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INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 INTRODUCTION

This Final Safety Analysis Report (FSAR) was submitted in support of an application by Energy Northwest for a Class 103 operating license for a single unit nuclear power plant. The facility is known as the Columbia Generating Station (CGS) and was formerly known as WNP-2.

Energy Northwest was the applicant for the operating license for CGS. The plant was designed, constructed, and is being operated under the responsibility of Energy Northwest.

CGS is located within the Hanford Site of the Department of Energy (DOE), Benton County, Washington, approximately 12 miles north of the City of Richland. The site is approximately 3 miles west of the Columbia River at River Mile 352.

This plant has a boiling water reactor (BWR) nuclear steam supply system (NSSS) designed and supplied by the General Electric Company (GE). The plant utilizes a single-cycle, forced-circulation system and is designated as a BWR/5.

The containment was designed by Burns and Roe, Inc., and consists of primary and secondary containment systems. The primary containment structure is a free-standing steel pressure vessel of a specific design by Pittsburgh Des Moines Steel Co. The vessel contains both a drywell and a suppression chamber, which is consistent with the features of a BWR/Mark II containment.

The secondary containment structure is composed of the reactor building, which completely encloses primary containment.

The authorized maximum rated power level limit of the reactor is 3486 MWt. The design power level limit is 3629 MWt. The net electrical power output is approximately 1190 MWe and the gross electrical output is 1230 MWe.

Energy Northwest was granted an operating license for CGS on December 20, 1983, and the plant began commercial operation on December 13, 1984.

1.2 GENERAL PLANT DESCRIPTION

1.2.1 PRINCIPAL DESIGN CRITERIA

The principal design criteria are presented in two ways. First, they are classified as either a power generation function or a safety function. Second, they are grouped according to system. Although the distinctions between power generation or safety functions are not always clear-cut and are sometimes overlapping, the functional classification facilitates safety analyses, while the grouping by system facilitates the understanding of both the system function and design.

1.2.1.1 General Design Criteria

1.2.1.1.1 Power Generation Design Criteria

a. The plant was designed so that it can be fabricated, erected, and operated to produce electric power in a safe and reliable manner. Plant design conforms to applicable codes and regulations as stipulated in Table 1.2-1;

b. The plant is designed to produce steam for direct use in a turbine-generator unit;

c. Heat removal systems are provided with sufficient capacity and operational adequacy to remove heat generated in the reactor core for the full range of normal operational conditions and abnormal operational transients;

d. Backup heat removal systems are provided to remove decay heat generated in the core under circumstances wherein the normal operational heat removal systems become inoperative. The capacity of such systems is adequate to prevent fuel cladding damage;

e. The fuel cladding, in conjunction with other plant systems is designed to retain integrity throughout the range of normal operational conditions and abnormal operational transients;

f. The fuel cladding can accommodate, without loss of integrity, the pressures generated by fission gases released from fuel material throughout the design life of fuel;

g. Control equipment has been provided to allow the reactor to respond automatically to minor load changes, major load changes, and abnormal operational transients;

h. Reactor power level can be manually controlled;

i. Control of the reactor is possible from a single location;

j. Reactor controls, including alarms, are arranged to allow the operator to rapidly assess the condition of the reactor system and locate system malfunctions; and

k. Interlocks or other automatic equipment are provided as backup to procedural controls to avoid conditions requiring the functioning of nuclear safety systems or engineered safety features (ESF).

1.2.1.1.2 Safety Design Criteria

a. The plant design conforms to applicable codes and regulations;

b. The plant is designed, fabricated, erected, and will be operated in such a way that the release of radioactive materials to the environment is limited to the limits and guideline values of applicable federal regulations pertaining to the release of radioactive materials for normal operations and abnormal transients and accidents;

c. The reactor core is designed so its nuclear characteristics do not contribute to a divergent power transient;

d. The reactor is designed so there is no tendency for divergent oscillation of any operating characteristic, considering the interaction of the reactor with other appropriate plant systems;

e. Gaseous, liquid, and solid waste disposal facilities are designed so the discharge and offsite shipment of radioactive effluents can be made in accordance with applicable regulations;

f. The design provides means by which plant operators can be informed when limits on the release of radioactive material are approached;

g. Sufficient indications are provided to allow determination that the reactor is operating within the envelope of conditions considered by plant safety analysis;

h. Radiation shielding is provided and access control patterns have been established to allow a properly trained operating staff to control radiation doses within the limits of applicable regulations in any mode of normal plant operations;

i. Those portions of the nuclear system that form part of the reactor coolant pressure boundary (RCPB) are designed to retain integrity as a radioactive material barrier following abnormal operational transients and accidents;

j. Nuclear safety systems and ESF act to ensure that no damage to the RCPB results from internal pressures caused by abnormal operational transients and accidents;

k. Where positive, precise action is immediately required in response to abnormal operational transients and accidents, such action is automatic and requires no decision or manipulation of controls by plant operations personnel;

l. Essential safety actions can be carried out by equipment of sufficient redundance and independence such that no single failure of active components can prevent the required actions. For systems or components to which IEEE-279 (Criteria for Protection Systems for Nuclear Power Generating Stations) and/or IEEE-308 (Criteria for Class 1E Electrical systems for Nuclear Power Generating Stations) applies, single failures of both active and passive electrical components were considered in recognition of the higher anticipated failure rates of passive electrical components relative to passive mechanical components;

m. Provisions have been made for control of active components of nuclear safety systems and ESF from the control room;

n. Nuclear safety systems and ESF are designed to permit demonstration of their functional performance requirements;

o. The design of nuclear safety systems and ESF includes allowances for natural environmental disturbances such as earthquakes, tornadoes, floods, and storms at the site;

p. Standby electrical power sources have sufficient capacity to power all nuclear safety systems and ESF requiring electrical power;

q. Standby electrical power sources are provided to allow prompt reactor shutdown and removal of decay heat under circumstances where offsite power sources are not available;

r. Features of the plant that are essential to the mitigation of accident consequences are designed, fabricated, and erected to quality standards that reflect the importance of the safety action to be performed;

s. A primary containment has been provided that completely encloses the reactor system, drywell, and suppression pool. The primary containment employs the pressure suppression concept;

t. The primary containment is designed to retain integrity as a radioactive material barrier during and following accidents that release radioactive material into the primary containment volume;

u. It is possible to test primary containment integrity and leaktightness at periodic intervals;

v. A secondary containment has been provided that completely encloses both the primary containment and fuel storage areas. The secondary containment includes the standby gas treatment (SGT) system for controlling release of radioactive materials leaking from the primary containment in the event of an accident and also has the capability for filtering radioactive materials directly from the primary containment atmosphere during shutdown conditions;

w. The secondary containment has been designed to act as a radioactive material barrier, if required, when the primary containment is open for expected operational purposes;

x. The primary containment and secondary containment, in conjunction with other ESF, limit radiological effects of accidents resulting in the release of radioactive material to the containment vessel to significantly less than 10 CFR 50.67 limits;

y. Provisions have been made for removing energy from within the containment vessel as necessary to maintain the integrity of the containment system following accidents that release energy to the primary containment;

z. Piping that penetrates the primary containment structure and could serve as a path for the uncontrolled release of radioactive material to the environs is automatically isolated whenever such uncontrolled radioactive material release is threatened. Such isolation shall be effected in time to limit radiological effects to less than specified acceptable limits;

aa. Emergency core cooling systems (ECCS) are provided to limit fuel cladding temperature to temperatures below the onset of fragmentation in the event of a loss-of-coolant accident (LOCA);

bb. The ECCS provide for continuity of core cooling over the complete range of postulated break sizes in the RCPB and are redundant;

cc. Operation of the ECCS is initiated automatically when required, regardless of the availability of offsite power supplies and the normal generating system of the plant;

dd. The control room has been shielded against radiation and provided with a high efficiency filtration system so that continued occupancy under accident conditions is possible;

ee. In the event that the control room becomes inaccessible, it is possible to bring the reactor from power range operation to cold shutdown conditions by utilizing the local controls and equipment that are available outside the control room on the remote shutdown control panels;

ff. Backup reactor shutdown capability has been provided independent of normal reactivity control provisions. This backup system has the capability to shut down the reactor from any normal operating condition and subsequently to maintain the shutdown condition; and

gg. Fuel handling and storage facilities are designed to prevent inadvertent criticality and to maintain adequate shielding and cooling of spent fuel. Provision is made for maintaining the cleanliness of spent fuel cooling and shielding water.

1.2.1.2 System Criteria

The principal design criteria for particular systems are listed in the following subsections.

1.2.1.2.1 Nuclear System Criteria

a. The fuel cladding is designed to retain integrity as a radioactive material barrier throughout the design power range. The fuel cladding is designed to accommodate, without loss of integrity, the pressures generated by the fission gases released from the fuel material throughout the design life of the fuel;

b. The fuel cladding, in conjunction with other plant systems, is designed to retain integrity throughout any abnormal operational transient;

c. Those portions of the nuclear system that form part of the RCPB are designed to retain integrity as a radioactive material barrier following abnormal operational transients and accidents;

d. Heat removal systems are provided in sufficient capacity and operational adequacy to remove heat generated in the reactor for the full range of normal operational conditions from plant shutdown to design power and for any abnormal operational transient. The capacity of such systems is adequate to prevent fuel cladding damage;

e. Heat removal systems are provided to remove decay heat generated in the core under circumstances wherein the normal operational heat removal systems become inoperative. The capacity of such systems is adequate to prevent fuel cladding damage. The reactor is capable of being automatically shut down in sufficient time to permit decay heat removal systems to become effective following loss of operation of normal heat removal systems;

f. The reactor core and reactivity control system is designed so that control rod action is capable of bringing the core subcritical and maintaining it so, even with the rod of highest reactivity worth fully withdrawn and unavailable for insertion;

g. The reactor core is designed so that its nuclear characteristics do not contribute to a divergent power transient; and

h. The nuclear system is designed so there is no tendency for divergent oscillation of any operating characteristic, considering the interaction of the nuclear system with other appropriate plant systems.

1.2.1.2.2 Power Conversion System Criteria

Components of the power conversion system have been designed to perform the following basic objectives.

a. Produce electrical power from the steam exiting from the reactor, condense the steam into water, and return the water to the reactor as heated feedwater, with a major portion of its gaseous and particulate impurities removed; and

b. Ensure that any fission products or radioactivity associated with the steam and condensate during normal operation are safely contained inside the system or are released under controlled conditions in accordance with waste disposal procedures.

1.2.1.2.3 Electrical Power Systems Criteria

Sufficient offsite and onsite standby sources of electrical power are provided to attain prompt shutdown and continued maintenance of the plant in a safe condition under all credible circumstances. The power sources are adequate to accomplish all required engineered safety feature functions under postulated design basis accident conditions.

1.2.1.2.4 Radwaste System Criteria

a. The gaseous and liquid radwaste systems are designed to limit the release of radioactive effluents from the plant during normal operation within those limits specified in 10 CFR 20 and 10 CFR 50, Appendix I;

b. The solid radwaste disposal system is designed so that during normal operation offsite shipments will be in accordance with applicable regulations, including 10 CFR 20, 10 CFR 71, and 49 CFR 171 through 10 CFR 179, as appropriate; and

c. The design of the systems provide means by which plant operations personnel are alerted whenever operational limits on the release of radioactive material are approached.

1.2.1.2.5 Auxiliary Systems Criteria

a. Fuel handling and storage facilities are designed to prevent criticality and to maintain adequate shielding and cooling for spent fuel. Provision is made for maintaining the cleanliness of spent fuel cooling and shielding water;

b. Other auxiliary systems, such as standby service water (SW), high pressure core spray (HPCS) SW, fire protection (FP), heating and ventilating, communications, and lighting systems, are designed to function during normal, abnormal, and/or accident conditions; and

c. Auxiliary systems that are not required to effect safe shutdown of the reactor or maintain it in a safe shutdown condition are designed such that failure of these systems shall not prevent the essential auxiliary systems from performing their design functions.

1.2.1.2.6 Shielding and Access Control Criteria

a. Radiation shielding is provided and access control patterns are established to allow a properly trained operating staff to control radiation doses within the limits of published regulations in any normal mode of plant operation; and

b. The control room is shielded against radiation and has a high efficiency filtration system, so that occupancy is possible under accident conditions and

TEDE doses are less than those set by Criterion 19 of 10 CFR Part 50, Appendix A and 10 CFR 50.67.

1.2.1.2.7 Nuclear Safety Systems and ESF Criteria

Principal design criteria for nuclear safety systems and ESF correspond to criteria j through q, aa through cc, and ee through ff in Section 1.2.1.1.2.

1.2.1.2.8 Process Control Systems Criteria

The principal design criteria for the process control systems are listed for the nuclear system, the power conversion system, and the electrical power system:

a. Nuclear System Process Control Criteria

    1. Control equipment is provided to allow the reactor to respond automatically to load changes within design limits.

    2. It is possible to manually control the reactor power level.

    3. Control of the reactor is possible from a central location.

    4. Nuclear systems process controls and alarms are arranged to allow the operator to rapidly assess the condition of the nuclear system and to locate process system malfunctions.

    5. Interlocks or other automatic equipment are provided as a backup to procedural controls to avoid conditions requiring the actuation of nuclear safety systems or ESF.

b. Power Conversion System Process Control Criteria

    1. Control equipment is provided to control the reactor pressure throughout its operating range.

    2. The turbine is able to respond automatically to minor changes in load.

    3. Control equipment in the feedwater system maintains the water level in the reactor vessel at the optimum level required by steam separators.

    4. Control of the power conversion equipment is possible from a central location.

5. Interlocks or other automatic equipment are provided in addition to procedural controls to avoid conditions requiring the actuation of ESF.

c. Electrical Power System Process Control Criteria

1. The redundant portions of the Class 1E power systems are designed with either division of the system being adequate to safely shut down the unit.

2. Protective relaying is used to detect and isolate faulted equipment from the system with a minimum of disturbance in the event of equipment failure.

3. Primary and secondary undervoltage relays are located on the 4.16-kV Class 1E equipment buses to isolate these buses from the normal auxiliary power system in the event of Class 1E bus under voltage and to initiate starting of the standby power system diesel generators.

4. Standby power diesel generators' start is initiated by control relays. The generators are also loaded by a sequenced control system to meet the existing emergency condition.

5. All electrically operated breakers can be operated from the main control room.

6. Metering for essential generators, transformers, and circuits is monitored in the main control room.

1.2.1.3 Plant Design Criteria

The plant design criteria are based on general design criteria given in Appendix A of 10 CFR Part 50. Conformance to these criteria is discussed in Section 3.1. The classification of structures, components, and systems is discussed in Section 3.2.

The principal regulations are codes that are used extensively in plant design are highlighted in Table 1.2-1. Note that the codes listed may not be applicable in their entirety. The many codes and regulations applicable to individual systems or structures are discussed throughout the FSAR.

The plant shielding and radiation zone classification can be found in Table 1.2-2. Chapter 12 provides further details.

1.2.2 PLANT DESCRIPTION

1.2.2.1 Site Characteristics

1.2.2.1.1 Site Location and Size

Columbia Generating Station (CGS) is located in the southeast area of the Department of Energy (DOE) Hanford Reservation in Benton County, Washington. The site is approximately 3 miles west of the Columbia River at River Mile 352, approximately 12 miles north of the City of Richland, 18 miles northwest of Pasco, and 21 miles northwest of Kennewick. The site is approximately square shaped with a corridor extending to the makeup water pump house located on the Columbia River as shown in Figure 1.2-1. The CGS site encompasses an area of approximately 1089 acres.

1.2.2.1.2 Description of Site Environs

1.2.2.1.2.1 Site Land. See Section 2.1 for site land description.

1.2.2.1.2.2 Population. See Section 2.1 for population description.

1.2.2.1.2.3 Land Use. Natural physical characteristics of the site which make it well-suited for operation of the plant include: favorable geographical, geological, and seismological characteristics; adequate water supply; ideal climatological characteristics; and remoteness from population centers or areas of special ecological concern. The site area had served as a nuclear industrial center since 1943 when it was selected by the federal government as the location for construction of one of the world's first nuclear production reactors. Since 1943, nine plutonium production reactors and a number of test reactors have been constructed and operated at the Hanford Site.

1.2.2.1.2.4 Meteorology. The climate around CGS is basically continental with a wide range of annual temperatures. See Section 2.3 for additional information.

1.2.2.1.2.5 Hydrology. The Columbia River is the major surface water resource of the region. The river also forms a potential discharge boundary for the aquifer. The surface soils at Hanford are sufficiently permeable to take in water from precipitation and industrial discharges. See Section 2.4 for additional information.

1.2.2.1.2.6 Geology. The Hanford site lies in the east central part of the Pasco Basin, a structural and topographic depression in the Columbia Plateau. The region is underlain by three major geologic units: (a) Tertiary basaltic lavas and intercalated sediments of the Columbia River Group at the base, (b) Plio-Pleistocene sediments of the Ringold Formation, and (c) the Pasco (glaciofluvial) gravels and associated sediments of late Pleistocene age at the surface. See Section 2.5 for additional information.

1.2.2.1.2.7 Seismology. The CGS site is situated in an area characterized by low seismicity and widely scattered epicenters. See Section 2.5 for additional information.

1.2.2.1.3 Design Basis Depending on Site Environs

a. Offgas System

An offgas (OG) system consisting of hold-up piping, charcoal adsorbers, and an elevated release is provided for the controlled release of gaseous effluent to the atmosphere. Gaseous releases will be as low as reasonably achievable (ALARA) in accordance with 10 CFR Part 50, Appendix I, and less than 10 CFR Part 20 limits;

b. Liquid Waste Effluents

Liquid waste will be processed and recycled, and releases of excess inventory will be such that concentrations at the point of discharge will be as low as reasonably achievable in accordance with 10 CFR Part 50, Appendix I, and less than 10 CFR Part 20 limits;

c. Wind Loading and Seismic Design

The structures and components whose failure might cause a design basis accident or result in an uncontrolled release of radioactive fission products will be designed to resist wind loads of tornado velocity and earthquake ground motions which are significantly higher than those expected to occur at the site during the service life of the plant; and

d. Flooding

The maximum assumed flood elevation for design purposes is the sum total of the elevations of water due to the following effects:

1. Breach of any of the upstream dams due to seismic forces,
2. High flow in the Columbia River, and
3. Wind and wave action.

1.2.2.2 General Arrangement of Structures and Equipment

The principal structures located on the plant site are the following:

a. Reactor building - the building that houses the major portion of the nuclear steam supply system (NSSS), the drywell, suppression pool, primary containment, new and spent fuel pools, refueling equipment, and ECCS;

b. Radwaste and control building - the building that houses the liquid and solids radwaste systems, components of the OG system, and the main control room;

c. Turbine building - the building that houses the power conversion equipment;

d. Diesel generator building - the building that houses the standby diesel generators, diesel fuel oil (DO) storage tanks, and associated controls and instrumentation;

e. Circulating water pump house (Wind River Building) - a structure housing the main circulating water (CW) pumps, plant service water (TSW) pumps, and FP pumps;

f. Standby service water pump houses - structures that house the redundant standby SW pumps and the HPCS SW pump;

g. Spray ponds - cooling ponds provided as the ultimate heat sink (UHS);

h. Makeup water pump house - a structure that houses the cooling tower makeup (TMU) water pumps;

i. General service building (Yakima Building) - a structure that houses the potable water (PWC) storage tank, demineralized water (DW) storage tank, offices for plant administration, lunch room, and machine shop;

j. Transformer yard;

k. Condensate storage tanks (CSTs);

l. Cooling towers; and

m. Plant Engineering Center (Deschutes Building).

The arrangement of these structures on the plant site is shown in Figure 1.2-1. The arrangement of the equipment inside the main buildings is shown in Figures 1.2-2 through 1.2-24.

1.2.2.3 Symbols Used on Engineering Drawings

Figure 1.2-25 defines General Electric's (GE) piping and instrumentation symbols, and Figure 1.2-26 through 1.2-28 shows Burns and Roe piping and instrumentation symbols. Figure 1.2-29 defines the logic symbols used on NSSS functional control diagrams.

1.2.2.4 Nuclear System

The nuclear system includes a direct-cycle, forced-circulation, GE boiling water reactor (BWR) that produces steam for direct use in the steam turbine. A heat balance showing the major parameters of the nuclear system for the rated power conditions is shown in Figure 10.1-1.

