter condition. Therefore, no volumetric flow control of the pressurization system is required. The system can be balanced at 1,000 cfm ±10 percent for dirty filter condition and allowed to increase beyond the ±10 percent for clean filter condition. The dose analysis in Section 15.6.5.4 and Table 15.6–12 is based on 100 percent outside air makeup of 1,000 cfm as a worst case assumption.

SRP 6.5.2

SRP TITLE: CONTAINMENT SPRAY AS A FISSION PRODUCT CLEANUP SYSTEM

TABLE 1.9-2 SRP DIFFERENCES AND JUSTIFICATIONS (CONTINUED)

A. Actual difference between FSAR and SRP

The containment spray system requirements of SRP 6.5.2 are not discussed in FSAR Section 6.5.2.

B. Justification for difference from SRP

No credit for containment spray fission product removal is taken in the Millstone 3 design.

SRP 7.2

SRP TITLE: REACTOR TRIP SYSTEM (RTS)

A. Actual difference between FSAR and SRP

The sensors (turbine low trip fluid pressure or all stop valves closed) for reactor trip on turbine trip when power level is 50 percent or more are not seismically qualified as specified in BTP ICSB 26.

B. Justification for difference from SRP

The sensors are isolated by digital isolators to prevent degrading the reactor trip system.

SRP 7.5

SRP TITLE: INFORMATION SYSTEMS IMPORTANT TO SAFETY

A. Actual difference between FSAR and SRP

The Safety Parameter Display System and the Emergency Response Facilities are not discussed in the FSAR as required by SRP 7.5, Paragraph III.6.

B. Justification for difference from SRP

As mentioned in FSAR Section 7.5.3, these items are currently being finalized and will be provided in a future amendment.

SRP 8.3.1

SRP TITLE: AC POWER SYSTEMS (ON SITE)

A. Actual difference between FSAR and SRP

NUREG/CR-0660 is not addressed in the FSAR as required by SRP 8.3.1, Paragraph II.4.f.

B. Justification for difference from SRP

NUREG/CR-0660 considerations have been addressed in responses provided to NRC questions. Refer to the 430 series of questions - Question 430.58 through Question 430.134 for details. This NUREG is only applicable to the emergency diesel engine and its support systems as described in Section 9.5.

SRP 9.1.2

SRP TITLE: SPENT FUEL STORAGE

A. Actual difference between FSAR and SRP

SRP 9.1.2, Paragraph III.2.e, requires an evaluation of lighter load drops at maximum heights. This evaluation has not been performed.

B. Justification for difference from SRP

Electrical interlocks and load paths prevent any load from being carried over the spent fuel pool with the new fuel handling crane. Spent fuel bridge and hoist only carries fuel assemblies at their normal lifting height.

TABLE 1.9-2 SRP DIFFERENCES AND JUSTIFICATIONS (CONTINUED)

SRP 9.1.3

SRP TITLE: SPENT FUEL POOL COOLING AND CLEANUP SYSTEM

A. Actual difference between FSAR and SRP

1. Decay heat removal is based on the DECOR computer code (based on ORIGEN2) and credit for evaporative cooling, not BTP ASB 9-2, as required by SRP 9.1.3, Paragraph II.1.d(4).
2. The maximum temperature for a normal heat load is 150°F, a single active failure at 150°F will cause an increase in temperature to approximately 155°F before cooling is restored, not 140°F as required by SRP 9.1.3 Paragraph III.1.d.
3. The decay time for the maximum heat load in the spent fuel pool is based on the heat removal capacity of the spent fuel pool heat exchangers and varies from 165 hours to 349 hours, not 150 hours as required by SRP 9.1.3, Paragraph III.1.h(ii).

B. Justification for difference from SRP

1. Decay heat removal analysis is based on the DECOR computer code (based on ORIGEN2) and credit for evaporative cooling in order to get a more accurate value of decay heat loads.
2. All SSCs associated with the Spent Fuel Pool have been evaluated and have been found to be acceptable for an increase over the SRP limit of 140°F. The decay heat of the fuel is removed and the water coverage of the fuel is maintained for all anticipated scenarios
3. The decay time for the maximum heat load in the spent fuel pool is based on heat removal capacities that are dependent on the actual cooling water temperatures. Colder cooling water temperatures result in greater heat removal capacities which permit larger heat loads to be placed in the pool and shorter decay times.

SRP 9.1.4

SRP TITLE: LIGHT LOAD HANDLING SYSTEM (RELATED TO REFUELING)

A. Actual difference between FSAR and SRP

SRP 9.1.4, Paragraph III.6, requires an evaluation of lighter load drops at maximum heights. This evaluation has not been performed.

B. Justification for difference from SRP

Electrical interlocks and load paths prevent carrying any load over the spent fuel pool with the new fuel handling crane. Spent fuel bridge and hoist only carry fuel assemblies at their normal lifting height.