1.2.2.4.1 Reactor Core and Control Rods

Fuel for the reactor core consists of slightly enriched uranium dioxide pellets sealed in Zircaloy-2 tubes. These tubes (or fuel rods) are assembled into individual fuel assemblies. Gross control of the core is achieved by movable, bottom-entry control rods. The control rods are cruciform in shape and are dispersed throughout the lattice of fuel assemblies. The control rods are positioned by individual control rod drives (CRDs).

Each fuel assembly has several fuel rods with gadolinia (Gd_2O_3) mixed in solid solution with UO_2. The Gd_2O_3 is a burnable poison which diminishes the reactivity of the fresh fuel. It is depleted as the fuel reaches the end of its first cycle.

A conservative limit of plastic strain is the design criterion used for fuel rod cladding failure. The peak linear heat generation for steady-state operation is well below the fuel damage limit even late in life. Experience has shown that the control rods are not susceptible to distortion and have an average life expectancy many times the residence time of the fuel loading.

1.2.2.4.2 Reactor Vessel and Internals

The reactor vessel contains the core and supporting structures; the steam separators and dryers; the jet pumps; the control rod guide tubes; the distribution lines for reactor feedwater (RFW), HPCS, low-pressure core spray (LPCS), and standby liquid control (SLC); the in-core instrumentation; and other components. The main connections to the vessel include main steam (MS) lines, reactor recirculation (RRC) lines, RFW lines, CRD and in-core nuclear instrument housings, HPCS and LPCS lines, residual heat removal (RHR) lines, SLC line, core differential pressure line, jet pump pressure-sensing lines, and water level instrumentation.

The reactor vessel is designed and fabricated in accordance with applicable codes for a pressure of 1250 psig. The nominal operating pressure in the steam space above the separators is 1035 psia. The vessel is fabricated of low-alloy steel and is clad internally with stainless steel (except for the top head, and certain nozzles and nozzle weld zones which are unclad).

The reactor core is cooled by demineralized water that enters the lower portion of the core and boils as it flows upward around the fuel rods. The steam leaving the core is dried by steam separators and dryers located in the upper portion of the reactor vessel. The steam is then directed to the turbine through the MS lines. Each MS line is provided with two MS isolation valves (MSIVs) in series, one on each side of the primary containment barrier.

1.2.2.4.3 Reactor Recirculation System

The RRC system pumps reactor coolant through the core. This is accomplished by two recirculation loops external to the reactor vessel but inside the primary containment. Each external loop contains a mechanical pump, two motor-operated maintenance valves, and one flow control valve which is mechanically blocked full open. The two motor-operated valves are used as pump suction and pump discharge shutoff valves. The flow control valves are no longer used to control reactor power level and therefore are kept in a mechanically blocked full open position.

The internal portion of the loop consists of the jet pumps, which contain no moving parts. The jet pumps provide a continuous internal circulation path for the major portion of the core coolant flow. The jet pumps are located in the annular region between the core shroud and the vessel's inner wall. Any recirculation line break would still allow core flooding to approximately two-thirds of the core height, the level of the inlet of the jet pumps.

1.2.2.4.4 Residual Heat Removal System

The RHR system is a system of pumps, heat exchangers, and piping that fulfills the following functions:

a. Removes decay and sensible heat during and after plant shutdown;

b. Injects water into the reactor vessel, following a LOCA, rapidly enough to reflood the core and maintain fuel cladding below the fragmentation temperature independent of other core cooling systems. This is further discussed in Section 1.2.2.5.8;

c. Removes heat from the primary containment, following a LOCA, to limit the increase in primary containment pressure. This is accomplished by cooling and recirculating the suppression pool water (containment cooling) and by spraying the drywell and suppression pool air spaces (containment spray) with suppression pool water; and

d. Removes some of the airborne radioactivity from the primary containment atmosphere following a LOCA by spraying the drywell.

1.2.2.4.5 Reactor Water Cleanup System

The reactor water cleanup (RWCU) system recirculates a portion of reactor coolant through a filter-demineralizer to remove particulate and dissolved impurities from the reactor system under controlled conditions. It also removes excess coolant from the reactor system under controlled conditions.

1.2.2.4.6 Nuclear Leak Detection System

The nuclear leak detection (LD) system consists of temperature, pressure, flow, and fission-product sensors with associated instrumentation and alarms. This system detects and annunciates leakage in the following systems:

a. Main steam system,
b. Reactor water cleanup system,
c. Residual heat removal system,
d. Reactor core isolation cooling (RCIC) system,
e. Reactor feedwater system,
f. High-pressure core spray system,
g. Low-pressure core spray system,
h. Reactor recirculation system, and
i. Reactor pressure vessel (RPV) flange.

Small leaks generally are detected by temperature and pressure changes, fill-up rate of drain sumps, and fission-product concentration inside the primary containment. Large leaks are also detected by changes in reactor water level and changes in flow rates in process lines.

1.2.2.5 Nuclear Safety Systems and Engineered Safety Features

1.2.2.5.1 Reactor Protection System

The reactor protection system (RPS) initiates a rapid, automatic shutdown (scram) of the reactor, if required, to prevent fuel cladding damage or nuclear system process barrier damage following abnormal operational transients. The RPS overrides all operator actions and process controls and is based on a fail-safe design philosophy that allows appropriate protective action even if a single component failure occurs.

1.2.2.5.2 Neutron Monitoring System

Although not all portions of the neutron monitoring system qualify as a nuclear safety system, those that provide high neutron flux signals to the RPS do. The intermediate range monitors (IRMs) and average power range monitors (APRMs), which monitor neutron flux via in-core detectors, signal the RPS to scram in time to prevent excessive fuel cladding damage as a result of overpower transients. The APRM modules also provide inputs to the thermal power monitors (TPMs) which approximate fuel thermal conditions and also provide scram signals to the RPS.

1.2.2.5.3 Control Rod Drive System

When a scram is initiated by the RPS, the CRD system inserts the negative reactivity necessary to shut down the reactor. Each control rod is controlled individually by a hydraulic control unit. When a scram signal is received, high-pressure water stored in an accumulator in the hydraulic control unit forces its control rod into the core.

1.2.2.5.4 Control Rod Drive Housing Supports

Control rod drive housing supports are located underneath the reactor vessel near the control rod housings. The supports limit the travel of a control rod in the event that a control rod housing is ruptured. The supports prevent a nuclear excursion as a result of a housing failure and thus protect the fuel barrier.

1.2.2.5.5 Control Rod Velocity Limiter

A control rod velocity limiter is attached to each control rod to limit the velocity at which a control rod can fall out of the core should it become detached from its CRD. This action limits the rate of reactivity insertion resulting from a rod drop accident. The limiters contain no moving parts.

1.2.2.5.6 Pressure Relief System (Nuclear System)

A pressure relief system consisting of safety/relief valves (SRVs) mounted on the MS lines is provided to prevent excessive pressure inside the nuclear system following either abnormal operational transients or accidents.

1.2.2.5.7 Reactor Core Isolation Cooling System

The RCIC system provides makeup water to the reactor vessel when the vessel is isolated. The RCIC system uses a steam-driven turbine-pump unit and operates automatically in time and with sufficient coolant flow to maintain adequate water level in the reactor vessel.

1.2.2.5.8 Emergency Core Cooling System

Four ECCS are provided to maintain fuel cladding below fragmentation temperature in the event of a breach in the RCPB that results in a loss of reactor coolant. The systems are

a. High-pressure core spray system,
b. Automatic depressurization system (ADS),
c. Low-pressure core spray system, and
d. Low-pressure coolant injection (LPCI), an operating mode of the RHR system.

1.2.2.5.8.1 High-Pressure Core Spray System. The HPCS system provides and maintains an adequate coolant inventory inside the reactor vessel to maintain fuel cladding temperatures below the fragmentation temperature in the event of breaks in the RCPB. The system is initiated by either high pressure in the drywell or low water level in the vessel. It operates independently of all other systems over the entire range of pressure differences from greater than normal operating pressure to zero. The HPCS cooling decreases vessel pressure to enable the low pressure cooling systems to function. The HPCS system is powered by its own diesel generator if auxiliary power is not available, and the system may also be used as a backup for the RCIC system.

1.2.2.5.8.2 Automatic Depressurization System. The ADS rapidly reduces reactor vessel pressure during a LOCA situation in which the HPCS system fails to maintain the reactor vessel water level. The depressurization provided by the system enables the low pressure ECCS to deliver cooling water to the reactor vessel. The ADS uses some of the relief valves that are part of the nuclear system pressure relief system. The automatic relief valves are arranged to open when conditions indicate that the HPCS system is not delivering sufficient cooling water to the reactor vessel to maintain the water level above a preselected value. The ADS will not be activated unless either the LPCS or LPCI pumps are operating. This is to ensure that adequate coolant will be available to maintain reactor water level after the depressurization.

1.2.2.5.8.3 Low-Pressure Core Spray System. The LPCS system consists of one independent pump and the valves and piping to deliver cooling water to a spray sparger over the core. The system is actuated by conditions indicating that a breach exists in the RCPB but water is delivered to the core only after reactor vessel pressure is reduced. This system provides the capability to cool the fuel by spraying water into each fuel channel. The LPCS loop functioning in conjunction with either the ADS or HPCS can maintain the fuel cladding below the prescribed temperature following a LOCA.

1.2.2.5.8.4 Low-Pressure Coolant Injection. The LPCI is an operating mode of the RHR system, but is discussed here because the LPCI mode acts as an engineered safety feature in conjunction with other ECCS. The LPCI uses the pump loops of the RHR to inject cooling water directly into the pressure vessel. The LPCI is actuated by conditions indicating a breach in the RCPB, but water is delivered to the core only after reactor vessel pressure is reduced. The LPCI operation provides the capability of core reflooding, following a LOCA, in time to maintain the fuel cladding below the prescribed temperature limit.

1.2.2.5.9 Primary Containment

1.2.2.5.9.1 Functional Design. The primary containment is part of the overall containment system which provides the capability to reliably limit the release of radioactive materials to the environs subsequent to the occurrence of the postulated LOCA so that offsite doses will be below the limits stated in 10 CFR Part 50.67. Its design employs an over-and-under, steel pressure vessel which houses the reactor vessel, the RRC loops, and other branch connections of the reactor primary system. The pressure suppression system consists of a drywell, a pressure suppression chamber which stores a large volume of water, a connecting submerged vent system between the drywell and water pool, isolation valves, containment cooling system, and other service equipment. In the event of a RCPB piping failure within the drywell, reactor water and steam would be released into the drywell air space. The resulting increase of drywell pressure would then force a mixture of air, steam, and water through the vents into the pool of water which is stored in the suppression pool, resulting in a rapid pressure reduction in the drywell. Air which is transferred to the suppression chamber, pressurizes the suppression chamber, and is subsequently vented back to the drywell.

1.2.2.5.9.2 Drywell Cooling System. The drywell cooling system is based on recirculating cooling water through the drywell air-handling units to maintain the required ambient temperature. Air is distributed through ductwork and/or up through the annular space between the reactor vessel insulation and the sacrificial shield wall. Air is distributed to areas requiring cooling, such as the RRC motors, the CRD area, and the bellows area. Return air is ducted back to the operating units. The arrangement simplifies the design, operation, and air distribution balance of the system.

Reactor building closed cooling water (RCC) is supplied to the air handling units to dissipate absorbed heat only under normal and loss of power conditions.

The drywell cooling system is not required for safe shutdown, but it is designed with redundant equipment and powered from essential buses to ensure continuous operation to satisfy the power-generation design objective.

The drywell cooling system is designed to operate during offsite power loss. Control switches for operating the equipment are located in the main control room.

1.2.2.5.9.3 Suppression Pool Cooling. The containment cooling subsystem of the RHR system is placed in operation to limit the temperature of the water in the suppression pool following a design basis LOCA, to control the pool temperature during normal operation of the SRVs and the RCIC system, and to reduce the pool temperature following an isolation transient. In the containment cooling mode of operation, the RHR main system pumps take suction from the suppression pool and pump the water through the RHR heat exchangers where cooling takes place by transferring heat to SW. The fluid is then discharged back to the suppression pool or the RPV.

1.2.2.5.9.4 Containment Spray. The redundant containment spray cooling subsystems of the RHR system provide containment cooling for postaccident conditions. Water pumped through the RHR heat exchangers can be diverted to spray headers in the drywell and above the suppression pool. The spray removes energy from the drywell atmosphere by condensing the water vapor. The drywell spray also removes particulate fission product from the drywell atmosphere. Approximately 5% of this flow can be directed to the suppression chamber to cool the gas above the water surface.

1.2.2.5.9.5 Containment Atmosphere Control. In the event of a LOCA, hydrogen and oxygen will be generated in the reactor. Containment atmosphere control is provided by inerted containment, containment atmosphere mixing, and hydrogen and oxygen monitoring in a post-LOCA event.

1.2.2.5.10 Primary Containment and Reactor Vessel Isolation System

The primary containment and reactor vessel isolation system includes sensors, trip channels, control switches and remotely activated valve closing mechanisms associated with the valves, which, when closed, effect isolation of the primary containment or reactor vessel or both.

The purpose of the system is to provide timely protection against the onset and consequences of accidents involving the gross release of radioactive materials from the fuel and the nuclear system process barrier. The primary containment and reactor vessel isolation control system initiates automatic isolation of the RCPB and the primary containment vessel whenever monitored variables exceed preselected operation limits.

All pipelines that both penetrate the primary containment and offer a potential release path for radioactive material are provided with redundant isolation capabilities.

1.2.2.5.11 Main Steam Line Isolation Valves

Although all pipelines that both penetrate the containment and offer a potential release path for radioactive material are provided with redundant isolation capabilities, the main steam lines, because of their large size and large mass flow rates, are given special isolation consideration. Automatic MSIVs are provided in each MS line. Each is powered by both air pressure and spring force. These valves fulfill the following objectives:

a. Prevent excessive damage to the fuel barrier by limiting the loss of reactor coolant from the reactor vessel resulting from either a major leak from the steam piping outside the primary containment or from a malfunction of the pressure control system resulting in excessive steam flow from the reactor vessel,

b. Limit the release of radioactive materials (i.e., iodine spiking) by isolating the RCPB in case of a rapid depressurization of RPV and resulting release of radioactive materials from the fuel to the reactor cooling water and steam, and

c. Limit the release of radioactive materials by closing the primary containment barrier in case of a major leak from the nuclear system inside the primary containment.

1.2.2.5.12 Main Steam Line Flow Restrictors

A venturi-type flow restrictor is installed in each MS line. These devices limit the loss-of-coolant from the reactor vessel before the MSIVs are closed in case of an MS line break outside the primary containment.

1.2.2.5.13 Main Steam Line Radiation Monitoring System

The main steam line radiation monitoring system consists of four gamma radiation monitors located externally to the main steam lines just outside the containment. The monitors are designed to detect a gross release of fission products from the fuel. On detection of high radiation, the trip signals generated by the monitors are used to initiate a closure to the reactor water sample valves, mechanical vacuum pump trip, the mechanical vacuum pump lines isolation, and alarms.

1.2.2.5.14 Standby Service Water and High-Pressure Cooling Spray Service Water Systems

The SW system consists of two completely redundant systems. Each system consists of a pump and piping supplying the associated RHR system heat exchanger, standby diesel generator, essential heating, ventilating, and air conditioning (HVAC) coolers, RHR pump seal coolers, SW motor bearing coolers, and sample coolers with safety grade cooling water from the UHS spray ponds. The Division I SW system also provides cooling water to the LPCS motor bearing cooler.

Cooling water is supplied during a postulated LOCA to the RHR heat exchangers to remove heat when the containment cooling mode of the RHR system is placed in operation. During normal operation, SW is also supplied to the RHR heat exchangers for the shutdown function of the RHR system.

The SW is available to the shell side of the fuel pool cooling and clean up (FPC) system heat exchangers in the event that the normal cooling water supply from the RCC system becomes unavailable.

The HPCS SW system shares spray pond A with the SW system. The pump supplies cooling water to the HPCS diesel generator and the essential HVAC coolers for the HPCS diesel generator and HPCS pump areas.

Cooling water is supplied to all diesel generator cooling systems whenever the diesel generators are started.

1.2.2.5.15 Reactor Building - Secondary Containment

The reactor building completely surrounds the primary containment. The building provides secondary containment when the primary containment is closed and in service, and serves as the primary barrier during operations with the potential to drain the reactor vessel (OPDRV). The reactor building also houses refueling and reactor servicing equipment, new and spent fuel storage facilities, and other reactor safety and auxiliary systems. Secondary containment is not required during movement of irradiated fuel assemblies or core alterations.

The design of the reactor building includes provisions for seismic load resistance and low infiltration and exfiltration rates. The building consists of poured-in-place, reinforced-concrete exterior walls up to the refueling floor. Above this level, the building structure is steel frame with insulated metal siding with sealed joints. Access to the building is through interlocked double doors.

1.2.2.5.16 Reactor Building Ventilation Exhaust Radiation Monitoring System

The reactor building ventilation exhaust radiation monitoring system consists of a number of radiation monitors arranged to monitor the activity level of the ventilation exhaust from the reactor building and primary containment. Upon detection of high radiation, the reactor building is automatically isolated and the SGT system is started.

1.2.2.5.17 Standby Gas Treatment System

The SGT system consists of two identical filter trains. Each filter train consists of a filter unit, two fans, ductwork, and associated valves.

Either filter train may be considered as an installed spare with the other train capable of passing the required amount of air. Either train alone is capable of exchanging the total reactor building volume once in a 24 hr period.

Each filter unit contains electric heaters, a prefilter, high-efficiency particulate filters (water and fire resistant), an iodine filter (high ignition temperature), and instrumentation to measure temperature and flow.

The system maintains a slightly negative internal building pressure and can process all gaseous effluent prior to its discharge from the reactor building.

All equipment is connected to the essential buses and is started either automatically or manually from the main control room.

1.2.2.5.18 Standby Alternating Current Power Supply System

The standby ac power supply system consists of two diesel generator sets, switchgear, and associated distribution system equipment and auxiliaries.

These diesel generator sets are associated with redundant (Divisions 1 and 2) separation divisions; each diesel generator set serves a particular division. The capacity of each diesel generator set is sufficient to attain shutdown under both normal and LOCA conditions, in the event that both the offsite and the normal auxiliary power sources are unavailable to supply plant loads. Since load distribution is such that redundant auxiliary systems are separated by division, safe shutdown can be achieved with only one of the two diesel generators operating.

The standby ac power supply system diesel generators and associated equipment are designed to Class 1E standards and are located within Seismic Category I structures. Equipment of each division is separated so that failure of any component of one division will not jeopardize proper functioning of the other division.

Although it is not a part of the standby ac power supply system, another independent diesel generator unit supplies ac power exclusively to the HPCS system (see Section 1.2.2.5.8.1) in the event that both the offsite and the normal auxiliary power sources are unavailable to supply plant loads.

The HPCS diesel generator may also be cross connected to either Division 1 or to Division 2 as described in Section 8.3.1.1.7.2.1.

1.2.2.5.19 Direct Current Power Supply System

The dc power supply system consists of station batteries, battery chargers, distribution equipment, and related auxiliaries.

The dc system furnishes power at three voltage levels: 250 V, 125 V, and +24 V. The 250-V and 125-V subsystems supply power to both Class 1E and non-Class 1E loads; the 24-V subsystem supplies power for the startup range and power range neutron monitoring systems.

The primary power sources for the system are the dc output station battery chargers. Station batteries associated with each charger operate in a "float-charge" configuration to ensure maintaining the batteries in a fully charged condition. In the event of loss of charger dc output, the station batteries furnish a secondary source of dc supply.

The 125-V and +24-V dc power supply subsystems are each divided into electrically and physically independent divisions. Each battery, together with its independent battery charger, is associated with one of the segregated divisions. The batteries and their associated chargers are located in separate rooms.

The ampere-hour capacity of each battery is capable of supplying all essential loads for a minimum of 2 hr in the event that dc output from the battery chargers is lost.

1.2.2.5.20 Standby Liquid Control System

Although not intended to provide prompt reactor shutdown, as the control rods are, the Standby Liquid Control (SLC) system provides a redundant, independent, and alternate method to bring the nuclear fission reaction to subcriticality and to maintain a subcritical condition as the reactor cools. The system makes possible an orderly and safe shutdown in the event that not enough control rods can be inserted into the reactor core to accomplish shutdown in the normal manner. The system is sized to counteract the positive reactivity effect from rated power to the cold, clean shutdown condition.