SRP 9.2.1

SRP TITLE:STATION SERVICE WATER SYSTEM (NUCLEAR SERVICE COOLING WATER SYSTEM)

A. Actual difference between FSAR and SRP

SRP 9.2.1, Paragraph III.3.d, requires that radiation monitors be located on system discharge, and at components susceptible to the leakage, and that these components can be isolated by one automatic and one manual valve in series. There are motor-operated valves at the inlet and discharge of the service water side of the containment recirculation coolers; however, there is no manual valve in series with the motor operated valve.

TABLE 1.9-2 SRP DIFFERENCES AND JUSTIFICATIONS (CONTINUED)

B. Justification for difference from SRP

There is a radiation monitor located on the discharge side of each containment recirculation cooler which alarms in the control room on a high radiation signal. The isolation valves at the inlet and discharge of the coolers are remote-manually operated. When the radiation monitor alarms in the control room, the operator can remotely close the isolation valves. In the event of loss of power to the valve motor operator, a hand wheel may be engaged to locally close the valve. If the valve operator malfunctions, the containment recirculation coolers may be isolated by closing the isolation valves on the containment recirculation system side of the coolers.

SRP 9.2.2

SRP TITLE:REACTOR AUXILIARY COOLING WATER SYSTEMS (COMPONENT COOLING WATER AND AUXILIARY COMPONENT COOLING WATER SYSTEM)

A. Actual difference between FSAR and SRP

1. The Millstone 3 reactor coolant pump has not, at this time, been tested to the 20-minute time requirements as specified in SRP 9.2.2, Paragraph II.3.e.

B. Justification for difference from SRP

1. A program is underway by Westinghouse to comply with the testing requirements.

SRP 9.4.1

SRP TITLE: CONTROL ROOM AREA VENTILATION SYSTEM

A. Actual difference between FSAR and SRP

The chlorine detectors are not Seismic Category I nor Electrical Class 1E (IEEE 323-1974 qualified) as required by SRP 9.4.1, Paragraph II.4. They are redundant and classified as non-seismic.

B. Justification for difference from SRP

The redundant chlorine detectors are used and located in a non-harsh environment (i.e., control equipment room, 50-104°F, 10-60 percent RH, 1.1×10^2 Rads). In the event of detector failure the control room envelope is automatically isolated.

SRP 9.4.5

SRP TITLE: ENGINEERED SAFETY FEATURES VENTILATION SYSTEM

A. Actual differences between FSAR and SRP

1. There are four fresh air intakes for engineered safety features ventilation systems that are not located 20 feet above grade elevation to prevent dust infiltration as required in SRP 9.4.5, Paragraph II.4.
2. SRP 9.4.5, Paragraph II.5, states that the total ventilation system shall have the capacity to detect and control leakage of airborne contamination from the system. The system presently provides for monitoring of the normal building ventilation and not the emergency ventilation.

TABLE 1.9-2 SRP DIFFERENCES AND JUSTIFICATIONS (CONTINUED)

3. SRP 9.4.5, Paragraph III.3.b, requires essential portions of the engineered safety features ventilation systems be protected from the effects of tornados. This SRP requirement also applies to the circulating and service water pumphouse and other yard structures ventilation system (FSAR Section 9.4.7) and the hydrogen recombiner building heating, ventilation, and air conditioning system (FSAR Section 9.4.10). Additionally, Regulatory Guide 1.76 requires structures, systems, and components important to safety be protected against tornado pressure drop and tornado-generated missiles. To meet these requirements, tornado dampers should be provided to prevent the ductwork from collapsing. Only the control building and emergency generator enclosure are protected by tornado dampers.

B. Justification for differences from SRP

1. The four fresh air intakes are located in the ESF, the emergency diesel generator, and the hydrogen recombiner buildings. The intake in the ESF building, for ventilation of the mechanical equipment rooms and auxiliary feedwater pump rooms, has a centerline 15 feet-10 inches above grade. This intake is equipped with a filter rated at 55-60 percent NBS efficiency. Periodic surveillance of the pressure drop across the filter and changing the filter upon setpoint alarm annunciation will preclude excessive dust accumulation. Each emergency diesel generator building ventilation intake has a centerline 19 feet-6 inches above grade. The intake for hydrogen recombiner cubicle 1B in the hydrogen recombiner building has a centerline 18 feet-8 inches above grade. These intakes are sufficiently close to the recommended 20 feet above grade so as to minimize entrainment of dirt or dust.

2. The emergency ventilation a/c system ductwork is designed to be of low leakage construction and only recirculates air, thereby, not allowing a direct path to the mechanical rooms emergency ventilation which utilizes outside air for cooling.

3. Only those ventilation systems that are required for habitability or heat removal requirements after a DBA or LOP were considered for installation of tornado dampers.

   Safety related equipment whose ventilation systems do not have tornado dampers have been qualified for adverse conditions during short-term operation, as described in Section 3.11B.1. Plant operational procedures will address long-term ventilation outages.