The SLC system is also used to maintain the suppression pool pH greater than 7.0 following a LOCA to minimize re-evolving gaseous iodine fission products to the containment atmosphere.

1.2.2.5.21 Safe Shutdown from Outside the Main Control Room

In the event that the control room becomes inaccessible, the reactor can be brought from power range operation to cold shutdown conditions by the use of local controls and equipment that are available outside the control room.

1.2.2.5.22 Main Steam Line Isolation Valve Leakage Control System (Deactivated)

The main steam line isolation valve leakage control (MSLC) system was designed to minimize the fission products which could bypass the SGT system after a LOCA. The MSLC system is not credited for accident mitigation and is no longer needed; MSLC is administratively de-activated. Connections between MSLC and other systems are physically isolated, MSLC components are de-energized, closed, or otherwise taken out of service.

1.2.2.5.23 Fuel Pool Cooling and Cleanup System

The FPC system provides for the removal of decay heat from stored spent fuel and maintains specified water temperature, purity, clarity, and level. This prevents boiling of the pool water and controls the buildup of excessive radioactive materials in the cooling water, thereby minimizing potential radiation exposure to plant personnel. The cooling portion of the system is designed to Seismic Category I requirements and may be isolated from the Seismic Category II cleanup portion of the system by automatic Seismic Category I isolation valves which actuate on low-fuel pool water level. Normally the RCC system furnishes non-safety grade cooling water to the FPC system. If required, safety grade cooling and makeup water is available to the FPC system from the SW system.

1.2.2.6 Power Conversion System

1.2.2.6.1 Turbine Generator

The turbine is an 1800 rpm, tandem-compound (one double-flow high-pressure turbine and three double-flow low-pressure turbines), reheat unit with an electrohydraulic governor for normal operation. The turbine generator is provided with an emergency trip system for turbine overspeed. The rating of the turbine generator is 1,173,046 kW.

The generator is a direct-driven, three-phase, 60 Hz, 25,000 V, 1800 rpm, hydrogen inner-cooled, synchronous generator rated at 1,230 MVA at 0.975 power factor, 0.58 short circuit ratio at a maximum hydrogen pressure of 78 psig.

1.2.2.6.2 Main Steam System

The MS system consists of four 26-in. diameter lines (which expand to 30-in. diameter lines inside the turbine building) extending from the outermost MSIVs to the main turbine stop valves. The use of four main steam lines permits testing of the turbine stop valves and MSIVs during station operation with only a minimum of load reduction. The design pressure and temperature of the MS system from the outermost MSIV to the turbine stop valve is 1250 psig at 575°F. Other features include drains and parts of the turbine bypass system.

1.2.2.6.3 Main Condenser

The main condenser is a triple-pressure, single-pass, deaerating-type condenser with a divided water box. The condenser includes provisions for accepting up to 25% of the MS flow at design conditions from the turbine bypass system and serves as a heat sink for several other flows, such as exhaust steam from the RFW pump turbines, cascading heater drains, feedwater heater shell operating vents, and condensate pump suction vents.

1.2.2.6.4 Main Condenser Evacuation System

The main condenser evacuation system is designed to remove noncondensable gases from the condenser, including air inleakage and dissociation products originating in the reactor, and to continuously exhaust them to the gaseous radwaste system during operation. The system consists of two 100%-capacity, twin-element first stage and single-element second stage steam jet air ejector units complete with intercondensers for normal plant operation and a mechanical vacuum pump for use during startup. Discharge from the vacuum pumps during startup is routed to the elevated release point.

1.2.2.6.5 Turbine Gland Seal System

The turbine gland seal system is designed to provide a means of preventing air leakage into or radioactive steam leakage out of the turbine. The system consists of two 100% steam evaporators, steam seal pressure regulators, steam seal header, gland seal steam condenser and blowers, and the associated piping, valves, and instrumentation.

1.2.2.6.6 Steam Bypass System and Pressure Control System

A turbine bypass system is provided which passes steam directly to the main condenser under the control of the pressure regulator. Steam is bypassed to the condenser whenever the reactor steaming rate exceeds the load permitted to pass to the turbine generator. The capacity of the turbine bypass system is 25% of the turbine design steam flow. The Digital Electro-Hydraulic (DEH) control system provides main turbine control (governor) valve and bypass valve position demands so as to maintain a nearly constant reactor pressure during normal plant operation.

1.2.2.6.7 Circulating Water System

The CW system provides the condenser with a continuous supply of cooling water. It is a closed system utilizing forced draft cooling towers. Makeup water to the system is provided from TMU pumps located in an intake structure on the Columbia River. The makeup water replaces the water lost by evaporation, drift, and blowdown.

1.2.2.6.8 Condensate and Feedwater System

The condensate and feedwater system pumps condensate from the condenser hotwell to the RPV. Condensate is pumped by three main condensate (COND) pumps through the gland seal steam condenser, the steam jet air ejector condensers, and the offgas condenser. After leaving the offgas condenser, the condensate is pumped through a full-flow condensate filter-demineralizer system. The filter-demineralizer effluent is then pumped by three condensate booster pumps through the five low-pressure heaters. The last low-pressure heater discharges to the suction of the RFW pumps. The discharge from the two turbine-driven RFW pumps passes through the sixth stage of feedwater heating and then flows to the RPV. Feedwater flow is controlled by varying the speed of the steam-driven turbine.

1.2.2.6.9 Condensate Filter-Demineralizer System

The full-flow condensate filter-demineralizer system with instrumentation and semiautomatic controls is designed to ensure a constant supply of high-quality water to the reactor.

1.2.2.7 Electrical Systems, Instrumentation, and Control

1.2.2.7.1 Electrical Power Systems

The plant consists of a single main generator directly connected to a main power transformer through an isolated phase electrical bus duct. The main power transformer steps up the output of the 25-kV generator to a nominal 500-kV transmission system voltage.

The output of the main power transformer is connected to a 500-kV switchyard consisting of circuit breakers, disconnect switches, buses, and associated equipment arranged in a ring bus configuration.

A 230-kV offsite supply is provided to a separate startup auxiliary transformer to supply maximum startup, operating and shutdown load requirements for a normal plant auxiliary loads and for safety loads. In addition, a separate 115-kV offsite supply serves a backup auxiliary transformer with sufficient capacity to provide the power requirements of plant safe shutdown loads.

1.2.2.7.2 Electrical Power Systems Process Control and Instrumentation

Main generator electrical controls are located in the main control room. These include main generator circuit breaker controls, synchronizing equipment, and generator excitation and voltage control equipment. Instrumentation is also provided in the main control room for the main generator connections and equipment. This includes indicating instruments for voltage, current, kW, MVAR, and frequency. Recording instruments are provided for generator MW output and main bus voltage. Kilowatt-hour meters are provided for main generator outputs and for auxiliary power system loads. Instrumentation is provided for monitoring generator and transformer temperatures. Other types of monitoring instrumentation are provided as required to ensure proper operation of equipment. Circuit breaker controls, metering, and indication for the auxiliary power system are also located in the main control room.

High-speed protective relaying equipment is provided for the main generator, main and auxiliary transformers, main buses, transmission lines, and interconnecting cables and bus ducts to provide proper isolation of this equipment in the event of electrical faults. The protective relay system includes breaker failure protection and backup relaying to ensure proper isolation of electrical faults in the event of a failure of the primary protective relaying.

1.2.2.7.3 Nuclear System Process Control and Instrumentation

1.2.2.7.3.1 Reactor Manual Control System. The reactor manual control system (RMCS) provides the means by which control rods are positioned from the control room for power control. The system operates valves in each CRD hydraulic control unit to change control rod position. Only one control rod can be manipulated at a time. The RMCS includes the logic that restricts control rod movement (rod block) under certain conditions as a backup to procedural controls.

1.2.2.7.3.2 Recirculation Flow Control System. During normal power operation, a variable frequency power supply is used to control flow by varying the RRC pump motor speed. Adjusting the frequency changes motor speed and the coolant flow-rate through the core, thereby changing the core power level.

1.2.2.7.3.3 Neutron Monitoring System. The neutron monitoring system is a system of in-core neutron detectors and out-of-core electronic monitoring equipment. The system provides indication of neutron flux, which can be correlated to thermal power level for the entire range of flux conditions that can exist in the core. The source range monitors (SRM) and the intermediate range monitors (IRM) provide flux level indications during reactor startup and low power operation. The local power range monitors (LPRM) and average power range monitors (APRM) allow assessment of local and overall flux conditions during power range operation. The traversing in-core probe system (TIP) provides a means to calibrate the individual LPRM sensors. The neutron monitoring system provides inputs to the reactor manual control system to initiate rod blocks if preset flux limits are exceeded, and inputs to the RPS to initiate a scram if other limits are exceeded.

1.2.2.7.3.4 Refueling Interlocks. A system of interlocks that restricts movement of refueling equipment and control rods when the reactor is in the refueling and start-up modes is provided to prevent an inadvertent criticality during refueling operations. The interlocks back up procedural controls that have the same objective. The interlocks affect the refueling platform, refueling platform hoists, fuel grapple, and control rods.

1.2.2.7.3.5 Reactor Vessel Instrumentation. In addition to instrumentation for the nuclear safety systems and ESF, instrumentation is provided to monitor and transmit information that can be used to assess conditions existing inside the reactor vessel and the physical condition of the vessel itself. This instrumentation monitors reactor vessel pressure, water level, coolant temperature, reactor core differential pressure, coolant flow rates, and RPV head inner seal ring leakage.

1.2.2.7.3.6 Process Computer System. An on-line process computer is provided to monitor and log process variables and to make certain analytical computations. The rod worth minimizer function of the computer prevents rod withdrawal under low power conditions if the rod to be withdrawn is not in accordance with a preplanned pattern. The effect of the rod block is to limit the reactivity worth of the control rods by enforcing adherence to the preplanned rod pattern.

1.2.2.7.4 Power Conversion Systems Process Control and Instrumentation

1.2.2.7.4.1 Digital Electro-Hydraulic Control System. The DEH control system maintains control of the turbine governor valves and turbine bypass valves to allow proper generator and reactor response to system load demand changes while maintaining the nuclear system pressure essentially constant. When the generator is not connected to the grid, the DEH control system maintains turbine-generator speed (frequency) in response to reactor pressure changes by adjusting steam flow to the turbine valves and bypass valves.

The turbine generator speed/load controls can initiate rapid closure of the turbine control (governor) valves and rapid opening of the turbine bypass valves to prevent turbine overspeed on a generator electric load loss.

1.2.2.7.4.2 Feedwater System Control. A three-element controller is used to regulate the feedwater system so that proper water level is maintained in the reactor vessel. The controller uses main steam flow rate, reactor vessel water level, and feedwater flow rate signals. The feedwater control signal is used to control the speed of the steam turbine-driven feedwater pumps. During startup, shutdown, and low plant load conditions, the steam turbine-driven feedwater pumps are run at constant speed, and the feedwater control signal is used to modulate a startup feedwater control valve to maintain proper reactor water level.

1.2.2.8 Radioactive Waste Systems

1.2.2.8.1 Liquid Radwaste System

This system collects, treats, stores, and disposes of all radioactive liquid wastes. These wastes are accumulated directly in radwaste tanks or in sumps at various locations throughout the plant for subsequent transfer to collection tanks in the radwaste facility. Wastes are processed on a batch basis with each batch being processed by such method or methods appropriate for the quality and quantity of materials determined to be present. Processed liquid wastes may be returned to the condensate system or discharged to the circulating water blowdown line to the river. The liquid wastes in the discharge piping are diluted with circulating water blowdown to achieve a concentration at the site boundary which is below the limits of 10 CFR Part 20.

Equipment is selected, arranged, and shielded to permit operation, inspection, and maintenance with minimum personnel exposure. For example, tanks and processing equipment which contain significant radiation sources are located behind shielding, and sumps, pumps, instruments, and valves are located in controlled access rooms or spaces. Processing equipment is selected and designed to require a minimum of maintenance.

Protection against accidental discharge of liquid radioactive waste is provided by design redundancy, instrumentation for detection and alarm of abnormal conditions, and procedural controls.

1.2.2.8.2 Solid Radwaste System

Solid radioactive wastes are collected, processed, and packaged for storage and ultimate burial. These wastes are generally stored on the site until the short half-lived isotopes have decayed. Wet solid wastes are collected, dewatered, and solidified in steel containers. Examples of these wastes are filter residue, concentrated wastes, and spent resins. Dry solid wastes such as paper, air filters, rags, and used clothing are compressed and packaged in steel containers.

1.2.2.8.3 Gaseous Radwaste System

The purpose of the gaseous radwaste system is to process and control the release of gaseous radioactive wastes to the site environs so that the total radiation exposure to persons outside the controlled area does not exceed the limits of the applicable regulations, 10 CFR 20 and 10 CFR 50, Appendix I, even with some defective fuel rods.

The offgases from the main condenser are the major source of gaseous radioactive waste. The treatment of these gases includes volume reduction through a catalytic hydrogen-oxygen recombiner, water vapor removal through a condenser, decay of short-lived radioisotopes through a holdup line, further condensation, filtration, adsorption of isotopes on activated charcoal beds, further filtration through high efficiency filters, and final release.

Continuous radiation monitors are provided which indicate radioactive release from the reactor and from the charcoal absorbers. The radiation monitors are used to isolate the OG system on high radioactivity to prevent gas of unacceptably high activity from release.

Since clean gland seal steam is used, the offgases from the gland seal steam condenser are not treated prior to release.

The design of the OG system is such that the annual exposure to any offsite person during normal operation from gaseous sources will be ALARA and less than 10 CFR 20.

1.2.2.9 Radiation Monitoring and Control

1.2.2.9.1 Process Radiation Monitoring

Radiation monitors are provided on various lines to monitor either for radioactive materials released to the environs via process liquids and gases or for process system malfunctions. All effluents from the plant which are potentially radioactive are monitored. Several of the effluent monitoring systems record the results prior to discharge as noted on the following list of the major monitoring systems provided.

a. Main steam line radiation monitoring system,

b. Air ejector and offgas radiation monitoring systems (results recorded except for the charcoal bed vault),

c. Liquid radwaste effluent radiation monitoring system,

d. Plant service water and circulating water blowdown radiation monitoring systems,

e. Standby service water radiation monitoring system,

f. Reactor building ventilation exhaust plenum radiation monitoring system (results recorded),

g. Reactor building elevated release point radiation monitoring system (results recorded except for particulate/iodine sample),

h. Turbine building ventilation exhaust radiation monitoring system, (results recorded),

i. Radwaste building ventilation exhaust radiation monitoring system (results recorded), and

j. Reactor building closed cooling water monitoring system.

1.2.2.9.2 Area Radiation Monitors

Radiation monitoring devices are provided in key areas throughout the plant buildings to ensure that plant personnel will not be inadvertently exposed to high radiation doses.

1.2.2.9.3 Site Radiological Environmental Monitoring

A comprehensive radiation surveillance program was initiated in the spring of 1978 to measure radiation levels in the environs surrounding the plant. The program is designed to measure radiation exposure or radioisotope levels in eight different media.

Ambient radiation dose will be monitored using thermoluminescent dosimeters (TLDs). Airborne particulates are measured by filtering known quantities of air and analyzing the filtered material. Radioiodine in the air is measured in the same way except it is adsorbed onto a charcoal cartridge rather than being filtered.

Water is sampled at the plant intake, from the plant discharge, in the river below the plant, and at the nearest downstream municipal water supply. Groundwater in the vicinity is also sampled.

The radiation monitoring program includes sampling of garden produce where available in the vicinity of the site, the collection of river sediment samples from above and below the plant discharge point, the collection of fish samples from the Columbia River and the Snake River, and the collection of milk samples at four or more locations near the site.

The details of this monitoring program are given in Section 5.0 of the Offsite Dose Calculation Manual (ODCM).

1.2.2.9.4 Liquid Radwaste System Control

Liquid wastes to be discharged are handled on a batch basis with protection against accidental discharge provided by procedural controls. Instrumentation with alarms to detect abnormal concentration of the radwaste is provided, including automatic closure of discharge valves isolating the system from the environment.

1.2.2.9.5 Solid Radwaste System Control

The solid radwaste system collects, treats, and stores solid radioactive wastes for offsite shipment. Wastes are handled on a batch basis. Radiation levels of the various batches are monitored by the operator.

1.2.2.9.6 Gaseous Radwaste System Control

Gaseous radwastes are discharged through a reactor building elevated release point. Radiation levels of the release are continuously monitored and recorded. Isolation of the main condenser offgas is automatically initiated prior to release should the activity of the offgas exceed discharge limits.

1.2.2.10 Shielding

The shielding in the plant is designed to minimize exposure of plant personnel to radiation. The radiation levels during operation or shutdown conditions have been considered in determining the shielding requirements.

1.2.2.11 Fuel Handling and Storage Systems

1.2.2.11.1 New and Spent Fuel Storage

New and spent fuel storage racks are designed to prevent inadvertent criticality and load buckling. Sufficient coolant and shielding are maintained to prevent overheating and excessive personnel exposure, respectively. The design of the fuel pool provides for corrosion resistance, adherence to Seismic Category I requirements, and prevention of K_{eff} from exceeding 0.95 under dry or flooded conditions.

1.2.2.11.2 Fuel Handling System

The fuel handling equipment includes a fuel inspection stand, fuel preparation machine, a 125-ton crane, a refueling platform, a new fuel transfer basket, jib cranes, and other related tools for fuel and reactor servicing.

1.2.2.11.3 Fuel Pool Cooling and Cleanup System

The FPC system removes decay heat from stored spent fuel and maintains specified water temperature, purity, clarity, and level. This prevents fuel pool boiling and buildup of excessive radioactive materials in the cooling water, thereby minimizing possible exposures to plant personnel.

Cooling of spent fuel is accomplished by the Seismic Category I cooling system as described in Section 9.1.3. It can be isolated from the Seismic Category II cleanup portion of the system by automatic, redundant, Seismic Category I isolation valves which actuate on low fuel pool water level. If required, safety grade cooling and makeup water from the SW system is available to the system by remote-manual operation of redundant Seismic Category I valves to provide long-term cooling and prevent fuel pool boiloff which might result in unacceptable building environmental conditions.

1.2.2.12 Cooling Water and Auxiliary Systems

1.2.2.12.1 Reactor Building Closed Cooling Water System

The RCC system consists of pumps, heat exchangers, controls, and instrumentation to provide adequate cooling for the reactor auxiliary systems. The system is designed to provide a closed cooling water loop between nonessential systems which are potentially radioactive and the TSW system.

1.2.2.12.2 Plant Service Water System

Normal TSW is supplied from service water pumps located in the circulating water pump house. Two service water pumps are provided. The TSW system is designed to remove heat from various auxiliary equipment located within the plant.

1.2.2.12.3 Ultimate Heat Sink

Two spray ponds that serve as the UHS conservatively have a combined equivalent storage of 30 days, assuming no makeup and maximum evaporation and drift losses. Provisions are made to replenish the sink to allow continued cooling capability beyond the initial 30-day period.

1.2.2.12.4 Demineralized Water Makeup System

The DW makeup system is comprised of the trailer-mounted demineralizers and the DW system.

The DW system is designed to provide demineralized water to the CSTs for plant makeup and demineralized water for other plant operating requirements.

1.2.2.12.5 Potable Water and Sanitary Drain Systems

The plant potable water (PW) system provides water for drinking and sanitary purposes. Potable water is normally supplied from the tower makeup system (see Section 9.2.3).

The sanitary drain system effluent is directed to a central sanitary waste treatment facility which uses aerated lagoons in series with lined facultative stabilization ponds. The treatment plant, about 2500 ft SE of the CGS reactor, also receives waste from the WNP-1/4, the Plant Support Facility, and the DOE’s 400 Area.

1.2.2.12.6 Process Sampling Systems

The process sampling system provides process information that is required to monitor plant and equipment performance and changes to operating parameters. Representative liquid and gas samples are taken automatically and/or manually during normal plant operation for laboratory or on-line analyses.

1.2.2.12.7 Condensate Supply System

The condensate storage facility provides a source of water for testing and makeup during operation. Two 400,000 gal CSTs are interconnected to simultaneously supply condensate to the main condenser via one header, to the CRD pumps via a second header, and to the RHR, RCIC, and HPCS systems and condensate supply and condensate filter/demineralizer backwash pumps via a third header. The condensate supply pumps deliver condensate to miscellaneous services in the reactor and radwaste buildings.

Condensate is returned to the CSTs from the HPCS, RCIC, and radwaste systems, from CRD, condensate supply, and condensate filter/demineralizer backwash pump mini-flows, and from the main condensate system (equivalent to excess CRD injection water). Initial fill and makeup is from the DW system.

1.2.2.12.8 Equipment and Floor Drainage Systems

Plant equipment and floor drainage systems handle both radioactive and nonradioactive drains. Drainage systems which carry radioactive waste are isolated from drainage systems which do not carry radioactive waste.

All drains in the reactor building and radwaste building are considered radioactive. Turbine building drains are divided into radioactive and nonradioactive but all are directed to the radwaste system for processing. Floor and equipment drains in the diesel generator building and service building are routed to the storm water drainage system. The storm water drainage system is normally nonradioactive, however some accumulation of radioactive material (notably tritium) can occur.

1.2.2.12.9 Compressed Air Systems

The compressed air system consists of the control and service air system and the containment instrument air (CIA) system.

The control air system (CAS) is designed to supply clean, dry, oil-free air to station instrumentation and controls and to the accumulators of the MSIVs located outside the primary containment.

The service air (SA) system is designed to supply clean, oil-free air for station services, such as backwashing demineralizers and filters, hose connections for maintenance throughout the station and breathing air at selected locations.

The CIA system is designed to deliver nitrogen or clean, dry, oil-free air for MSIVs, SRV accumulators, and pneumatic operators located inside the primary containment.

1.2.2.12.10 Heating, Ventilating, and Air Conditioning Systems

The HVAC systems are designed to maintain proper air quality for personnel comfort and safety. In addition, the main control room, the critical switchgear area, the cable spreading room HVAC systems, the SW pump room heat removal systems, the reactor building emergency pump and critical electric equipment area cooling systems, and the ventilation system for the standby diesel generators are designed to operate under all station conditions. The primary containment drywell cooling and ventilation system is designed to operate during normal operation and under most upset conditions except a LOCA. All air distribution systems are designed so that airflow is directed from areas of lesser potential contamination to areas of progressively greater potential contamination.

Three separate and redundant HVAC systems service the main control room, cable spreading room, and critical switchgear areas. SW is used as the cooling medium for each system when the normal cooling water supply is unavailable.

Heating and ventilation for the standby diesel generator rooms is provided continuously for each diesel generator unit. Water cooled air handling units provide additional cooling when the diesel generators operate.

The turbine building is provided with a once-through ventilation system based on the use of evaporative coolers.

Ventilation for the radwaste building is provided by means of a once-through ventilation system with particulates filtered before release to the atmosphere.

The SW pump room heat removal systems consist of two independent and separate fan coil units.

The reactor building emergency pump and critical electric equipment area cooling system consists of 13 air handling units which operate to supply cool air to each of the critical equipment rooms when pumps are started and during abnormal conditions.

The primary containment drywell cooling and ventilation system consists of five fan coil units and nine recirculation fans. During normal operation, a minimum three out of five fan coil units are operating.

Ventilation for the reactor building is provided by a once-through ventilation system based on the use of evaporative coolers. The system incorporates the necessary isolation valves to ensure the necessary secondary containment integrity. A drywell and suppression chamber purge capability is provided as part of this system.

Other HVAC systems provide ventilation to the service building and other miscellaneous areas.

1.2.2.12.11 Fire Protection System

The FP system is designed to provide for the detection and extinguishing of fires.

Manual pull stations and automatic fire detectors are located appropriately throughout the plant and fire alarms are annunciated in the main control room.

The FP system provides a reliable water distribution system for extinguishing fires.
Two motor-driven fire pumps are used for normal service, with a diesel-engine-driven fire pump as a backup. A second diesel-driven fire pump with a separate water supply provides an additional backup. Motor-driven jockey pump is provided to maintain system pressure and to prevent cycling of the main fire pumps.

Automatic suppression systems provide protection to higher hazard areas of the plant including:

- Deluge systems protect the transformers and most other areas containing oil piping and oil storage equipment.
- A low-pressure carbon dioxide (CO_2) system is provided for the generator exciter housing.
- A total flooding Halon system is provided for the main control room power generation control complex (PGCC) subfloor.
- Wet pipe sprinklers protect the turbine/generator bearings and other miscellaneous areas.
- Preaction sprinkler systems protect diesel generators, day tank/transfer pump rooms, and areas with high concentrations of electrical cables.

Manual suppression includes:

- Fire hydrants spaced around the yard fire main loop.
- Fire hose stations located throughout the plant.

- Portable fire extinguishers of appropriate types are strategically and conspicuously placed throughout the plant.

1.2.2.12.12 Communications Systems

The plant communication systems are designed to provide reliable communication inside and outside the plant and from the plant to local fire protection and law enforcement authorities. The system utilizes a public address and building wide alarm system, a public telephone system, a private digital telephone system, a sound powered telephone system, a radio communication system, and an automatic transmission telephone link to the Dittmer Control Center of the Bonneville Power Administration (BPA).

1.2.2.12.13 Lighting Systems

The plant lighting systems are normal ac lighting, normal-emergency ac lighting, dc lighting, and battery-pack emergency lighting. Lighting intensities are designed to provide indoor and outdoor illumination consistent with the July 1974 Illumination Engineering Society recommendations, and meet or exceed Occupational Safety and Health Act (OSHA) requirements.

1.2.2.12.14 Normal Auxiliary Alternating Current Power System

The plant normal auxiliary ac power system consists of two normal auxiliary transformers, the 4.16-kV and 6.9-kV normal auxiliary (non-Class 1E) distribution system, the 480-V auxiliary power distribution system and the 120/208-V non-Class 1E distribution system.

The normal ac auxiliary transformers provide power to all plant auxiliaries and comprise the normal plant ac power source when the main generator is operating. One of the normal auxiliary transformers is a dual secondary type with both secondary windings stepping down the generator voltage to 4.16 kV for supply to 4.16-kV non-Class 1E switchgear buses. The other normal auxiliary transformer steps down the generator voltage to 6.9 kV for supply of 6.9-kV non-Class 1E switchgear buses.

The plant 480-V ac auxiliary power system distributes ac power necessary for normal auxiliary and ESF 480-V plant loads. All non-ESF elements of this distribution system are capable of being supplied from the normal auxiliary power source or from the startup power source via the 4.16 kV-non-Class 1E switchgear. The ESF portions of the 480-V distribution system are supplied via the 4.16-kV Class 1E switchgear, and therefore are capable of being supplied by either the normal, startup, backup, or standby sources.

The 120/208-V non-Class 1E ac power system provides power for non-ESF loads.

1.2.2.12.15 Diesel Generator Fuel-Oil Storage and Transfer System

The diesel fuel oil storage and transfer system consists of separate, independent diesel oil supply subsystems serving each of two emergency diesel generators and the HPCS diesel generator. Each full capacity subsystem consists of a fuel oil storage tank, a transfer pump, a day tank, interconnecting piping, strainers and valves, and associated instrumentation and controls.

1.2.2.12.16 Auxiliary Steam System

The auxiliary steam (AS) system normally operates only when the heating steam evaporators are inoperative during plant shutdown. The system then supplies steam to HVAC systems for air and water space heating and for humidification and also to the radwaste system. The system consists of fuel oil storage tank and transfer pumps, auxiliary boiler, blowdown tank, chemical feed tank and metering pump, deaerator and boiler feed pumps, condensate return tank pumps, steam supply and condensate return piping and valves, and associated instruments and controls.

1.2.3 COMPLIANCE WITH NRC REGULATORY GUIDES

The CGS conformance to the NRC regulatory guides is documented in Section 1.8 and in appropriate sections of this FSAR.

Table 1.2-1

Principal Regulations and Codes Followed in Plant Design

Number	Title
10 CFR series	Code of Federal Regulations, principally:
10 CFR 20	Standards for Protection Against Radiation
10 CFR 50	Licensing of Production and Utilization Facilities
10 CFR 50, Appendix A	General Design Criteria for Nuclear Power Plant Construction Permits
10 CFR 50, Appendix B	Quality Assurance Criteria
10 CFR 50, Appendix I	Numerical Guides for Design Objectives and Limiting Conditions for Operation to Meet the Criterion "As Low As Is Reasonably Achievable"
10 CFR 100	Reactor Site Criteria
IEEE-279	IEEE Criteria for Nuclear Power Generating Station Protection Systems
IEEE-308	IEEE Criteria for Class IE Electrical Systems for Nuclear Power Generating Stations
ASME B&PV	ASME Boiler and Pressure Vessel Code:
Section III	Nuclear Components
Section VIII	Pressure Vessels
Section XI	Inservice Inspection
AEC Press Release IN-817	Tentative Regulatory Supplementary Criteria for ASME Code-Constructed Pressure Vessels
ANSI-B31.1.0	ANSI Standard Code for Pressure Piping, Power Piping

NOTE: Additional codes and regulations applying to specific areas of system design are referenced in discussions of individual systems.

Table 1.2-2

Plant Shielding and Zone Classification

Zone	Description	Design Dose Rate (mrem/hr)
I	Uncontrolled, unlimited access	≤1.0
II	Controlled, limited access	≤2.5
III	Controlled, occupancy for short periods, normally inaccessible	≤100
IV	For very short periods. Secured and controlled entrance.	>100

NOTES:

1. Radiation Zone I areas can be occupied by plant personnel or visitors for unlimited periods.
2. Radiation Zone II areas are areas where whole body dose is not expected to exceed 1.25 rem per calendar quarter.
3. Areas having dose rates in excess of 100 mrem/hr are posted as high radiation areas and access is secured and controlled.
4. Radiation Zone III and IV areas can be entered only after the radiation level is determined and the working time limit is established.
5. Accessible areas have dose rates of less than 100 mrem/hr.
6. Access to all controlled areas is through controlled check points.
7. Controlled and limited access areas are identified by warning signs.

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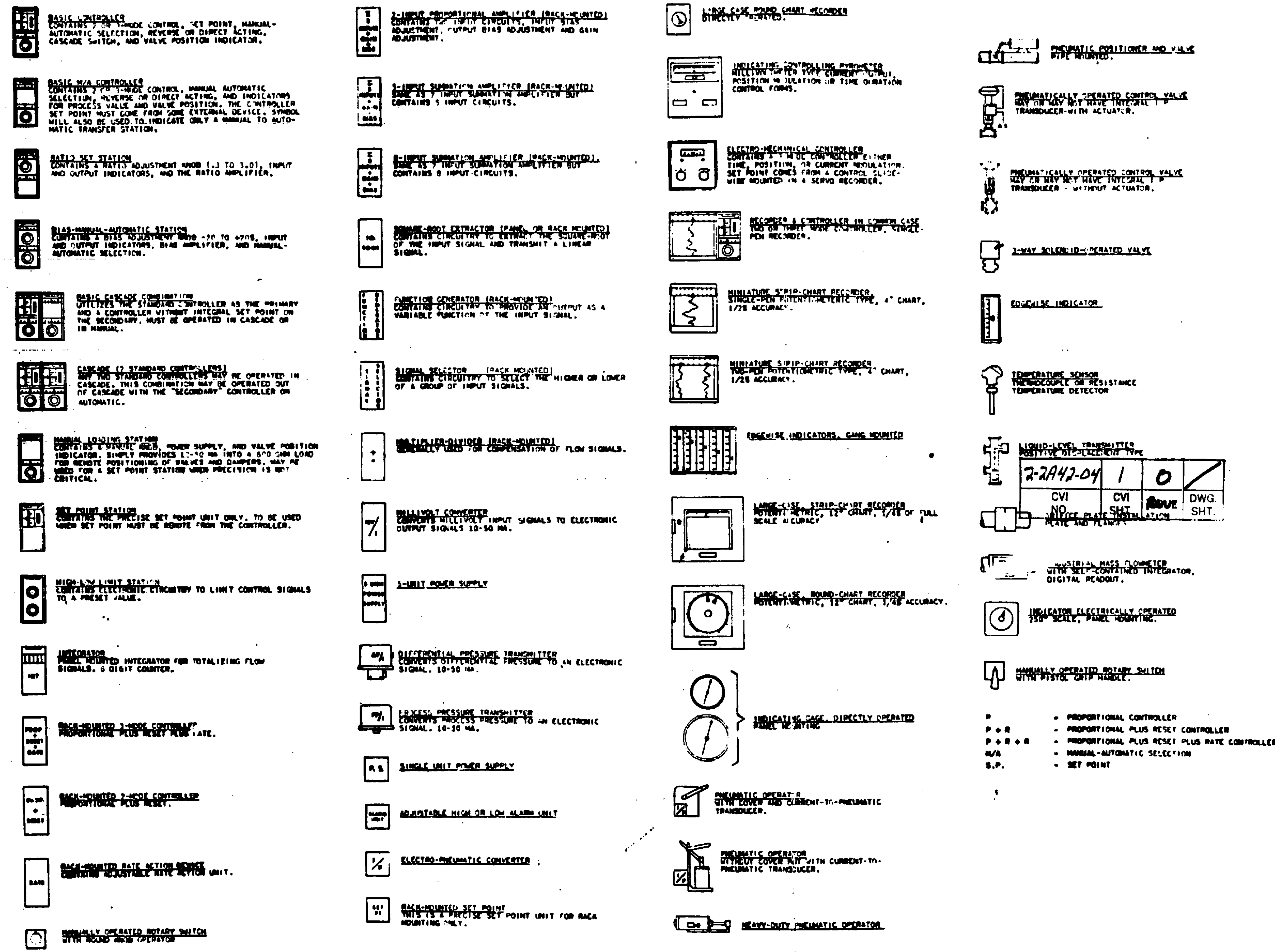
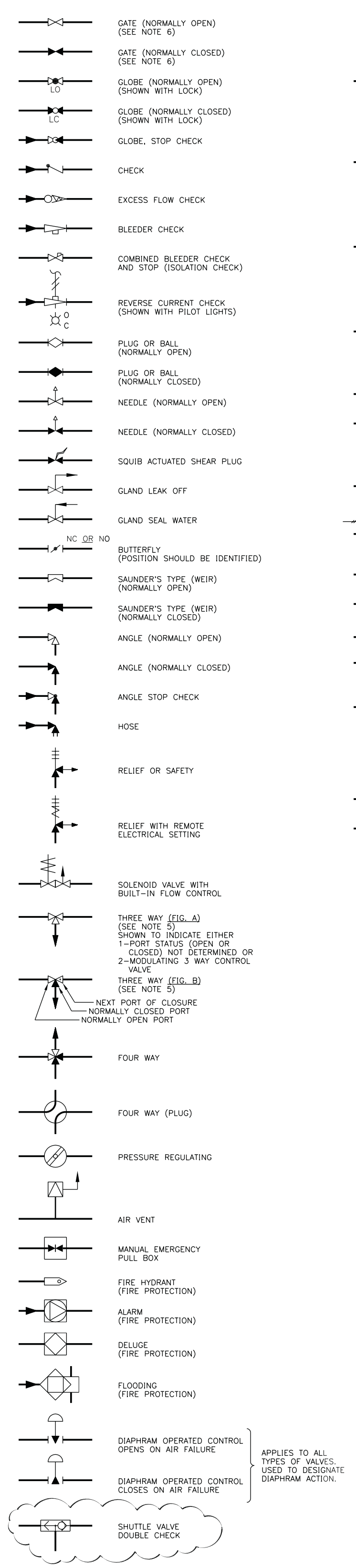
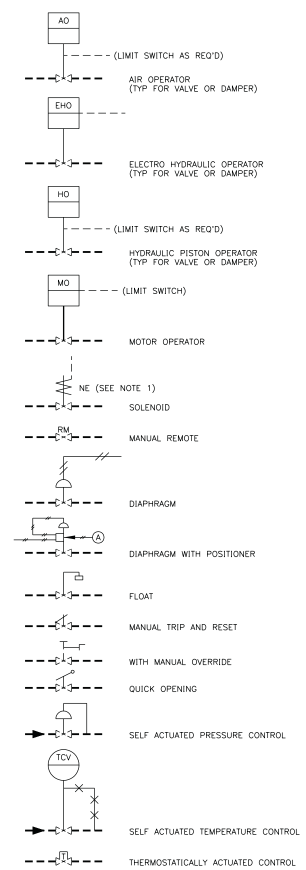


Columbia Generating Station Final Safety Analysis Report	General Electric Piping and Instrumentation Drawing Symbols		
	Draw. No. 02A42-04,1	Rev. 0	Figure 1.2-25

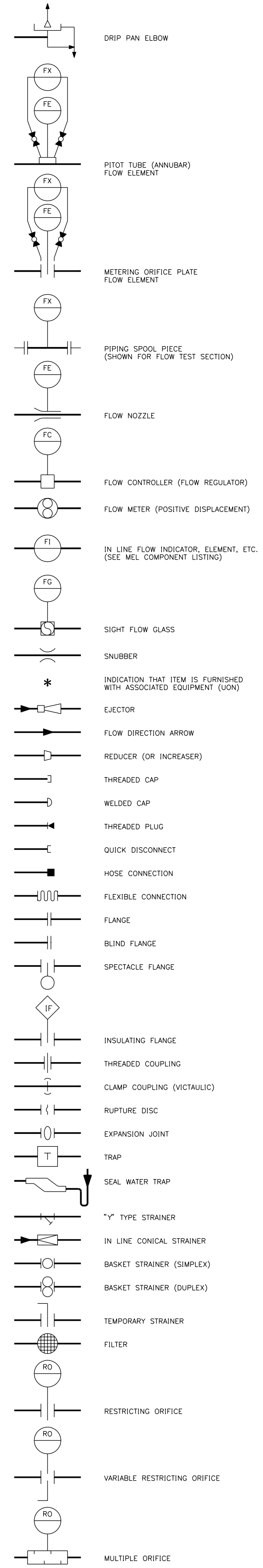
VALVES



OPERATORS



MISC. SYMBOLS



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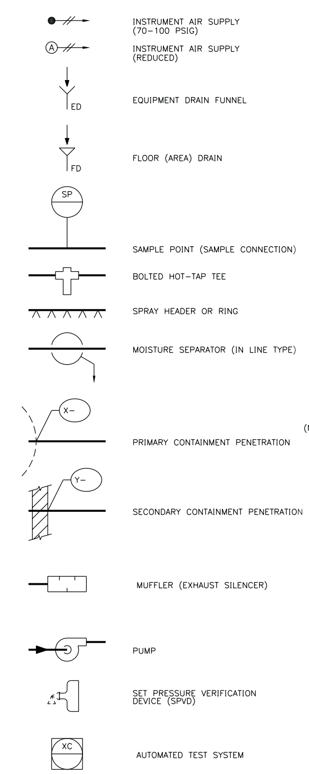
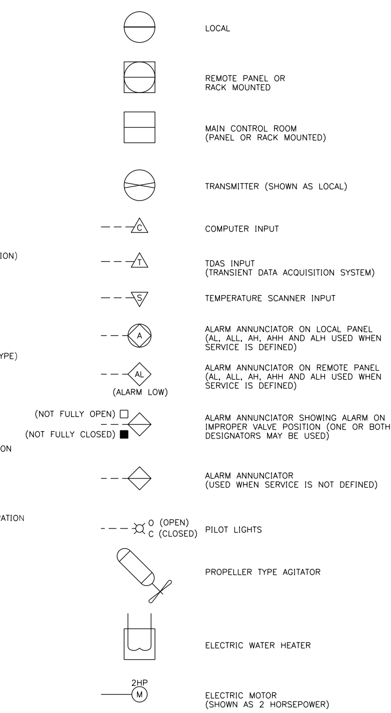
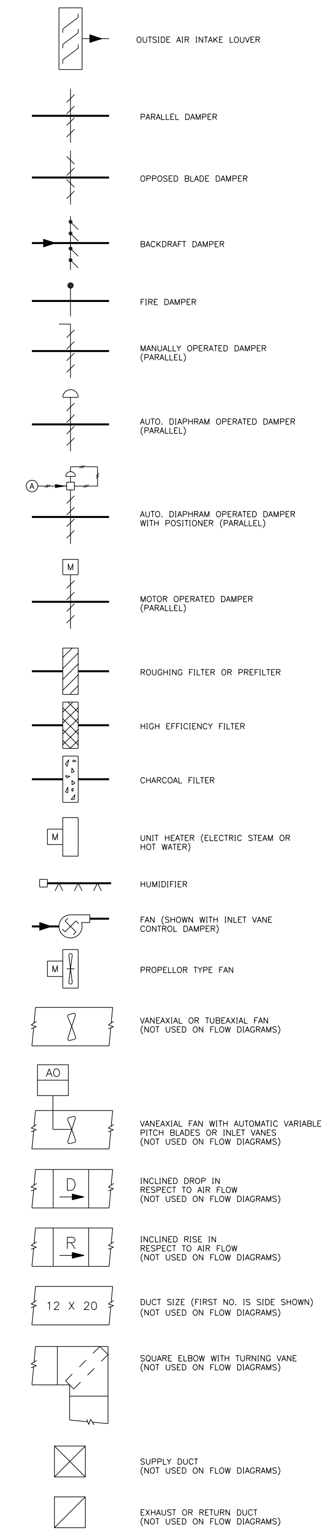