SRP BTP CMEB 9.5-1 (Section 9.5.1)

SRP TITLE: GUIDELINES FOR FIRE PROTECTION FOR NUCLEAR POWER PLANTS

Refer to the Fire Protection Evaluation Report, Appendix B, for a comparison of Millstone 3 design to BTP CMEB 9.5-1 guidelines.

SRP 9.5.4

SRP TITLE:EMERGENCY DIESEL ENGINE FUEL OIL STORAGE AND TRANSFER SYSTEM

A. Actual differences between FSAR and SRP

1. SRP 9.5.4, Paragraph II.4.b, requires that each diesel generator be capable of operating continuously for 7 days. Each diesel fuel oil tank at Millstone 3 has a 3.5 day capacity of fuel oil.

2. There are no tank design features which minimize turbulence of sediments as specified in SRP 9.5.4, Paragraph III.5.

TABLE 1.9-2 SRP DIFFERENCES AND JUSTIFICATIONS (CONTINUED)

3. The fill lines for the diesel generator fuel oil vaults are not missile protected as required by SRP 9.5.4, Paragraph III.6.a.

B. Justification for differences from SRP

1. The Applicant can obtain fuel oil from sources nearby within 24-hours after a need for such oil is identified. The Applicant also has determined that, for the power plants within the Northeast Power Coordinating Council area, off site power can be restored to the site 95 percent of the time within a 24-hour period after it is lost. In addition, a loss of off site power has occurred only once in the 14 years the Millstone switchyard has been in operation. Steps have been taken to preclude occurrence of such a LOP (it was caused by salt contamination of insulators) in the future. The Applicant also notes that by running two diesel generators at part load (starting 8 hours after a postulated accident occurred) and judiciously realigning one train of ESF loads between two operating diesel generators, individual engine operating time per each fuel oil storage tank can be extended to 5-6 days.

2. Formulation of corrosive product sediment is minimized by means of a sump and a sump pump with suitable controls for removal of condensation. Additionally, the tank interiors are coated with epoxy resin to preclude corrosion. Filters of progressively smaller mesh size, some of which are alarmed for pressure drop, also assure fuel oil sediment does not become a problem.

3. Alternate ways to fill the tank are provided through the tank manhole or through the flame arrestor/vent line.

SRP 9.5.8

SRP TITLE:EMERGENCY DIESEL ENGINE COMBUSTION AIR INTAKE AND EXHAUST

A. Actual difference between FSAR and SRP

SRP 9.5.8, Paragraph III.8, states that a minimum of 20 feet should exist between the bottom of all fresh air intakes and the grade elevation. The actual design is not consistent with this requirement. The bottom of the intake hoods is at elevation 40 feet-9 inches, which is 16 feet-3 inches above grade.

B. Justification for difference from SRP

The actual design has the center line of the intake at 19 feet-6 inches above grade, and also employs air filter silencers to control dust. Periodic surveillance of the pressure drop across the air filter and changing the filter when necessary will preclude excessive pressure drop from dust.

SRP 10.2.3

SRP TITLE: TURBINE DISK INTEGRITY

A. Actual difference between FSAR and SRP

The turbine manufacturer (GE) states that the turbine materials have the lowest FATT and highest Charpy V-notch energies available, but provides no data for comparison with SRP 10.2.3, Paragraph II.1.

B. Justification for difference from SRP

GE considers data and calculations requested by the SRP to be proprietary information which, if requested, can be made available to the NRC under the provisions of 1 0CFR 2.790.

TABLE 1.9-2 SRP DIFFERENCES AND JUSTIFICATIONS (CONTINUED)

SRP 10.3

SRP TITLE: MAIN STEAM SUPPLY SYSTEM (MSSS)

A. Actual difference between FSAR and SRP

FSAR Section 10.3 does not tabulated and describe all flow paths that branch off the main steam lines between the main steam isolation valves and the turbine stop valves as required by SRP 10.3, Paragraph III.5.d.

B. Justification for difference from SRP

All flow paths are shown on the appropriate P&IDs.

SRP 11.5

SRP TITLE:PROCESS AND EFFLUENT RADIOLOGICAL MONITORING INSTRUMENTATION AND SAMPLING SYSTEMS

A. Actual differences between FSAR and SRP

1. SRP 11.5, Table 1, Item 6, requires an automatic control feature which automatically terminates effluents of the fuel storage area ventilation system. Monitor 3HVR-RE17 extracts a sample from this system but provides no automatic termination.
2. SRP 11.5, Table 2, Item 5, also requires an automatic control feature, which automatically terminates effluents of the spent fuel pool treating system. No such provision is provided on Millstone 3.
3. SRP 11.5, Table 2, Items 16 and 17, require an automatic control feature, which automatically terminates effluents of the steam generator blowdown system.