ELECTRICAL SYMBOLS

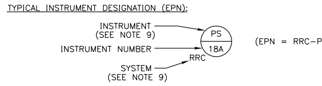


HVAC SYMBOLS

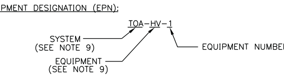


NOTES

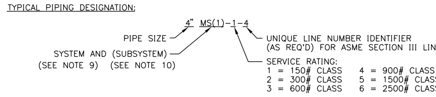
1. ALL SOLENOID VALVES SHOWN IN DE-ENERGIZED POSITION. "NE" DENOTES SOLENOID NORMALLY ENERGIZED DURING PLANT OPERATION, "ND" DENOTES NORMALLY DE-ENERGIZED.
2. TYPICAL INSTRUMENT DESIGNATION (EPN):



3. TYPICAL EQUIPMENT DESIGNATION (EPN):



4. TYPICAL PIPING DESIGNATION:



5. WHERE THREE WAY VALVE OPERATIONS ARE NOT FINAL, "FIG. A" IS USED (SEE VALVE SYMBOLS). WHERE VALVE OPERATION IS FINAL, "FIG. B" IS USED.
6. "NORMALLY OPEN" SYMBOL (UNSHADED) AND "NORMALLY CLOSED" SYMBOL (SHADED) VALVE POSITIONS OR ORIENTATIONS ARE DRAWN TO DEPICT THE SYSTEMS IN NORMAL POWER OPERATION OF THE PLANT, WITH STANDBY, SUPPORTING, OR EMERGENCY SYSTEMS SHOWN IN STANDBY CONDITION. TO DETERMINE ACTUAL VALVE POSITIONS THE USER MUST CONSULT THE SYSTEM OPERATION PROCEDURES AND THE OPERATIONS CREW.
7. FOR OPERATING CONDITIONS, DESIGN CONDITIONS AND COMPLETE PIPING DESIGNATION SEE THE WNP-2 SPECIFICATION TABLES 1 & 2; SECTION 15B.1 FOR B31.1 PIPING; SECTION 1 FOR ASME SECTION III PIPING AND SECTION 2 FOR ASME SECTION III TUBING.
8. CODE GROUP DESIGNATION APPLIES ONLY THROUGH PRIMARY ROOT VALVES UNLESS SPECIFICALLY NOTED OTHERWISE. M619 DRAWING SERIES GIVES CODE GROUP FOR INSTRUMENTATION.
9. SYSTEM AND COMPONENT ABBREVIATIONS CAN BE FOUND IN THE MASTER EQUIPMENT LIST (MEL) OR IN SECTION 1.2 OF THE FSAR WHICH IS UPDATED ANNUALLY.
10. THE SUBSYSTEM (NOTE 4) DESIGNATES A PROCESS FUNCTION.

MISC. ABBREVIATIONS

A/S	AIR SUPPLY
E/S	ELECTRONIC POWER SUPPLY
FAI	FAIL AS IS
FC	FAIL CLOSED (CLOSES ON AIR OR ELECTRICAL FAILURE)
FO	FAIL OPEN (OPENS ON AIR OR ELECTRICAL FAILURE)
LC	LOCKED CLOSED
LC+	HANDWHEEL LOCKED AND ELECTRICAL BREAKER LOCKED OPEN
LN	MANUAL HANDWHEEL LOCKED IN NEUTRAL POSITION
LO	LOCKED OPEN
LT	LOCKED THROTTLED
NC	NORMALLY CLOSED
ND	NORMALLY DE-ENERGIZED
NE	NORMALLY ENERGIZED
NO	NORMALLY OPEN
TH	THROTTLED

MISC. EQUIPMENT LEGEND

ACV	AIR CONTROL VALVE
GLA	GAS AND LIQUID ANALYZER
H/A	HAND/AUTO STATION
HS	HAND SWITCH
HSE	HOSE
LE	LABORATORY EQUIPMENT
MA	MANIFOLD
MS	MANUAL SWITCH
S/S	SELECTOR SWITCH
TFC	TANK FILL CONNECTION
VPI-LVDT	VALVE POSITION INDICATION- LINEAR VARIABLE DIFFERENTIAL TRANSFORMER

HVAC EQUIPMENT LEGEND

ATD	AIR TURNING DEVICE
BD	BACKDRAFT DAMPER
BG	BOTTOM GRILLE (VERTICAL BELOW 6'-0")
BR	BOTTOM REGISTER (VERTICAL BELOW 6'-0")
CD	CEILING DIFFUSER (HORIZONTAL WITH OR WITHOUT CEILING)
CG	CEILING GRILLE (HORIZONTAL WITH OR WITHOUT CEILING)
CR	CEILING REGISTER (HORIZONTAL WITH OR WITHOUT CEILING)
DC	DUST COLLECTOR
FC	FAN COIL UNIT
FRP	FIBERGLASS REINFORCED POLYESTER DUCT
PF	PORTABLE FILTER
PRD	PRESSURE REDUCING DAMPER
RHC	REHEAT COIL
SD	SPLITTER DAMPER
TG	TOP GRILLE (VERTICAL ABOVE 6'-0")
TR	TOP REGISTER (VERTICAL ABOVE 6'-0")
VD	VOLUME DAMPER
WHA	WATER HAMMER ARRESTOR

PIPING AND SYSTEM CODE GROUPS

CODE GROUP	APPLICABLE CODE
A	ASME BOILER AND PRESSURE VESSEL CODE SECTION III CLASS 1.
B	ASME BOILER AND PRESSURE VESSEL CODE SECTION III CLASS 2.
C	ASME BOILER AND PRESSURE VESSEL CODE SECTION III CLASS 3.
D	ANSI B31.1 CODE FOR PRESSURE PIPING.
E	NSSS ENGINEERED INSTRUMENTS.

QUALITY ASSURANCE CLASSES

REQUIREMENTS FOR QUALITY ASSURANCE CLASSES ARE SPECIFIED IN THE WNP-2 SPECIFICATIONS.

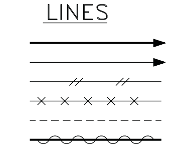
SEISMIC REQUIRMENTS

REQUIREMENTS FOR SEISMIC CLASSES ARE SPECIFIED IN THE WNP-2 SPECIFICATIONS.

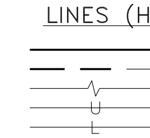
BUILDING LOCATION LEGEND

A	AUXILIARY BUILDINGS
C	PART OF OR WITHIN PRIMARY CONTAINMENT
D	DIESEL GENERATOR BUILDING
L	OFFSITE LOCALE
M	MAKEUP WATER PUMP HOUSE
O	OUTDOORS ON SITE
P	PUMP HOUSE
R	REACTOR BUILDING
S	SERVICE BUILDING
T	TURBINE GENERATOR BUILDING
W	RADWASTE/CONTROL BUILDING

LINES



LINES (HVAC)



Columbia Generating Station
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Flow Diagram Legend, Symbols and Abbreviations

Draw. No. M501 | Rev. 51 | Figure 01.2-26

System Acronyms

AAP	Alternate Access Point Bldg. and Appurtenances
AEA	Tech. Support Cntr. Exhaust Air
AMA	Tech. Support Cntr. Mixed Air
ANN	Annunciators
AOA	Tech. Support Cntr. Outside Air
APRM	Average Power Range Monitors
APWH	Tech. Support Cntr. Potable Hot Water
AR	Air Removal
ARA	Tech. Support Cntr. Return Air
ARE	Tech. Support Cntr. Refrig. Equipment
ARI	Alternate Rod Insertion
ARM	Area Radiation Monitoring
AS	Auxiliary Steam
BA	Backwash Air
BAS	Breathing Air Supply
BCF	Boiler Chemical Feed
BD	Cond. Blowdown or Rad. Boards
BS	Bleed (Extraction) Steam
CAC	Containment Atmosphere Control (Deactivated)
CAS	Control Air System
CBD	Circ. Water Blowdown
CCH	Control Room Chilled Water
CEP	Containment Exhaust Purge
CF	Chemical Feed
CHEM	Chemistry Equipment
CIA	Containment Instrument Air
CJW	Cooling Jacket Water
CL	Chlorine
CMS	Containment Monitoring System
CN	Containment Nitrogen
CND	Condenser Drains & Vents
CO	Condensate (Auxiliary)
CO_2	Carbon Dioxide
COMM	Communications
COND	Condensate (Nuclear)
CP	Cathodic Protection
CPR	Condensate Demineralizer
CRA	Containment Recirculating Air
CRD	Control Rod Drive
CSP	Containment Supply Purge
CTMA	Cooling Tower Electrical Bldg. Mixed Air
CVB	Containment Vacuum Breakers
CW	Circulating Water
DCN	CRD Decontamination
DCW	Diesel Cooling Water
DE	Diesel Exhaust (Engine)
DEA	Diesel Building Exhaust Air
DEH	Digital Electro-hydraulic Control
DG	Diesel Generator
DLO	Diesel Lube Oil
DMA	Diesel Building Mixed Air
DO	Diesel Oil
DOA	Diesel Building Outside Air
DRA	Diesel Building Return Air
DSA	Diesel (Engine) Starting Air
DW	Demineralized Water
EDR	Equipment Drains Radioactive
ELEC	Electrical Maintenance Equipment
EOF	Emergency Offsite Facility
ERM	Environmental Rad. Monitoring
ES	Exhaust Steam (Turbine)
FAC	Facilities Generic Equipment
FD	Floor Drain
FDR	Floor Drain Radioactive
FO	Fuel Oil
FP	Fire Protection
FPC	Fuel Pool Cooling
FW	Filtered Water
GEA	Guard House Exhaust Air
GFP	Guard House Fire Protection
GH	Main Guard House
GMA	Guard House Mixed Air
GOA	Guard House Outside Air
GPWH	Guard House Water Hot Potable
GRA	Guard House Return Air
GY	Glycol
H_2	Hydrogen (Turbine Generator)
HCO	Heating Steam Condensate
HD	Heater Drain
HHW	Heating Hot Water
HP	Health Physics
HPCS	High Pressure Core Spray
HS	Heating Steam
HSSF	Hydrogen Storage and Supply Facility
HSV	Heating Steam Vent
HT	Heat Tracing
HV	Heater Vent
HWC	Hydrogen Water Chemistry
HY	RRC Hydraulic Control
IBD	ISO Phase Bus Duct Cooling
IR	Instrument Rack
IRM	Intermediate Range Monitor
IRON	Chemical Feed
LD	Leak Detection
LE	Laboratory Equipment (Permanent Plant)
LF	Laundry Facility
LPCS	Low Pressure Core Spray
LPDS	Loose Parts Detection System
LPRM	Local Power Range Monitor
MD	Miscellaneous Drain
MECH	Mechanical Maintenance Equipment
MEL	Master Equipment List
MET	Meteorological
MLF	Mobile Laundry Facility
MS	Main Steam (Nuclear)
MSH	Machine Shop Equipment
MSLC	Main Steam Leakage Control (Deactivated)
MT	Material Transport
MW	Miscellaneous Waste
MWR	Miscellaneous Waste Radioactive
NSSE	Nuclear System Servicing Equipment
NSSS	Nuclear Steam Supply System
OFEA	Offsite Facility Exhaust Air
OFMA	Offsite Facility Mixed Air
OFOA	Offsite Facility Outside Air
OFRA	Offsite Facility Recirculation Air
OG	Off Gas
OL	Obstruction Lighting
PDIS	Plant Data Information System
PEA	Pumphouse Exhaust Air
PI	Process Instrumentation
PL	Plant Equipment
PMA	Pumphouse Mixed Air
POA	Pumphouse Outside Air
PPC	Plant Process Computer
PRA	Pumphouse Return Air
PRM	Process Radiation Monitoring
PS	Process Sampling
PSD	Plant Sanitary Drain
PSR	Process Sampling Radioactive
PVMS	Plant Vibration Monitoring System
PVR	Process Vents Radioactive
PWC	Potable Cold Water
PWH	Potable Hot Water
PWR	Process Waste Radioactive
RBM	Rod Block Monitor
RCC	Reactor Closed Cooling Water
RCIC	Reactor Core Isolation Cooling
RD	Roof Drain
REA	Reactor Building Exhaust Air
RFT	Reactor Feedwater Turbine
RFW	Reactor Feedwater
RHR	Residual Heat Removal
ROA	Reactor Building Outside Air
RPIS	Rod Position Indicator System
RPS	Reactor Protection System
RPWH	Reactor Building Potable Hot Water
RRA	Reactor Building Return Air
RRC	Reactor Recirculation
RSE	Reactor Service Equipment
RWCU	Reactor Water Cleanup
RWM	Rod Worth Minimizer
S	Sampling
SA	Service Air
SAT	Sulfuric Acid Treatment
SCH	Service Building Chilled Water
SCI	Supervisory Control
SCW	Stator Cooling Water
SEA	Service Building Exhaust Air
SEC	Plant Security
SEIS	Seismic Monitoring System
SFS	Spent Fuel Storage
SGT	Standby Gas Treatment
SHCO	Service Building Heating Condensate
SHHW	Service Building Heating Hot Water
SLC	Standby Liquid Control
SM	Salinity Monitoring
SMA	Service Building Mixed Air
SO	Seal Oil
SPTM	Suppression Pool Temp Monitoring
SPWH	Service Building Potable Hot Water
SRA	Service Building Return Air
SRM	Source Range Monitoring
SS	Sealing Steam
SW	Standby Service Water
SWA	Solid Waste
SWCF	Service Water Chemical Feed
TDAS	Transient Data Acquisition System
TEA	Turbine Building Exhaust Air
TG	Turbine Generator
TIP	Traversing Incore Probe
TMU	Tower Makeup Water
TO	Turbine (Lube) Oil
TOA	Turbine Building Outside Air
TPWH	Turbine Bldg. Potable Hot Water
TRA	Turbine Building Return Air
TSC	Technical Support Center
TSW	Plant Service Water
VES	Vessel (Sect. 8, Non Power Block)
VRMA	Variable Speed Drive Bldg. Mixed Air
WCH	Radwaste Building Chilled Water
WEA	Radwaste Building Exhaust Air
WHCO	Radwaste Heating Condensate
WMA	Radwaste Building Mixed Air
WNP2	Washington Nuclear Plant 2 (Columbia Generating Station)
WOA	Radwaste Building Outside Air
WPWH	Radwaste Bldg. Potable Hot Water
WRA	Radwaste Building Return Air
WRE	Radwaste Building Refrigeration
WRM	Wide Range Monitoring
ZINC	Chemical Feed System

Equipment Acronyms

AA	Audio Alarm
AC	Air Conditioning Unit
ACC	Accumulator
ACM	Acoustic Monitor/Sensor
AD	Air Damper
AH	Air Handling Unit
AI	Air Indicator
ALM	Alarm Annunciator-Do Not Use
ALT	Alternating Relay
AM	Ammeter
AMP	Amplifier
ANN	Annunciator
AO	Air Operator
AR	Air Receiver
AR/FR	Analyzer and Flow Recorder
ASM	Assembly
ASW	Air Switch (4-way Valve)
AT	Air Transmitter
ATD	Amp Transducer
ATS	Automatic Transfer Switch
AUD	Audio Monitor
AUX	Auxiliary Unit
AV	Air Valve
AW	Air Washer
AY	Analyzer
B0	24 Volt Battery
B1	125 Volt Battery
B2	250 Volt Battery
B3	12 Volt Battery
B4	48 Volt Battery
BDET	Badge (Keycard) Detector
BELL	Bell (Fire Protection)
BFI	Blown Fuse Indicator
BL	Baler
BLDG	Bldg (For PSD System Only)
BLR	Boiler
BT	Bolted Tee (For SA System)
BU	Emerg Lighting Battery Unit
BUOY	Buoy
C	Compressor
C0	24 Volt Battery Charger
C1	125 Volt Battery Charger
C2	250 Volt Battery Charger
C3	12 Volt Battery Charger
CAB	Cabinet
CAP	Capacitor
CB	Circuit Breaker
CC	Cooling Coil
CCTV	Closed Circuit Television
CCU	Central Control Unit
CE	Conductivity Element
CERA	Cond Element Retractor Assembly
CF	Charcoal Filter
CFG	Centrifuge
CH	Channel
CHL	Chlorinators
CHM	Chamber
CHR	Chiller
CHS	Chassis
CI	Conductivity Indicator
CIC	Conductivity Ind Controller
CIS	Conductivity Ind Switch
CIT	Conductivity Ind Transmitter
CITS	Conductivity Ind Transmitter Switch
CJW	Cooling Jacket Water
CM	Communications Monitor
CNTR	Contractor
COE	Corrosivity Element
COIC	Corrosivity Indic Cont
COMP	Computer
CONN	Connector
COR	Corrosivity Recorder
COS	Carbon Monoxide Sensor
COT	Corrosivity Transmitter
CP	Control Panel
CPL	Data Coupler
CPTR	Compactor
CPU	Central Processing Unit
CR	Conductivity Recorder; Control Room Chiller
CRA	Crane
CRB	Control Rod Blade
CRM	Control Module
CRS	Conductivity Recorder Switch
CRT	Terminal Display Screen
CS	Conductivity Switch
CSK	Shield Transfer Cask
CT	Current Transformer/Cooling Tower
CU	Condensing Unit
D	Damper (Backdraft Or Motor)
DC	Decoder
DCM	Dry Cleaning Machine
DCN	CRD Decontamination System
DDR	Disk Drive Recorder
DE	Density Element
DET	Detector
DFS	Differential Flow Switch
DG	Digital Display Generator
DH	Drywell Head
DIF	Diffuser
DIO	Diode, Control Rectifier
DISC	Disconnect Switch
DLR	Differential Level Recorder
DLS	Differential Level Switch
DLT	Differential Level Transmitter
DM	Demineralizer
DMM	Display Memory Module
DMS	Demister
DMTR	Demand Meter
DOE	Dissolved Oxygen Element
DOIT	Dissolved Oxygen Indic Trans
DOOR	Door
DOR	Dissolved Oxygen Recorder
DP	Distribution Panel
DPC	Diff Press Controller
DPE	Drip Pan Elbow
DPI	Diff Press Ind
DPIC	Diff Press Ind Controller
DPIR	Diff Press Ind Recorder
DPIS	Diff Press Ind Switch
DPIT	Diff Press Ind Transmitter
DPR	Diff Press Recorder
DPS	Diff Press Switch
DPT	Diff Press Transmitter
DR	Demand Recorder
DRVE	Drive Mechanism For CRD
DS	Density Switch
DT	Dens Trans Or Drive Turbine
DTIS	Diff Temp Indicating Switch
DTRS	Diff Temp Recording Switch
DTS	Diff Temp Switch
DTT	Diff Temp Transmitter
DU	Deaerator
DV	Deluge Valve
DVSP	Dump Valve Solenoid Pilot
DVSPV	Dump Valve Solenoid Pilot Valve
DWS	Demineralized Water Shower
DY	Dryer
E/I	Volt To Current Converter
E/P	Electro Pneumatic Converter
E/S	Electronic Power Supply
EAMP	Preamplifier
EC	Electronic Controller
ECG	Electrochemical Generator
ED	Eductor
EF	Electronic Filter
EFC	Excess Flow Check Valve
EHC	Electric Heating Coil
EHO	Electrohydraulic Operator
EI	Power Supply Monitor
EIS	Power Supply Monitor Switch
EJ	Expansion Joint
EJC	Ejector
ELEV	Elevator
ELP	Emergency Lighting Panel
EMSQ	Mean Square Voltage Device
ENG	Engine
EPA	Electrical Protection Assem
EPP	Emergency Power Panel
EQ	Speciality Equip and Tools
ERB	Emerg Rmt Ballast (Lighting)
ES	Exhaust Silencer
ESH	Electric Strip Heater
EUH	Electric Unit Heater
EV	Evaporator
EX	Exhauster
EXC	Exciter
F	Filter
F/U	Flow Unit
FA	Flame Arrestor
FC	Flow Controller
FCN	Fuel Oil Tk Fill Connector
FCV	Flow Control Valve
FD	Fire Damper
FDg	Freon Degreaser
FE	Flow Element
FG	Flow Glass
FGEN	Function Generator
FH	Fume Hood
FHB	Fuel Handling Box
FI	Flow Indicator
FIC	Flow Indicating Controller
FICS	Flow Indicating Controller Switch
FIS	Flow Indicating Switch
FIT	Flow Indicating Transmitter
FL	Filter
FLP	Fillport Assem
FLT	Filter
FIX	Flexible Connection
FN	Fan
FO	Freon Actuated Operator
FP	Filter Polisher
FQ	Flow Integrator
FQI	Flow Integrating Indicator
FQS	Flow Integrating Switch
FR	Flow Recorder
FR/DL	Flow and Diff. Level Recorder
FRC	Flow Recording Controller
FRDLR	Flow and Diff Level Recorder
FRS	Flow Recording Switch
FS	Flow Switch
FSPV	Flow Solenoid Pilot Valve
FT	Flow Transmitter
FTD	Frequency Transducer
FU	Filter Unit
FUSE	Fuse
FX	Flow Test Connection
FY	Flow Sig. Cond.
GATE	Gate
GCAL	AGS Calibrator
GEN	Generator
GOV	Governor
GVT	Gravity Ventilator
H	Heater
H_2E	Hydrogen Element
H_2I	Hydrogen Indicator
H_2IS	H_2 Indicating Switch/Monitor
H_2IT	Hydrogen Ind Transmitter
H_2R	Hydrogen Recorder
H_2T	Hydrogen Transmitter
HAS	High Amplitude Selector
HC	Heating Coil
HCU	Hydraulic Control Unit
HF	HEPA Filter
HM	Hour Meter
HO	Hydraulic Operator
HOI	Hoist
HP	Valve Act. Hyd. Power Unit
HPU	Hydraulic Power Unit
HR	Hydrogen Recombiner
HS	Hose Station
HSS	High Selector Switch
HT	Hydrant
HTC	Heat Trace Cable
HTP	Heat Trace Panel
HU	Humidifier
HUM	Humidifier (Obsolete. Use HU)
HV	Heating and Ventilation Unit
HVRB	High Voltage Rubber Blanket
HX	Heat Exchanger
HZM	Hertz Meter
I/P	Current Pneumatic Converter
ID	Ionization Detector
IL	Indicating Light
IMD	Inductive Motor Drive
IN	Inverter
IND	Inductor
INDX	Indexer
IOS	Current Operated Switch
IR	Instrument Rack
IS	Intake Silencer
ISOL	Isolator, Isolation Device
ITD	Current Transducer
IX	Ion Exchanger
JB	Junction Box
JP	Jet Pump
KBD	Computer Keyboard (Security)
L	Lubricator
LA	Lightning Arrestor
LAG	Dynamic Compensator
LAS	Low Amplitude Selector
LC	Level Controller
LCRM	Log Count Rate Meter
LCV	Level Control Valve
LE	Level Element
LF	Lighting Fixture
LG	Level Glass
LI	Level Indicator
LIC	Level Indicating Controller
LIS	Level Indicating Switch
LITS	Level Indic Trans Switch
LMS	Limit Switch
LMTR	V/I Signal Limiter
LNR	Linear Reactor
LOC	Lube Oil Conditioner
LP	Lighting Panel
LPW	24 Volt Lambda Power Supply
LR	Level Recorder
LR/PR	Level/Pressure Recorder
LRS	Level Recording Switch
LS	Level Switch
LSC	Lightning Strike Counter
LSPV	Sol. Pilot Valve TMU-level
LSS	Low Selector Switch
LT	Level Transmitter
LTD	Level Transmitter Detector
LVDT	Linear Var. Dif. Transformer
LVS	Low Volume Selector
LWR	Unknown Equipment Type ?
LWS	Low Differential Pressure
M	Motor
M/A	Manual/Auto Station
MA	Manifold
MACH	Machine
MBS	Maint. Bypass Switchgear
MC	Moisture Controller
MDET	Metal Detector
MDS	Manual Discharge Station
MDU	Motion Detection Unit
ME	Moisture Element
MG	Motor-Generator Set
MHDD	Moving Head Disc Drive
MI	Moisture Indicator
MIC	Moisture Indicating Controller
MIS	Moisture Indicating Switch
MM	Motor Module (TIP System)
MO	Motor Operator
MODEM	Modem
MON	Monitor
MPDS	Microprocessor Data System
MPS	Manual Pull Station
MR	Moisture Recorder
MS	Moisture Separator
MT	Mositure Transmitter
MTA	Dew Point Transmitter Amplif
MTS	Manual Transfer Switch
MUX	Multiplexer
MV	Manifold Valve
MV/I	M/Volt To Current Converter
MW	Microwave Receiver
MX	Mixer

Columbia Generating Station
Final Safety Analysis Report

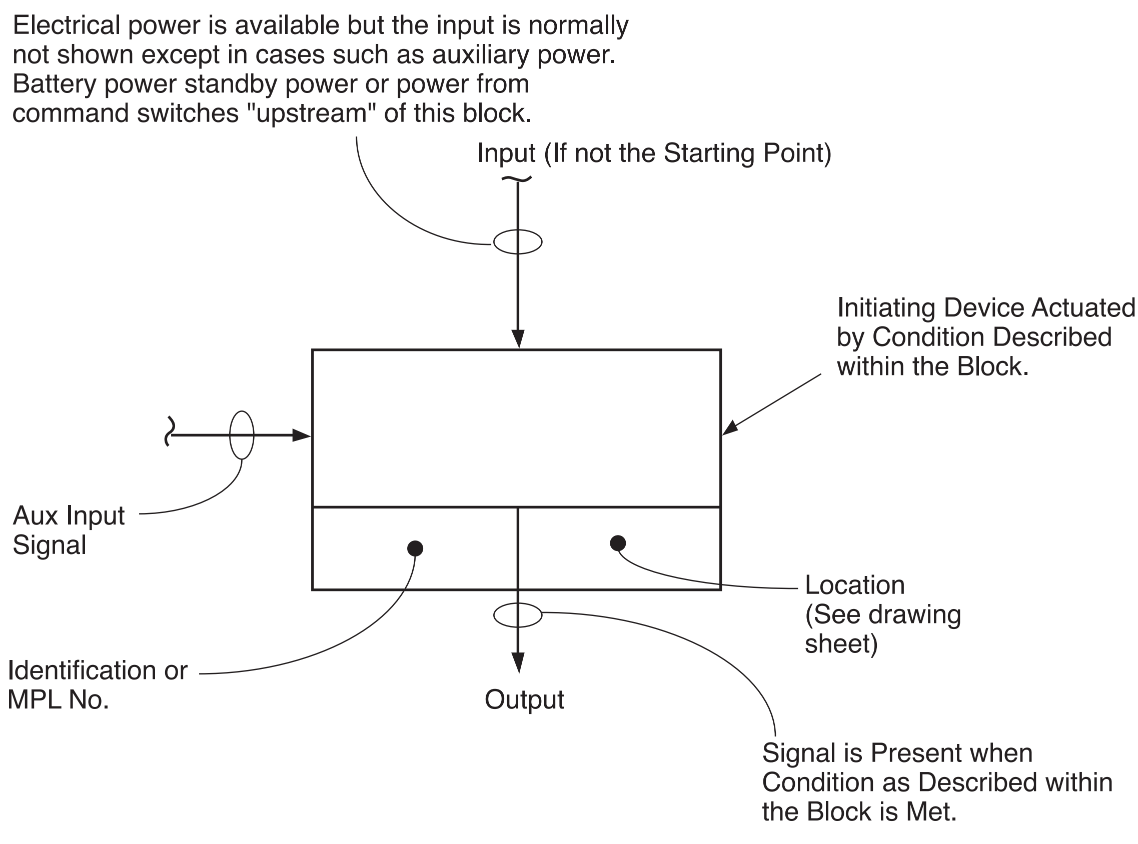
Equipment Acronyms

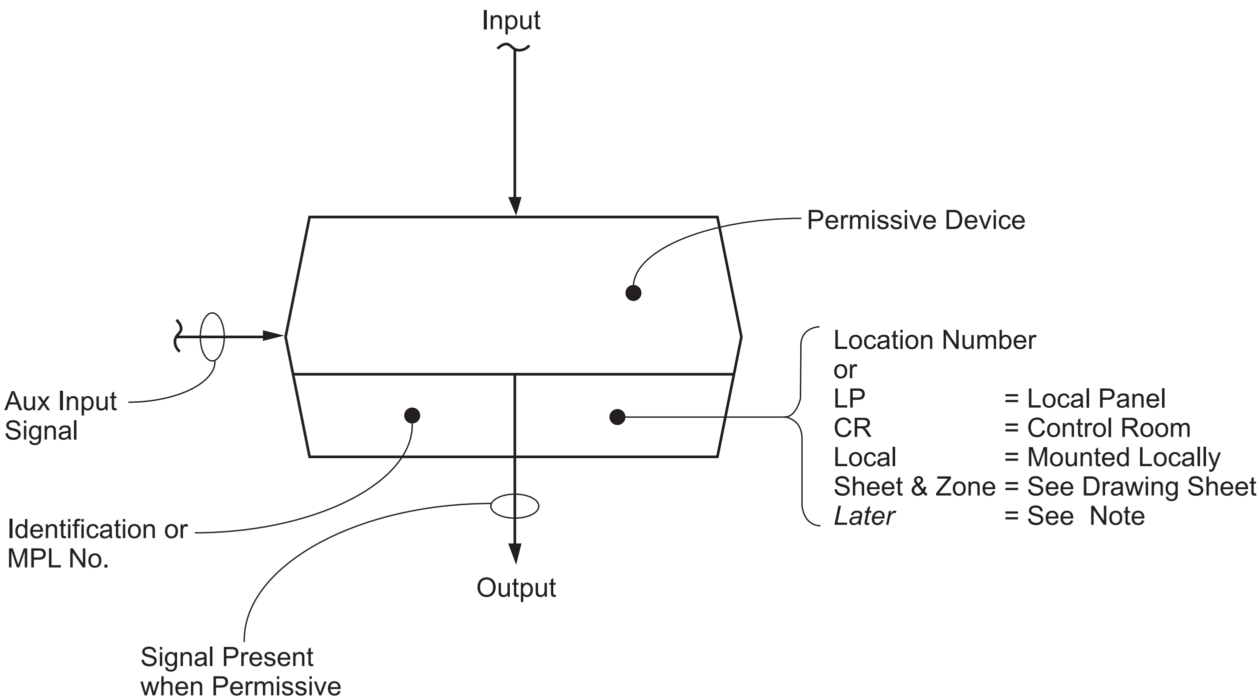
Draw. No. 950021.42	Rev.	Figure 1.2-28.1

Equipment Acronyms (con't)

MZ	Multizone Air Conditioner
NR	Neutral Grounding Resistor
O/M	Input/Out Module
O_2/H_2	Oxygen /Hydrogen$_2$
O_2E	Oxygen Element$_2$
O_2H_2R	Oxygen/Hydrogen Recorder$_2$
O_2I	Oxygen Indicator$_2$
O_2R	Oxygen Recorder$_2$
OS	Oil Separator
OSC	Oscillograph
OZG	Ozone Generator
P	Pump
P/B	Push Button
P/E	Pneumatic/Electric Converter
P/I	Pressure/Current Converter
P/P	Pressure Inverter
PA	Pre-Amps
PBU	Seismic Playback Unit
PC	Pressure Controller
PCV	Pressure Control Valve
PDM	Power Distribution Module
PDP	Power Distribution Panel
PE	Pressure Element
PH	Ph Ind Transmitter Recorder
PHB	Pneumatic Hydraulic Booster
PHC	Ph Controller
PHE	Ph Element
PHEC	Photoelectric Controller
PHED	Photoelectric Detector
PHIC	Ph Indicating Controller
PHIT	Ph Indicating Transmitter
PHITS	Ph Indicating Transmitter Switch
PHR	Ph Recorder
PHT	Ph Transmitter
PI	Pressure Indicator
PIC	Press Indicating Controller
PICS	Press Indicating Controller and Switch
PIS	Pressure Indicating Switch
PL	Programmable Logic Card
PLC	Programmable Logic Controller
PNL	Panel
POC	Disc Position Signal Conv
POE	Position Indication Element
POI	Position Indicator
POIC	Position Indicating Controller
POS	Position Switch
POT	Position Transmitter
POTR	Potentiometer "CL.1E Only"
POV	Pilot Operated Pop Off Valve
PP	Power Panel
PR	Pressure Recorder
PRN	Line Printer
PROG	Programmer
PRTM	Programmable Timer
PRV	Pressure Reg. Valve
PS	Pressure Switch
PT	Poten. Xmfer Or Press. Transm.
PTA	Barometric Pressure Amplifier
PTD	Pressure Transducer
PTZM	Pan Tilt Zoom Monitor
PUI	Purity Indicator
PUIT	Purity Indicator Transmitter
PUS	Purity Switch
QCC	Quick Couple Connection
QDC	Quick Disconnect
QHM	Run Time Meter
QSV	Quick Acting Solenoid Valve
R	Reservoir
R/I	Resistance/Current Converter
RA	Radiation Amplifier
RAD	Radiation Mon. Control Board
RC	Radiation Controller
RCM	Respirator Cleaning Module
RD	Rupture Disc
RDCC	Rod Drive Control Cabinet
RDD	Rod Detector Display
RE	Radiation Element
RECT	Rectifier
REL	Relay
RES	Resistor
RF	Refrigeration Machine(OG)
RFM	Radio Frequency Monitor
RG	Regulator
RI	Radiation Indicator
RIS	Radiation Indicating Switch
RLY	Relay
RM	Radiation Monitor
RMC	Remote Manual Controller
RMS	Remote Manual Switch
RO	Restricting Orifice
ROD	Control Rod
RPIS	Rod Position and Info Sys.
RPV	Reactor Pressure Vessel
RR	Radiation Recorder
RRM	Refrigerant Recovery Machine
RSA	Response Spectrum Annunciator
RSCC	Rod Sequence and Control Cab
RSDP	Rod Sequence Display Panel
RSM	Radiation Sampler
RSMD	Rod Select Module
RSR	Response Spectrum Recorder
RSRT	RSR Transducer for RSA
RST	Resin Trap
RT	Radiation Transmitter
RTM	Run Time Meter
RV	Relief Value
RVT	Roof Ventilator
S	Electronic Trip Unit
SC	Speed or Seismic Controller
SCAN	Scanner
SCL	Scaler
SCR	Screen
SE	Speed Element
SEW	Safety Eye Wash/Shower
SF	Spectacle Flange
SH	6.9 Kv Switch Gear
SHRED	Shredder
SI	Speed Indicator
SIOA	Silicon and Oxygen Analyzer
SL	480 Volt Switch Gear
SM	4.16 Kv Switch Gear
SMA	Smoke Alarm, Surface Mt. Acceler.
SMD	Smoke Detector
SNB	Snubber
SOL	Solenoid (Mech. Linkage)
SP	Sample Probe
SPC	Spacer
SPS	Speed Switch (Temp. Entry)
SPV	Solenoid Pilot Valve
SPVD	Set Press Verification Device
SQRT	Square Root Extractor
SR	Sample Rack
SRU	Signal Resistor Unit
SS	Speed or Seismic Switch
SSW	Step Switch
ST	Strainer
SUH	Steam Unit Heater
SUM	Summer
SUMP	Sump
SV	Solenoid Valve
SYNC	Synchroscope Meter
T	Trap
T/SS	Temp Selector Switch
TA	Trip Auxiliary Unit
TAPE	Magnetic Tape Unit
TAS	Tamper Alarm Switch
TB	Terminal Box
TBE	Turbidity Element
TBIT	Turbidity Indicating Trans
TBR	Turbidity Recorder
TBS	Turbidity Switch
TBT	Turbidity Transmitter
TC	Temperature Controller
TCV	Temperature Control Valve
TD	Time Delay
TDS	Time Delay Relays
TE	Temperature Element
TE/ME	Temperature/Moisture Element
TEST	Test (MEL Diagnostics)
THD	Thermal Detector
TI	Temperature Indicator
TIC	Temperature Indicating Controller
TIS	Temperature Indicating Switch
TJR	Temperature Scanning Recorder
TK	Tank
TM	Timer
TN	Turn Style
TNG	Turning Gear
TPA	Triaxial Peak Accelerograph
TPSA	Testable Pipe Spool Assembly
TQ	Time Totalizer
TQR	Torque Recorder
TQS	Torque Switch
TQT	Torque Transmitter
TR	Temp./ Triax. Record./Transform.
TRB	Terminal Block
TRC	Temperature Recorder Controller
TRL	Translator
TRS	Temperature Recording Switch
TS	Temperature Switch
TSC	Temperature Scanner
TT	Temperature Transmitter
TT/MT	Temperature/Moisture Transmitter
TUBE	LPRM Guide Tube Assembly
TV	Test Valve
TW	Thermal Well
TY	SMA HVAC, Special Func. Relay
UFM	Uniplex Field Module
USG	Ultra-Sonic Generator
UTD	Ultra-Sonic Transducer
UV/OR	UV Oxidation Reactor
UVD	Ultra-Violet Detector
V	Valve
V/F	Voltage/Freq. Converter
VARM	Var. Meter
VATD	Var. Transducer
VBAM	Vibration Differential Amp
VBE	Vibration Element
VBEC	Vibration/Eccentricity Indicator
VBI	Vibration Indicator
VBIS	Vibration Indicating Switch
VBR	Vibration Recorder
VBS	Vibration Switch
VCR	Video Cassette Recorder
VD	Viewing Device
VE	Vibration Element
VIR	Vibration Instrument Rack
VM	Voltmeter
VMP	Vibration Monitoring Panel
VPI	Valve Pos. Indication System
VSC	Variable Speed Controller
VT	Velocity Transmitter
VTD	Voltage Transducer
VX	Process Instrument Valve
VZ	Vaporizer
W	Watt
WDA	Wind Direction Amplifier
WDR	Wind Direction Recorder
WDT	Wind Direction Transmitter
WELL	Well (For PSD System Only)
WHM	Watt Hour Meter
WM	Watt Meter
WR	Water Reprocessing Unit
WSA	Wind Speed Amplifier
WSR	Wind Speed Recorder
WST	Wind Speed Transmitter
WTD	Watt Transducer
WUH	Water Unit Heater
X	Primary Containment Penetration
XAR	Resid. Chlorine Analyzer Recorder
XAY	Analyzer, Special Types
XD	Explosives Detector
XE	Element, Special Types
XI	Indicator, Special Types
XR	Recorder, Special Types
XS	Sensor, Special Types
XT	Transmitter, Special Types
ZONE	Fire Protection Zone Desig.
ZS	Tamper Switch

This block is the command switching or primary actuating function. This block can represent a switch, valve probe timer, or trip circuit. This block is normally the starting point of a functional sequence with an output only, but can have input and aux. input depending on the type of device. The same device may have a number of outputs, but each functional sequence initiated shall be shown by an individual block showing the same identification number and cross-reference. (See drawing sheet.)