B. Justification for differences from SRP

1. During fuel handling activities, the fuel building ventilation is processed by the fuel building filtration units. Accident analysis indicates that the filters prevent the release of excessive amounts of radioactive effluent.
2. The spent fuel pool cooling and purification is a closed system; therefore, termination of effluents is unnecessary. Monitoring is accomplished using the reactor plant sampling system radiation monitor, 3SSR-RE08, and area radiation monitors surveying the fuel pool. Safety evaluations described in FSAR Section 9.1.3 show this to be adequate.
3. Monitoring of the steam generator blowdown system is provided by the reactor plant sampling system radiation monitor, 3SSR-RE08. An evaluation of the accident scenario for a steam generator tube rupture shows that such an event would be identified by the air ejector system monitor, 3ARC-RE21 or the main steam line monitors, 3MSS*RE75-78. The main steamline monitors would identify which steam generator is affected and operator action would close valves to prevent release of steam generator blowdown effluents.

SRP 12.2

SRP TITLE: RADIATION SOURCES

A. Actual difference between FSAR and SRP

TABLE 1.9-2 SRP DIFFERENCES AND JUSTIFICATIONS (CONTINUED)

SRP 12.2, Paragraph I.2, requires tabulation of the calculated concentrations of radioactive material, by nuclide, expected during normal operation, anticipated operational occurrences, and accident conditions for equipment cubicles, corridors, and operating areas normally occupied by operating personnel. FSAR Section 12.2 does not tabulate the calculated concentrations of radioactive material expected during accident conditions.

B. Justification for difference from SRP

During accident conditions, local surveys and measurements will be performed as required and exposures will be limited to the requirements of NUREG-0737.

SRP 13.5.2

SRP TITLE: OPERATING AND MAINTENANCE PROCEDURES

A. Actual difference between FSAR and SRP

SRP 13.5.2, Paragraph II.C.2, references Section 5.3 of ANSI/ANS 3.2 - 1981 (Draft 7). The FSAR is based on Section 5.3 of ANSI N18.7 - 1976/ANS 3.2 which is endorsed in Regulatory Guide 1.33.

B. Justification for difference from SRP

The NRC endorses the 1976 version of ANSI/ANS 3.2, Section 5.3 in Regulatory Guide 1.33, Revision 2, which is in accordance with Regulatory Guide 1.70, Revision 3.

Emergency Operating Procedures are developed based on the Westinghouse Owner's Group Emergency Response Guidelines as approved by the NRC. The Emergency Operating Procedures are functional-based as described in the FSAR. Since the requirement for functional-based Emergency Operating Procedures is already explicitly addressed, and the approach is to comply with NRC approved procedure guidelines, little will be gained by committing to a partial standard which is not addressed within existing regulatory guides.

SRP 14.2

SRP TITLE: INITIAL PLANT TEST PROGRAMS - FSAR

A. Actual difference between FSAR and SRP

SRP 14.2, Paragraph II.4, requires the Applicant to have recognized categories of reportable occurrences that are repeatedly being experienced at other facilities. FSAR Section 14.2 does not provide categories of occurrences.

B. Justification for difference from SRP

The review of operating and testing experience in development of the test program is independent of categories and frequency of occurrence. The criterion for incorporation into Millstone 3 procedures or design is the application of the experience to the safety, reliability, and cost of Unit 3.

SRP 15.4.6

SRP TITLE:CHEMICAL AND VOLUME CONTROL SYSTEM MALFUNCTION THAT RESULTS IN A DECREASE IN THE REACTOR COOLANT

A. Actual difference between FSAR and SRP

FSAR Section 15.4.6 does not address this accident scenario.

B. Justification for difference from SRP

TABLE 1.9-2 SRP DIFFERENCES AND JUSTIFICATIONS (CONTINUED)

Based upon the evaluation of current industry reviews and an NRC internal review, the consequences and risk of an inadvertent boron dilution is minimal. Based on cost benefit analysis, modifications also were not shown to be justifiable. Refer to the Applicant's response to the NRC Acceptance Review Request Number 440.8.

SRP 15.4.8

SRP TITLE: SPECTRUM OF ROD EJECTION ACCIDENTS (PWR)

A. Actual difference between FSAR and SRP

SRP 15.4.8, Subsection III, implies that the stresses should be evaluated to emergency conditions. Westinghouse considers this a faulted condition as stated in ANSI N18.2. Faulted condition stress limits are applied for this accident.

B. Justification for difference from SRP

System overpressurization due to a rod ejection transient was evaluated in WCAP-7588, Revision 1-A, and received NRC acceptance in the Topical Report Evaluation.

SRP 15.6.5

SRP TITLE: LOCA INSIDE CONTAINMENT

A. Actual difference between FSAR and SRP

No modifications have been made to the small break LOCA model in accordance with NUREG-0737, Items II.K.3.30 and II.K.3.31, as required by SRP 15.6.5, Paragraph II.3.

B. Justification for difference from SRP

The small break LOCA analysis model, currently approved by the NRC, is conservative and in compliance with Appendix K of 10 CFR 50. However, Westinghouse believes that improvement in the realism of small break calculations is a worthwhile effort and has committed to revise its small break LOCA analysis model to address the NRC concerns mentioned in II.K.3.30 and II.K.3.31 (see FSAR Section 1.10).