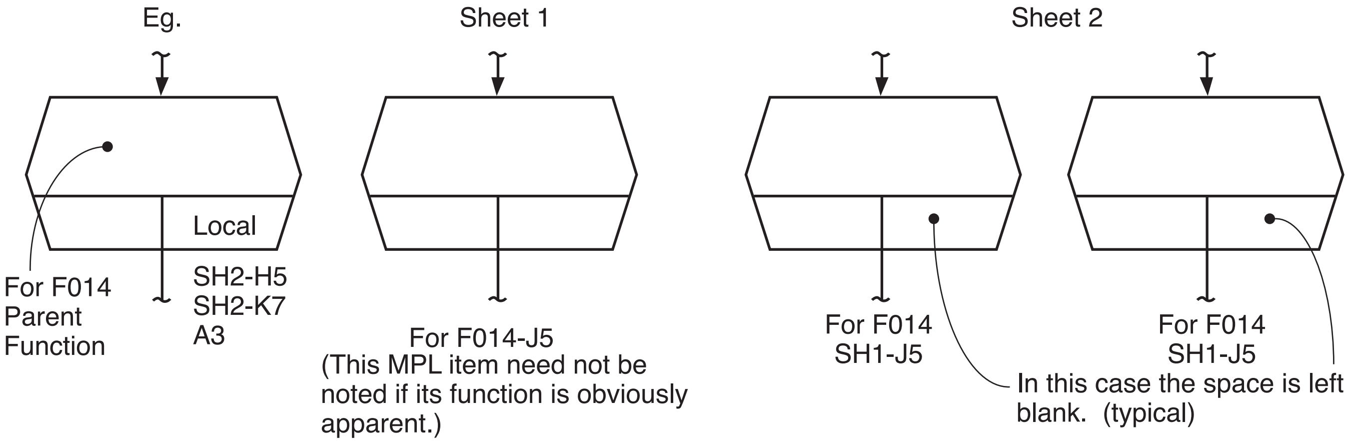


This block defines a permissive function which must be satisfied to permit the signal flow to pass to the next block. This block has incoming, outgoing, and may have auxiliary signals. The output from this permissive may be sealed in.

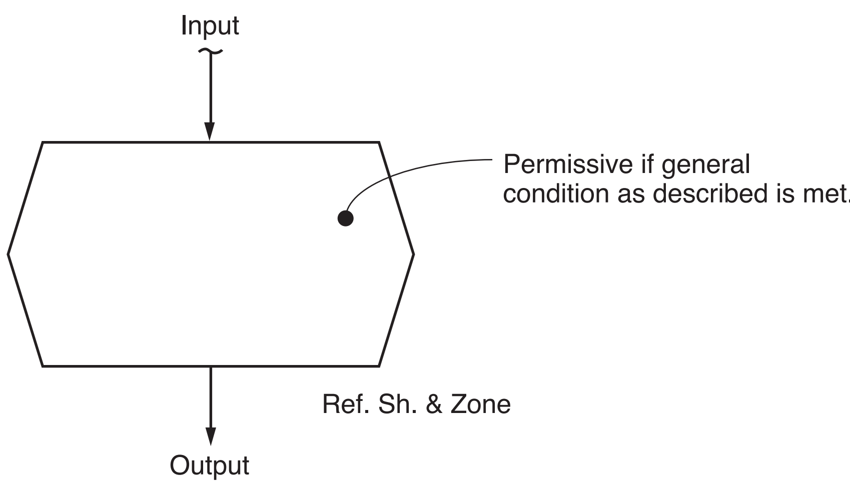
Note:

The word *later* may be used if the location is unknown but the correct location shall be noted on a future revision.

If a permissive or a primary function is shown in more than one place on drawings, provide a cross-reference to the parent function. (Formally an "X" was shown in the location of other switch handle positions, indicating that their blocks were an intricate part of the numbered switch assembly, but a different position of the switch handle. The "X" in location is inactive for new design.)



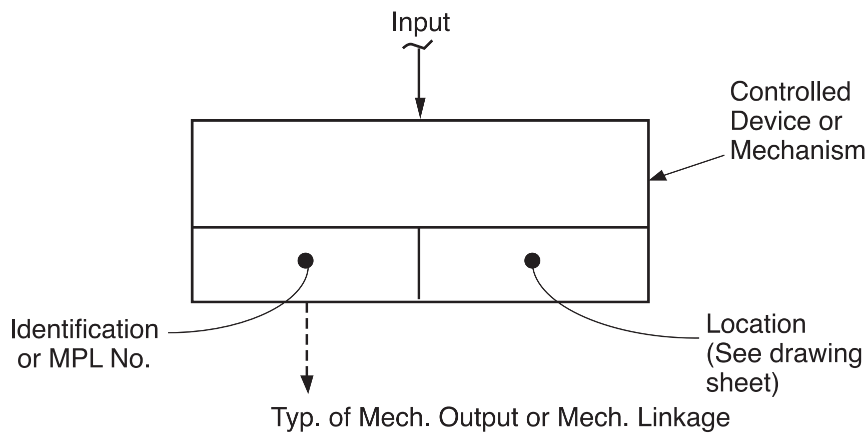
Show MPL item number of the valve or equipment served adjacent to the permissive or primary function. (See example)



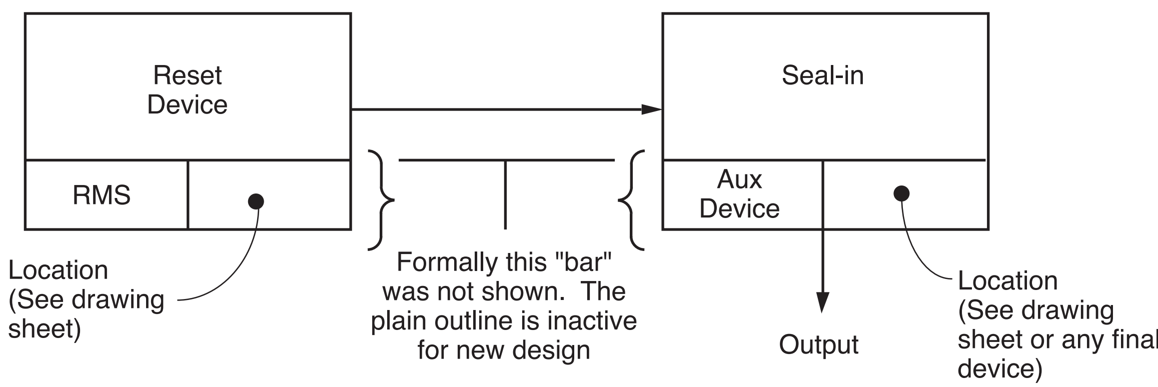
This block is a permissive condition.

Where the permissive is a general condition and not identified with a single device, the outer enclosure only is shown. It only has an input and output.

Logic Symbols for NSSS Functional Control Diagrams

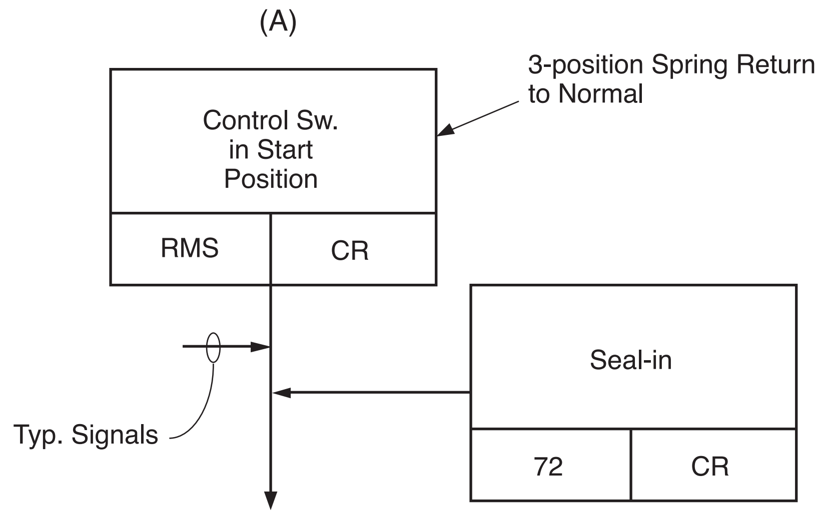


This block is a final device. It can be a relay, valve, electro-mech. sw., etc. Normally it has only inputs, but can have mech. outputs or position switch outputs.

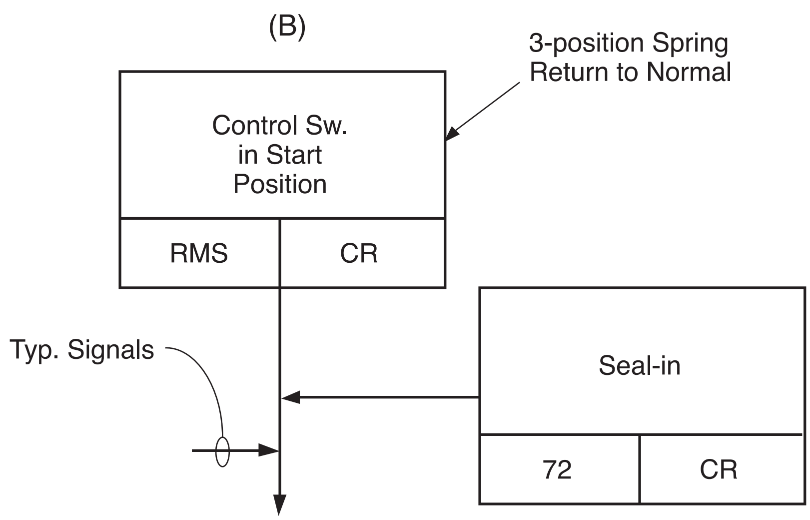


This is a seal-in with a manual reset device. The function of the seal-in is to latch in a signal and to continue that signal until manually reset. A seal-in shown without a reset device implies that the reset device is part of, and located on the nearest valve or contactor and is automatically reset by breaking the signal downstream of the seal-in signal. In all other cases the reset device shall be shown in conjunction with the seal-in.

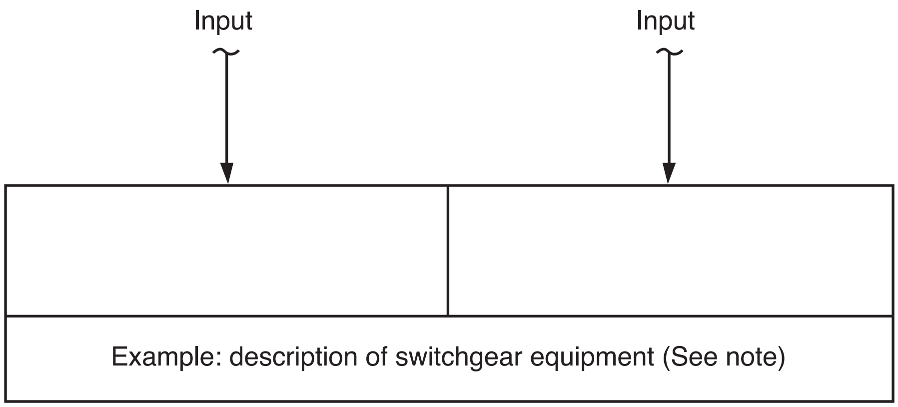
Examples of Typical Seal-in Blocks.



In this case, all signals at this point would be sealed-in.



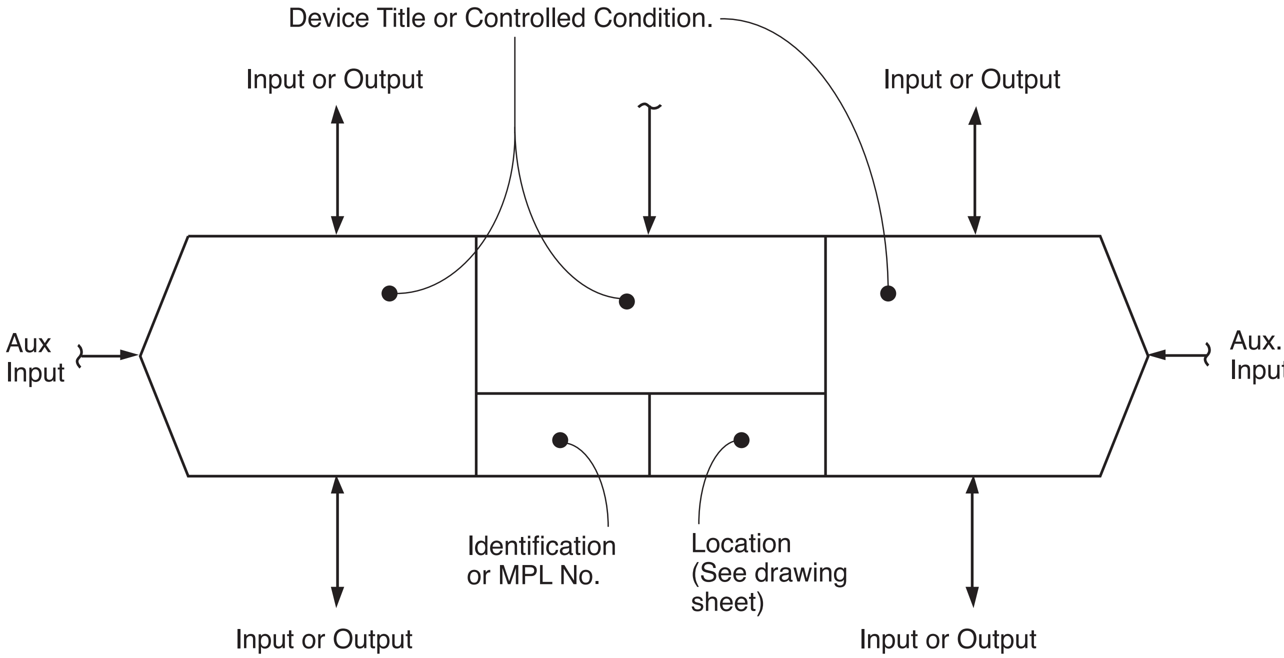
In this case, only the control sw. signal would be sealed-in.



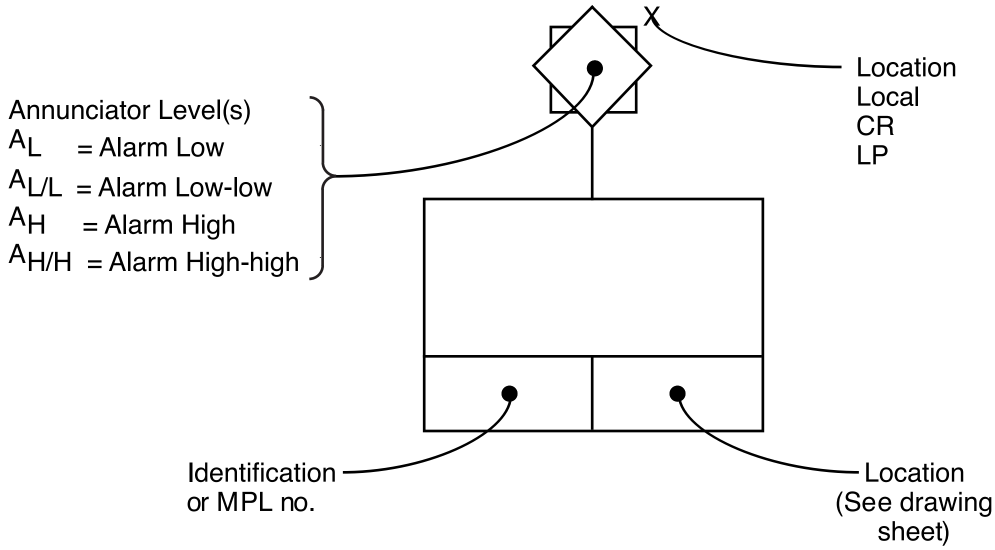
This block is a final device used to represent motor starters, circuit breakers, etc. It has only input signals. The input to the right causes an opposed action to the input on the left, such as left-open: right-close.

Note:

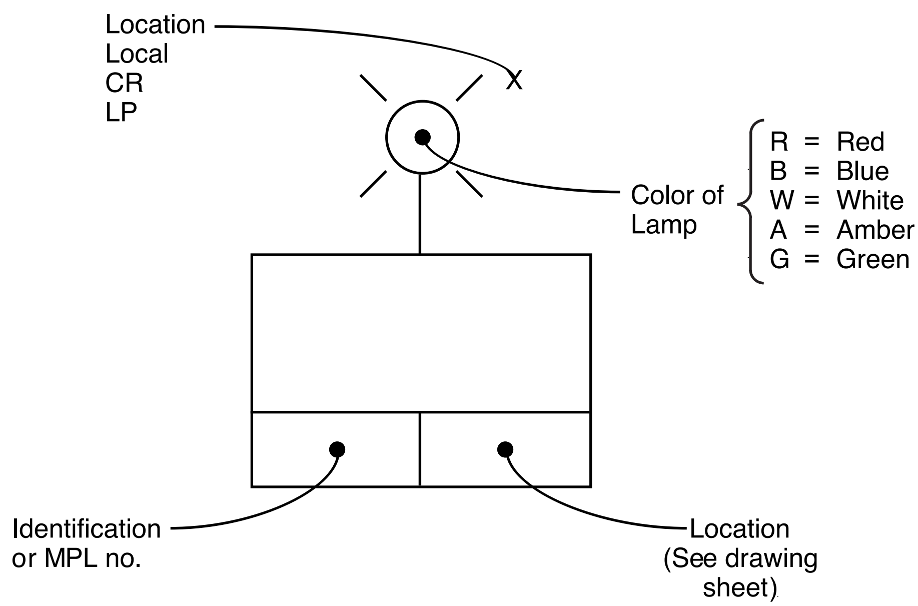
A final device may have more than one input. Each of these inputs can initiate the block. The block can have electrical inputs to indicating devices. Switchgear descriptions are found in ANSI spec. C37.2.



This block is a permissive operated by devices such as valve or pump switchgear designated in the inner block. This condition or device effects the operation of the final device. It has elect. inputs, mech. inputs, aux. inputs (mech. or elec.), and mech. or elect. outputs. This device is normally a valve. This is also used for other input/output power sources such as air or hydraulic. A solenoid pilot valve for an air operated valve is an example of this type of device (see Figure 1.2-26). When the two side blocks are the controlling blocks they have aux. input signals.



This block is a primary function for annunciators.



This block is a primary function for ind. lights.

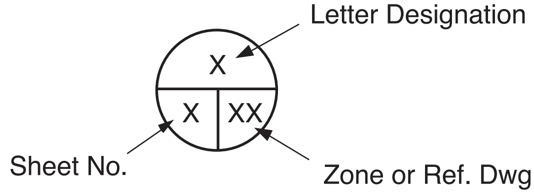
Logic Symbols for NSSS
Functional Control Diagrams

This line represents an elec. flow signal. This line may actuate a final device and may be used to represent actuation of a permissive block.

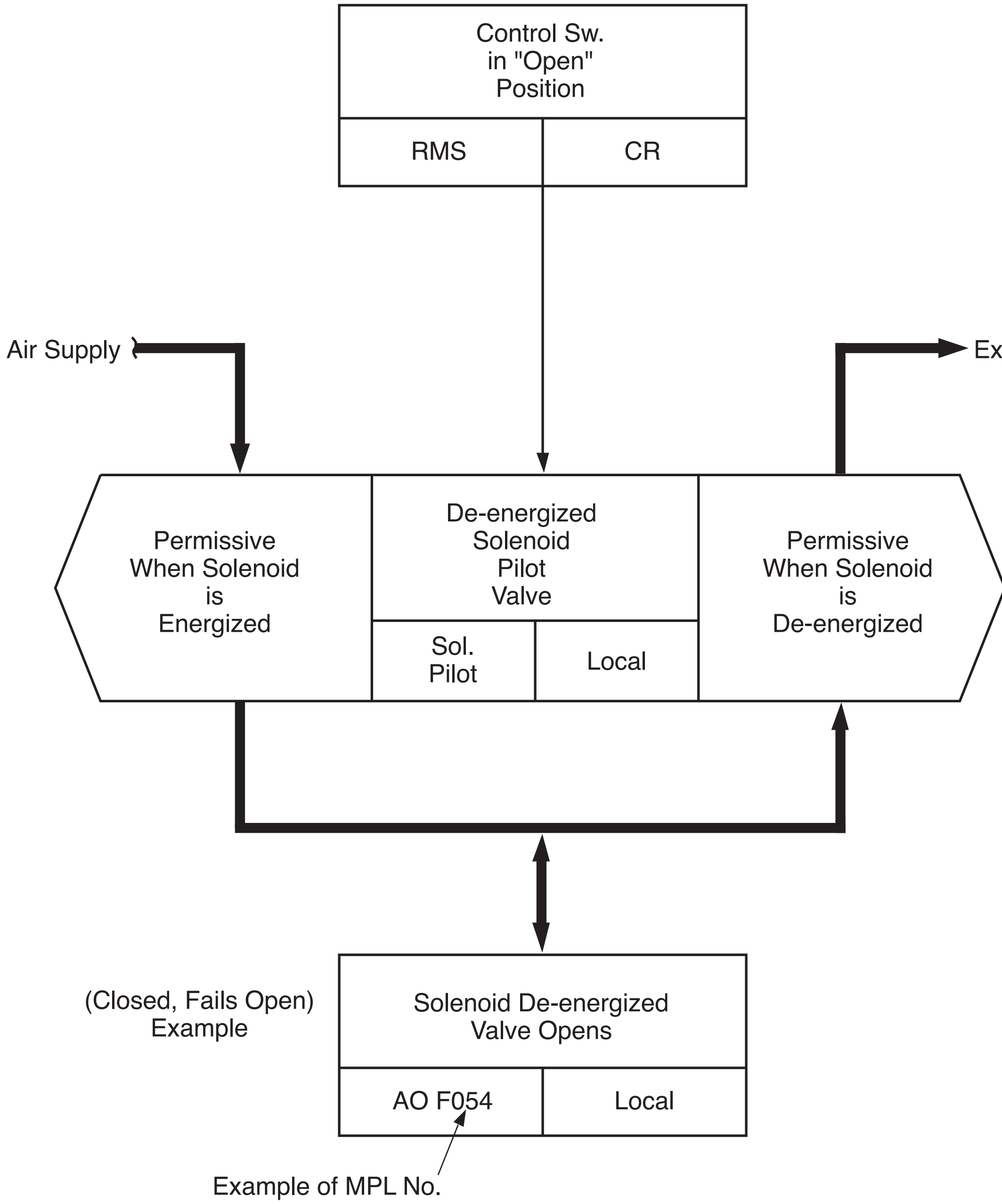
This line represents an auxiliary signal source such as air or hydraulic, and is not electrical.