SRP 15.7.3

SRP TITLE:POSTULATED RADIOACTIVE RELEASES DUE TO LIQUID-CONTAINING TANK FAILURES

A. Actual difference between FSAR and SRP

The FSAR has analyzed postulated tank failure using 1 percent fuel defects, while SRP 15.7.3, Paragraph III.1.a, uses 0.12 percent fuel defects.

B. Justification for difference from SRP

FSAR analysis is more conservative.

1.10 TMI ACTION ITEMS

Table 1.10-1 presents the MNPS-3 positions on the PWR applicable items from the US Nuclear Regulatory Commission's post-TMI action plan requirements for applicants for an operating license, NUREG-0737, Enclosure 2, dated November 1980.

TABLE 1.10-1 TMI ACTION ITEMS

Item and Title	Position	FSAR/TS Reference
I.A.1.1 Shift Technical Advisor	MPS-3 meets the provisions of the Commission's Policy Statement for providing engineering expertise on shift (STA).	TS 6.2.2 & 6.2.4
I.A.1.2 Shift Supervisor Administrative Duties	MPS-3 meets the requirements of this item.	13.1.2.2
I.A.1.3 Shift Manning	MPS-3 meets the requirements of this item.	TS 6.2.2
I.A.2.1 Immediate Upgrade of RO and SRO Training and Qualifications	MPS-3 meets the requirements of this item.	13.2
I.A.2.3Administration of Training Programs	MPS-3 meets the requirements of this item.	13.2
I.A.3.1 Revise Scope and Criteria for Licensing Examinations	This item is not applicable to MPS-3.	*
I.B.1.2 Evaluation of Organization and Management	MPS-3 meets the requirements of this item.	13.4.4 & TS 6.2.3
I.C.1 Short Term Accident Analysis and Procedures Revision	MPS-3 Emergency Operating Procedures are based on Westinghouse Owners Group emergency procedure guidelines which are approved by the NRC.	13.5.2.1
I.C.2 Shift and Relief Turnover Procedures	MPS-3 meets the requirements of this item.	13.5.1.3
I.C.3 Shift Supervisor Responsibility	MPS-3 meets the requirements of this item.	13.1.2
I.C.4 Control Room Access	MPS-3 meets the requirements of this item.	13.5.1
I.C.5 Procedures for Feedback of Operating Experience	MPS-3 meets the requirements of this item.	13.3.5
I.C.6 Procedures for Verification of Correct Performance of Operating Activities	MPS-3 meets the requirements of this item.	13.5

TABLE 1.10-1 TMI ACTION ITEMS (CONTINUED)

Item and Title	Position	FSAR/TS Reference
I.C.7 NSSS Vendor Review of Procedures	MPS-3 emergency operating procedures are based on NRC approved Westinghouse Emergency Response Guidelines and therefore eliminates the requirements for additional NSSS vendor review of emergency operating procedures.	13.5.2
I.C.8 Pilot Monitoring of Selected Emergency Procedures for NTOLS	MPS-3 emergency operating procedures are based on NRC approved Westinghouse Emergency Response Guidelines and therefore eliminates the requirement for pilot monitoring of selected emergency procedures for near term operating license applicants.	13.5.2
I.D.1 Control Room Design Review	A control room design review was performed for MPS-3 to meet the requirements of this item. Modifications are reviewed in accordance with the Design Control Manual.	18
I.D.2 Plant Safety Parameter Display Console	MPS-3 meets the requirements of this item.	7.5.1
I.G.1 Training during Low-Power Testing	This item was completed prior to low power testing.	*
II.B.1Reactor Coolant System Vents	Safety grade reactor vessel and pressurizer venting capability is provided in the MPS-3 design.	5.4.15, 7.5
II.B.2 Plant Shielding	The MPS-3 plant shielding design is outlined in Chapter 12.	12.3
II.B.3 Post-Accident Sampling	Not applicable to MPS-3 (Reference Technical Specifications Amendment number 201)	*
II.B.4 Training for Mitigating Core Damage	MPS-3 has implemented a training program utilizing the INPO guidelines for Recognizing and Mitigating the Consequences of Severe Core Damage as the basis for the program.	13.2

TABLE 1.10-1 TMI ACTION ITEMS (CONTINUED)

Item and Title	Position	FSAR/TS Reference
II.D.1 Testing Requirements for Reactor Coolant System Relief and Safety Valves	MPS-3 meets the requirements of this item as noted below with the exception with the exception of ATWS testing. MPS-3 meets the requirement of this item for the safety valves via the EPRI test program and for the PORVs through extrapolation of EPRI test data to the MPS-3 specific PORV design. Verification of block valve functionability is demonstrated by application of test data and analyses.	5.4.13
II.D.3 Relief and Safety Valve Position Indication	The pressurizer PORVs have a reliable direct position indication in the control room.	5.4.13, 7.5
II.E.1.1 Auxiliary Feedwater System Evaluation	An auxiliary feedwater system reliability analysis has been performed as specified. Incorporation of cavitating ventures has solved any possibility of excessive flowrate; therefore, MPS-3 meets the requirements of this item.	10.4.9

TABLE 1.10-1 TMI ACTION ITEMS (CONTINUED)

Item and Title	Position	FSAR/TS Reference
II.E.1.2 Auxiliary Feedwater System Automatic. Initiation and Flow Indication	The present design of the MPS-3 auxiliary feedwater system incorporates the requirements to have reliable automatic initiation and flow indications in accordance with IEEE 279-1971. During normal power operation, manual AFW initiation is accomplished by the operator starting the pumps and isolating Steam Generator blowdown lines. At below 10% rated thermal power, the MDAFW pumps may be in normal operation, feeding the steam generators to maintain Steam Generator inventory level. In this mode, there is no automatic function associated with the MDAFW pump flow control valves and operator action is required to open the MDAFW system control valves. The motor driven auxiliary feedwater pumps may be aligned to take suction from the non-safety grade condensate storage tank (CST). Motor-driven auxiliary feedwater pump suction automatically switches to the demineralized water storage tank (DWST), including isolation from the CST, in the event of an SIS, LOP, CDA, two of four low-low water level condition in any one steam generator, or AMSAC signal. Manual initiation may require operator action to realign the motor-driven auxiliary feedwater pump suction source in this mode of operation.	10.4.9, 7.3
II.E.3.1 Emergency Power Supply for Pressurizer Heaters	The emergency power supply for pressurizer heaters meets the requirements of this item.	8.3.1, 5.4.10.3
II.E.4.1 Dedicated Hydrogen Penetrations (Containment Design)	The MPS-3 design includes redundant hydrogen recombiners and analyzers outside the containment. Penetrations for this equipment are dedicated to that service as described in Section 6.2.4 and 6.2.5. Use of recombiners and analyzers is no longer a requirement for DBAs.	6.2.4, 6.2.5
II.E.4.2 Containment Isolation Dependability	MPS-3 meets the requirements of this item.	6.2.4, 7.3.1

TABLE 1.10-1 TMI ACTION ITEMS (CONTINUED)

Item and Title	Position	FSAR/TS Reference
II.F.1 Additional Accident Monitoring Instrumentation	MPS-3 meets the requirements of this item. Hydrogen monitors meet the requirements of 10 CFR 50.44 using guidance of Regulatory Guide 1.7.	11.5, 7.5, 6.2.1, 6.2.5, TS 3.3.3.10, TS 3.3.3.6, TS3.6.4.1
II.F.2 Identification of and Recovery From Conditions Leading to Inadequate Core Cooling	MPS-3 utilizes a subcooled margin monitor system, and a core exit thermocouple system and Combustion Engineering heated junction thermocouple system to meet the requirements of this item.	4.4.6.5
II.G.1 Power Supplies for Pressurizer Relief Valves, Block Valves, and Level Indicators	Emergency power for pressurizer equipment meets the requirements for this item.	5.4.13
II.K.1.5 Review ESF Valves	MPS-3 meets the requirements of this item.	13.5.2
II.K.1.10 Operability Status	MPS-3 meets the requirements of this item	13.5.1
II.K.1.17 Trip per Low-Level B/S	An MPS-3 pressurizer low pressure signal initiates both a reactor trip and the start of safety injection. Pressurizer low-level trips are not utilized on MPS-3.	7.2, 7.3
II.K.2.13 Thermal Mechanical Report	MPS-3 meets the requirements of this item.	5.2.3

TABLE 1.10-1 TMI ACTION ITEMS (CONTINUED)

Item and Title	Position	FSAR/TS Reference
II.K.2.17 Potential for Voiding in the Reactor Coolant System During Transients	MPS-3 meets the requirements of this item Westinghouse has performed a study which addresses the potential for void formation in Westinghouse designed nuclear steam supply systems during natural circulation cooldown/depressurization transients. This study has been submitted to the NRC by the Westinghouse Owners Group (Jurgensen 1981c) and is applicable to MPS-3. The Staff has accepted the study Reference - Safety Evaluation Report for Millstone 3, NUREG-1031. In addition, the Westinghouse Owners Group has developed a natural circulation cooldown guideline that takes the results of the study into account so as to preclude void formation in the upper head region during natural circulation cooldown/depressurization transients, and specifies those conditions under which upper head voiding may occur. These Westinghouse Owners Group generic guidelines have been submitted to the NRC (Jurgensen 1981e). The generic guidance developed by the Westinghouse Owners Group (augmented as appropriate with plant specific consideration) has been utilized in the implementation of MPS-3 plant specific operating procedures.	*
II.K.2.19 Sequential Auxiliary Feedwater Flow Analysis	Not applicable to MPS-3. The NRC has completed a generic review on this subject and concluded that the concerns expressed in this item are not applicable to NSSS with inverted U-tubes such as the one utilized in MPS-3 (Varga 1981).	*

TABLE 1.10-1 TMI ACTION ITEMS (CONTINUED)

Item and Title	Position	FSAR/TS Reference
II.K.3.1 Installation and Testing of Automatic Power-Operated Relief Valve Isolation System	The addition of an automatic isolation system for the PORVs will not be utilized for MPS-3. Modifications implemented under Item II.K.3.2 will reduce the probability of a LOCA caused by a stuck open PORV to an acceptably low level. Also automatic closure of the block valve may inhibit operator detection of a stuck open PORV.	*
II.K.3.2 Report on Overall Safety Effects of Power-Operated Relief Valve Isolation System	MPS-3 meets the requirements of this item. A generic report responsive to this item was produced by Westinghouse (WCAP-9804, 1981). The post-TMI modifications discussed in the above report have been implemented. These modifications will reduce the probability of a small break LOCA caused by a stuck open PORV to an insignificant level relative to all other small break LOCA events. This report determined the frequency of a small-break LOCA caused by a stuck open PORV is reduced to about 2.1×10^{-6} per reactor year for MPS-3 design. This is well below the WASH-1400 medium frequency of 10^{-3} for a small break LOCA.	*
II.K.3.3 Reporting SV and RV Failures and Challenges	The licensee is responsible for ensuring that any failure of a PORV or safety valve to close will be reported promptly to the NRC. All challenges to the PORVs or safety valves will be documented in the annual report.	TS 6.9.1

TABLE 1.10-1 TMI ACTION ITEMS (CONTINUED)

Item and Title	Position	FSAR/TS Reference
II.K.3.5 Automatic Trip of Reactor Coolant Pumps During Loss-of Coolant Accident	Automatic trip of reactor coolant pumps during LOCA is not provided. Westinghouse has performed an analysis of delayed reactor coolant pump trip during small break LOCAs. This analysis is documented in WCAP-9584 and WCAP-9585 (1979). In addition, Westinghouse has performed test predictions of LOFT Experiments L3-1 and L3-6. The results of these predictions are documented in Jurgensen (1981a,b,d). NNECO provided additional information related to Generic Letter 85-12 concerning implementation of TMI Action Item II-K.3.5 in letters dated September 16, 1985, November 19, 1985, and June 30, 1987. NRC Letter dated March 29, 1989 closes out this issue by finding the plant-specific RCP trip setpoint development acceptable. The letter also states there are no longer safety significant concerns for the plant specific information.	*
II.K.3.7 Evaluation of PORV Opening Probability	This item is applicable to B&W plants only and therefore does not apply to MPS-3.	*
II.K.3.9 Proportional Integral Derivative Controller Modification	The MPS-3 design does not include a proportional integral derivative (PID) controller in the power-operated relief valve control circuit (see Figures 7.7-4 and 7.2-1, Sheet 11).	7.7.1.5

TABLE 1.10-1 TMI ACTION ITEMS (CONTINUED)

Item and Title	Position	FSAR/TS Reference
II.K.3.10 Proposed Anticipatory Trip Modification	The MPS-3 design incorporates this trip modification. The NRC has raised the question of whether the pressurizer power-operated relief valves would be actuated for a turbine trip without reactor trip below a power level of 51 percent which is the highest P-9 trip setpoint allowed per Technical Specifications Table 2.2-1. An analysis has been performed using realistic yet conservative values for the core physics parameters (primarily reactivity feedback coefficients and control rod worths), and a conservatively high initial power level of 53 percent. All operating parameters (RCS and secondary temperature, pressure and flow) are at initial values without uncertainties, corresponding to 53 percent power. The transient was initiated from a power level of 53 percent, which is the highest P-9 trip setpoint allowed per Technical Specifications plus 2 percent for power measurement uncertainty. This is a conservative starting point, and would bracket all transients initiated from a lower power level. The core physics parameters used were the ones that would result in the most positive reactivity feedbacks (i.e., highest power levels). NSSS Control Systems (i.e., steam dump, rod control, pressurizer spray) are assumed to be operational in the automatic mode of control. Based upon the results from the analysis, the peak pressure reached in the pressurizer would be 2,328 psia. Which remains below the nominal PORV actuation setpoint of 2.350 psia. The result also indicate that the peak pressure reached in the steam generator would be 1.124 psia which remains below the nominal secondary relief valve actuation setpoint of 1.140 psia.	10.4.4.1, 7.2.1
II.K.3.11 Justification Use of Certain PORVs	The PORVs used in the MPS-3 design are pilot-operated relief valves supplied by Garrett.	*

TABLE 1.10-1 TMI ACTION ITEMS (CONTINUED)

Item and Title	Position	FSAR/TS Reference
II.K.3.12 Confirm Anticipatory Trip	The MPS-3 design includes an anticipatory reactor trip on turbine trip as discussed in FSAR Section 7.2.1.1.2 (Item 6). The logic for this trip is shown on Figure 7.2–1 (Sheet 16).	7.2.1.1.2
II.K.3.17 Report on Outages of Emergency Core Cooling Systems Licensee Report and Proposed Technical Specification Changes	The ECCS Component Reliability and Tracking Program commitment has been satisfied. The required reliability information required by NUREG 0737 on ECCS components involved in outages during the first five years of commercial operation has been compiled, formally documented, and is retrievable on site from Nuclear Document Services for review. The reliability trending and monitoring program committed to within the SER has been superseded by and is currently performed under the existing Maintenance Rule program consistent with the requirements contained with 10 CFR 50.64.	*
II.K.3.25 Effect of Loss of Alternating Current Power on Pump Seals	During normal operation, seal injection flow from the chemical and volume control system is provided to cool the RCP seals, and the component cooling water system provides flow to the thermal barrier heat exchanger to limit the heat transfer from the reactor coolant to the RCP internals. In the event of loss of off site power, the RCP motor is de-energized and both of these cooling supplies are terminated; however, the diesel generators are automatically started and either seal injection flow or component cooling water to the thermal barrier heat exchanger is automatically restored within seconds. Either of these cooling supplies is adequate to provide seal cooling and prevent seal failure due to loss of seal cooling during loss of off site power for at least 2 hours. See Section 8.3 for diesel loading sequence.	*

TABLE 1.10-1 TMI ACTION ITEMS (CONTINUED)

Item and Title	Position	FSAR/TS Reference
II.K.3.30 Revised Small Break Loss-of-Coolant Accident Methods to Show Compliance With 10 CFR Part 50, App. K	MPS-3 meets the requirements of this item.	15.6.5
II.K.3.31 Plant Specific Calculations to Show Compliance With 10 CFR Part 50.46	MPS-3 meets the requirements of this item.	15.6.5 & TS 6.9.1.6
III.A.1.1 Emergency Preparedness, Short Term	Existing plans for Millstone site apply.	13.3
III.A.1.2 Upgrade Emergency Support Facilities	The emergency support facilities identified in the Emergency Plan are functional.	13.3
II.A.2 Emergency Preparedness	Existing plans for the Millstone site apply.	13.3
III.D.1.1 Primary Coolant Sources Outside Containment	MPS-3 has implemented a program to reduce leakage from systems outside containment that would or could contain highly radioactive fluids during a serious transient or accident to as-low-as practical levels.	5.2.5 & TS 6.8.4.a
III.D.3.3 Inplant I Radiation Monitoring	Continuous air monitors with direct readout and alarm capabilities are located in the MPS-3 control room and in the Technical Support and Emergency Operations Centers.	11.5, 12.3, TS 6.8.4.b
III.D.3.4 Control Room Habitability	The requirements of this item have been addressed.	6.4, 2.2.3, 9.4.1

TABLE 1.10-1 TMI ACTION ITEMS (CONTINUED)

Item and Title	Position	FSAR/TS Reference

NOTES:

* Statement stands alone, no FSAR Section reference.

References for Table 1.10-1:
Anderson, T. M. (Westinghouse) September 26, 1980. Letter (NS-TMA-2318) to Eisenhut, D. G. (NRC).
Jurgensen, R. W. (Chairman, Westinghouse Owners Group) March 3, 1981a. Letter (OG-49) to Ross, D. F., Jr. (NRC).
Jurgensen, R. W. March 23, 1981b. Letter (OG-50) to Ross, D. F., Jr. (NRC).
Jurgensen, R. W. April 20, 1981c. Letter (OG-57) to Check, P. S. (NRC).
Jurgensen, R. W. June 15, 1981d. Letter (OG-60) to Check, P. S. (NRC).
Jurgensen, R. W. November 30, 1981e. Letter (OG-64) to Eisenhut, D. G. (NRC).
Rahe, E. P. (Westinghouse) November 25, 1981. Letter (NS-EPR-2524) to Eisenhut, D. G. (NRC).
Varga, S. A. (NRC) June 29, 1981. Letter to Carey, J. J. (Duquesne Light Company).
WCAP-9584 (Proprietary) and WCAP-9585 (Non-proprietary), August 1979 “Analysis of Delayed Reactor Coolant Pump Trip During Small Loss-of-Coolant Accidents for Westinghouse Nuclear Steam Supply System.”
WCAP-9804, February 1981. “Probabilistic Analysis and Operational Data in Response to NUREG-0737 Item II.K.3.2 for Westinghouse NSSS Plants.”
Jaffe, D. H. (NRC), March 29, 1989 Letter to Mroczka, E. J. (NNECO).

1.11 MATERIAL INCORPORATED BY REFERENCE

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

1. Millstone Unit 3 Technical Requirements Manual (TRM).

2. As identified in the List of Figures, the engineering controlled plant drawings that are, coincidentally, MPS-3 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)).