This line represents mechanical outputs and /or mechanical linkage.

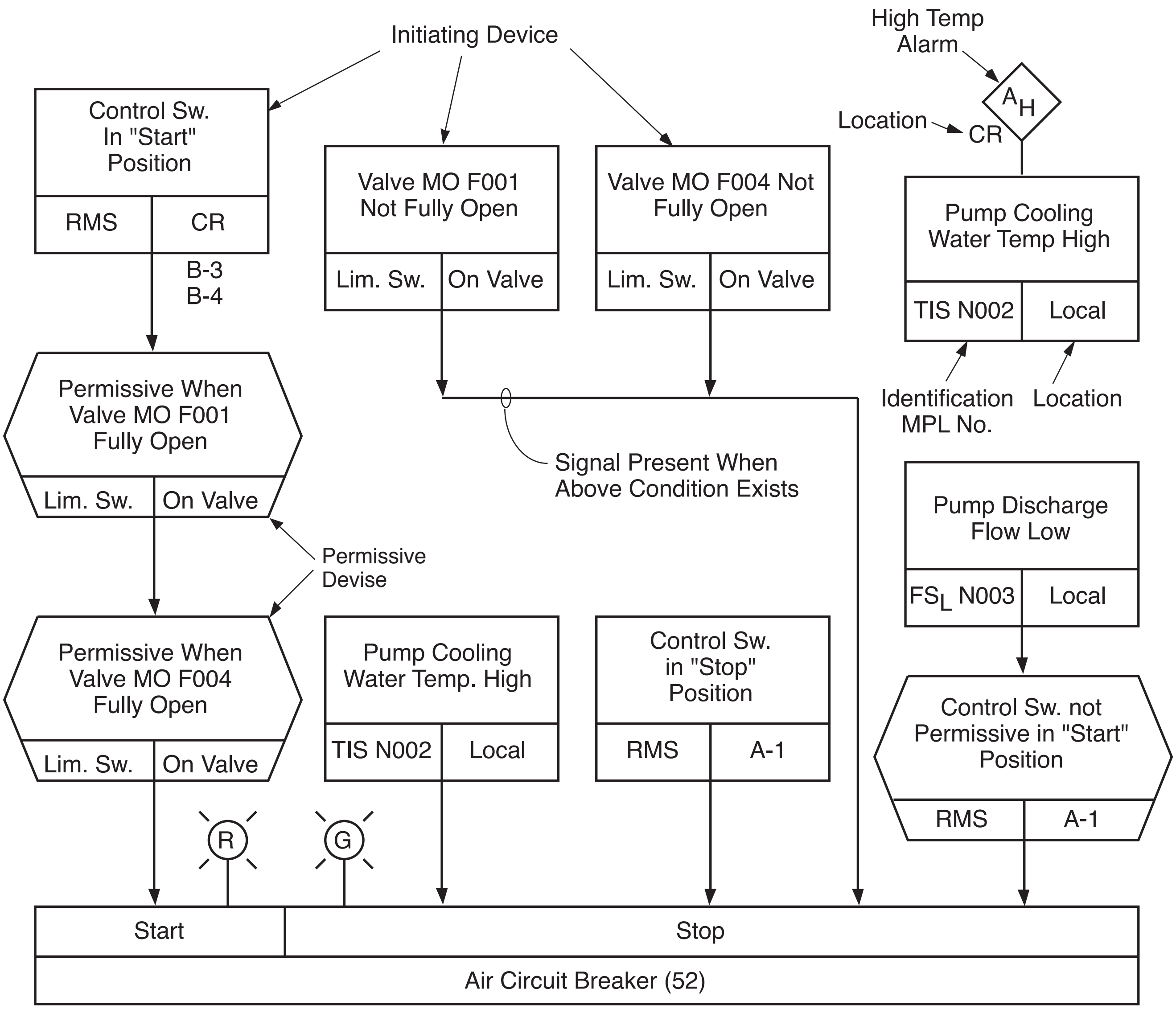
This symbol represents the start of the primary initiating signal.



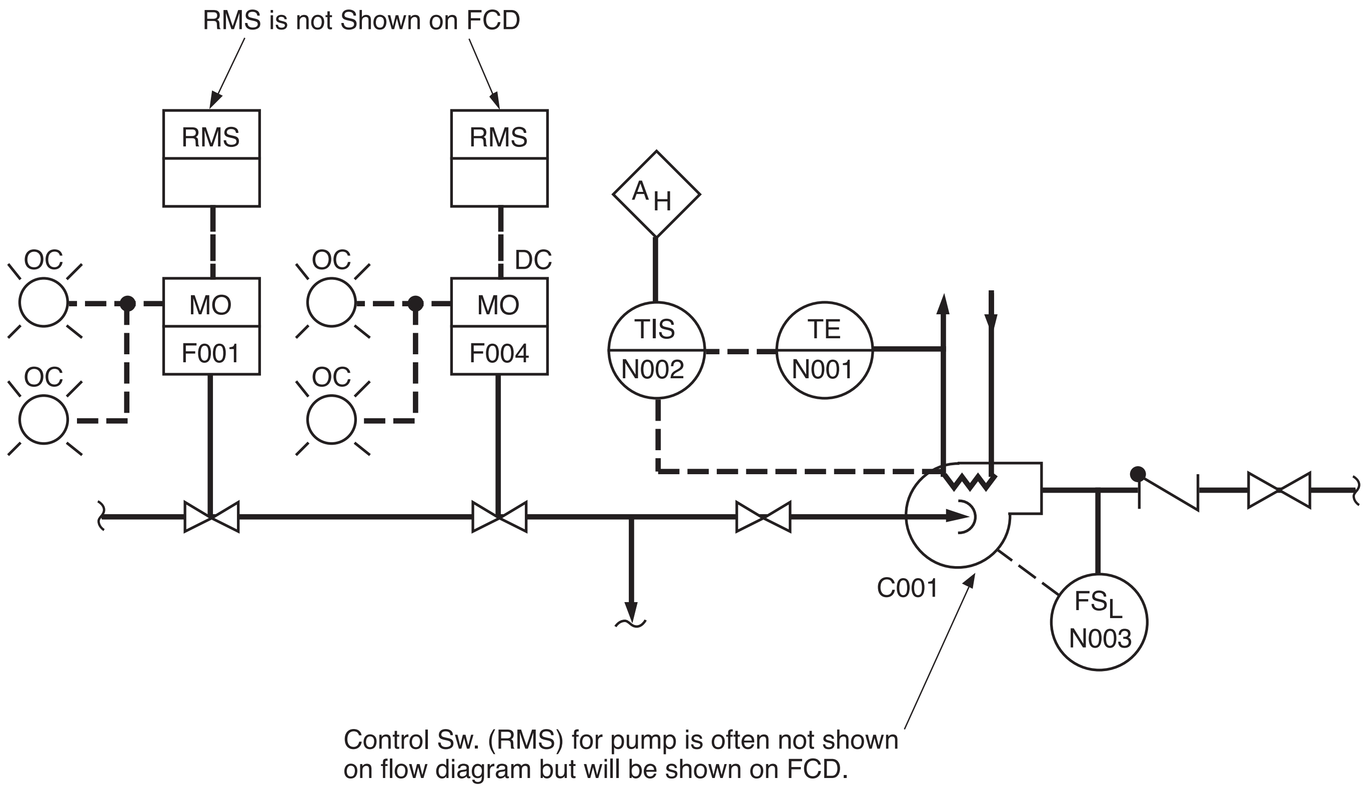
This symbol represents a match circle. The letter designation on one dwg. must match the letter on the interfacing dwg.

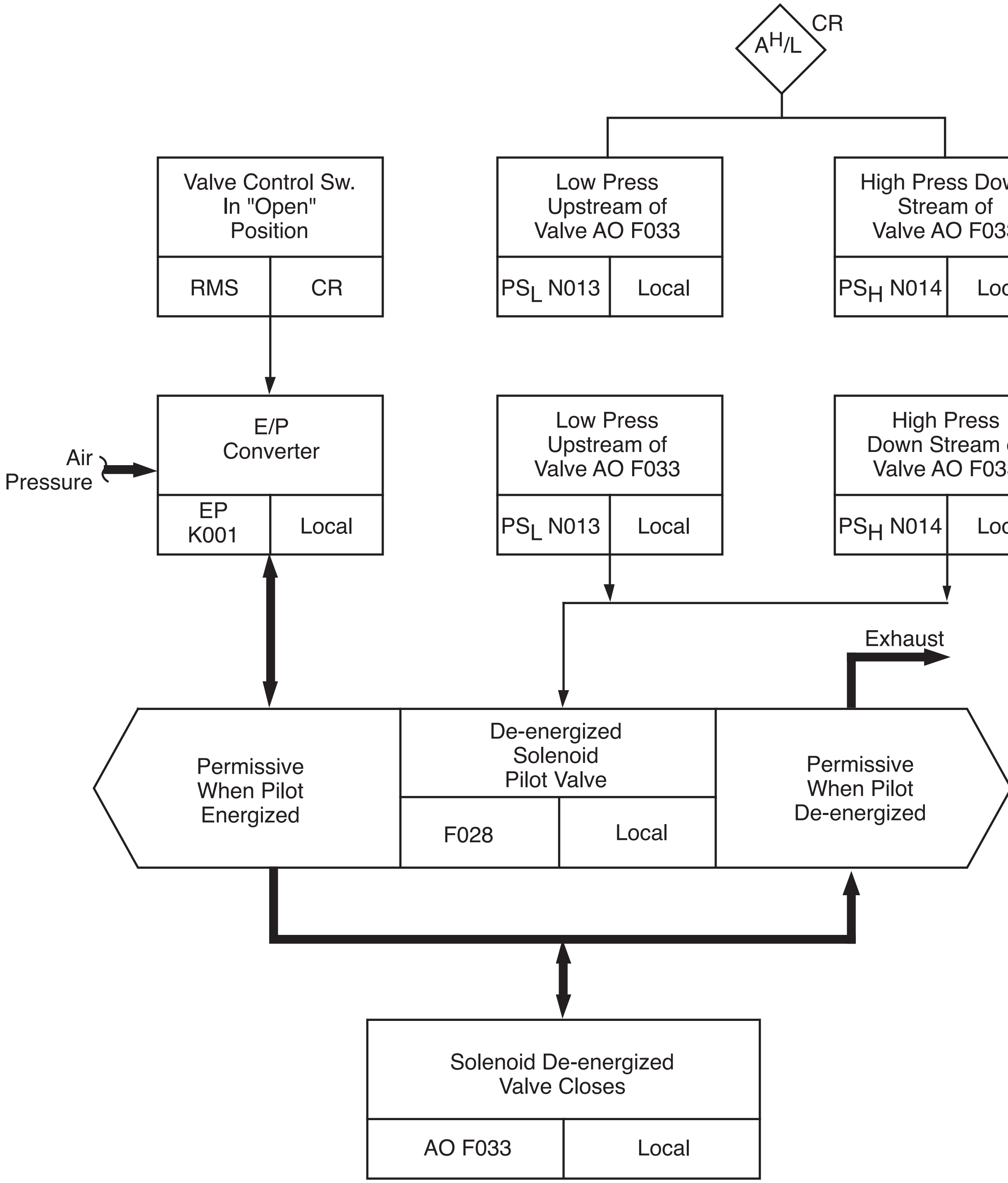


Typical A.O. Valve Example


Initiating Device
High Temp
Alarm
Location
CR
AH
Control Sw.
In "Start"
Position
RMS
CR
B-3
B-4
Valve MO F001
Not Fully Open
Lim. Sw.
On Valve
Valve MO F004 Not
Fully Open
Lim. Sw.
On Valve
Pump Cooling
Water Temp High
TIS N002
Local
Identification
MPL No.
Location
Permissive When
Valve MO F001
Fully Open
Lim. Sw.
On Valve
Signal Present When
Above Condition Exists
Pump Discharge
Flow Low
FSL N003
Local
Permissive
Devise
Permissive When
Valve MO F004
Fully Open
Lim. Sw.
On Valve
Pump Cooling
Water Temp. High
TIS N002
Local
Control Sw.
in "Stop"
Position
RMS
A-1
Control Sw. not
Permissive in "Start"
Position
RMS
A-1
R
G
Start
Stop
Air Circuit Breaker (52)

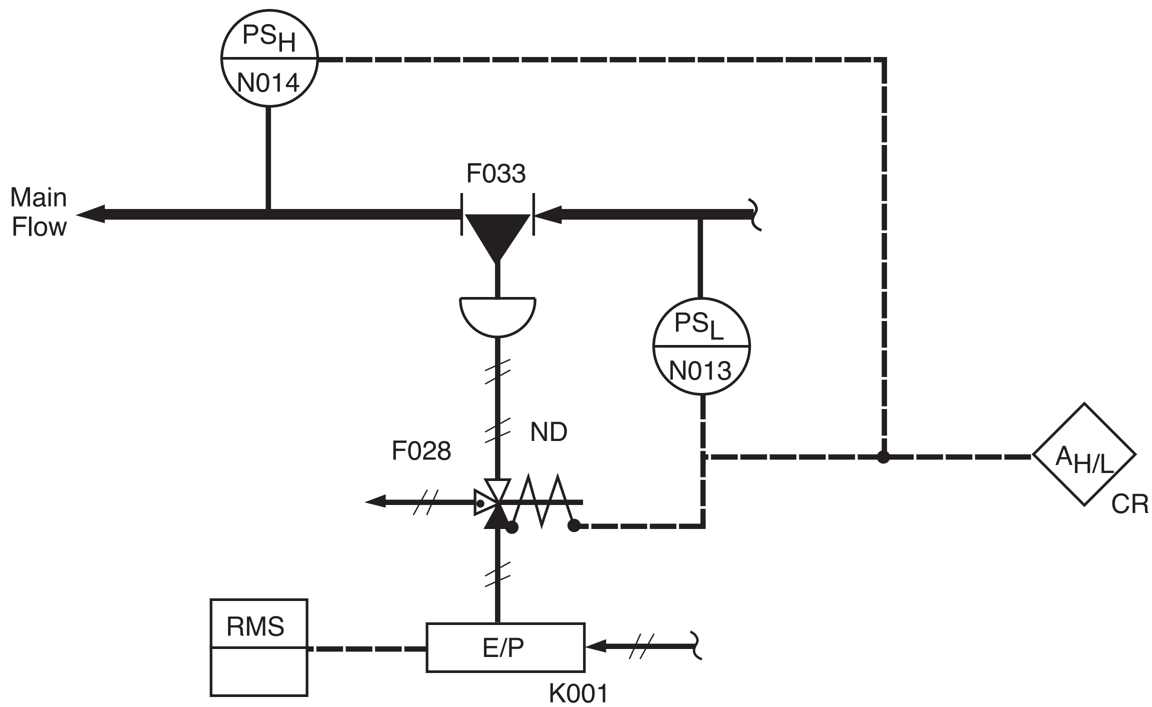
Typical Flow Diagram Example





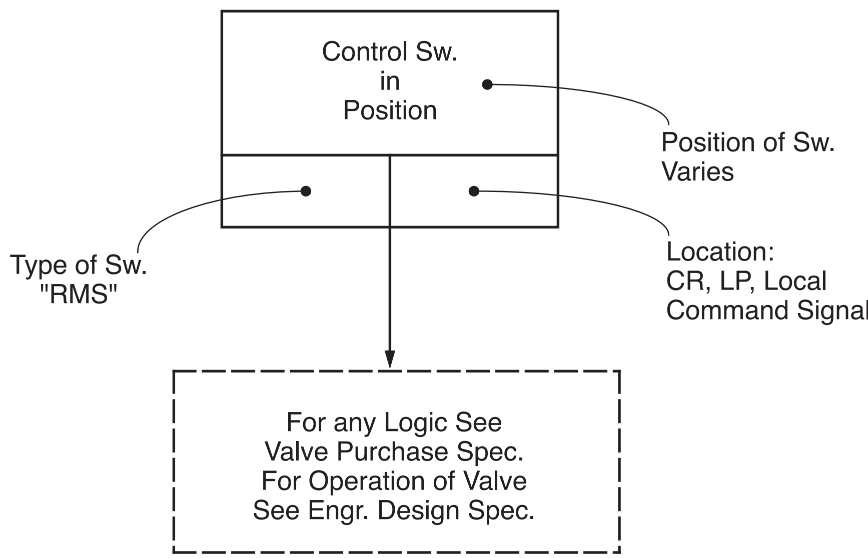
Bleed-off Flow Control Valve AO F033
Functional Control Diagram

Typical Flow Diagram Example



Notes: 1. Aux relays and devices are not shown on FCD.

This Figure is for a Typical Check Valve.



Testable Check Valve AO _ _ _ _

Reason for the above change is to have one standard logic for all testable check valve AO regardless of manufacturer.

1.3 COMPARISON TABLES

The italicized information is historical and was provided to support the application for an operating license.

1.3.1 COMPARISONS WITH SIMILAR FACILITY DESIGNS

This section highlights the principal design features of CGS and compares its major features with other boiling water reactor (BWR) facilities. The design of this facility is based on proven technology obtained during the development, design, construction, and operation of BWRs of similar types. The data, performance, characteristics, and other information presented here represent the design of the facilities at the time of the CGS operating license review.

1.3.1.1 Nuclear Steam Supply System Design Characteristics

Table 1.3-1 summarizes the design and operating characteristics for the nuclear steam supply systems. Parameters are related to rated power output for a single plant unless otherwise noted. The fuel thermal, hydraulic, and nuclear design data are that for the initial core load. Cycle specific data are provided in Chapter 4, Section 5.2, and Appendix 15F.

1.3.1.2 Power Conversion System Design Characteristics

Table 1.3-2 compares the power conversion system design characteristics.

1.3.1.3 Engineered Safety Features Design Characteristics

Table 1.3-3 compares the engineered safety features design characteristics.

1.3.1.4 Containment Design Characteristics

Table 1.3-4 compares the containment design characteristics.

1.3.1.5 Radioactive Waste Management Systems Design Characteristics

Table 1.3-5 compares the radioactive waste management design characteristics.

1.3.1.6 Structural Design Characteristics

Table 1.3-6 compares the structural design characteristics.

1.3.1.7 Electrical Power Systems Design Characteristics

Table 1.3-7 compares the electrical power systems design characteristics.

1.3.2 COMPARISON OF FINAL AND PRELIMINARY INFORMATION

Significant changes that have been made in the facility design since submission of the PSAR are listed in Table 1.3-8. Items in Table 1.3-8 are cross referenced to the appropriate portion of the FSAR that describes the changes and the bases for them.

Table 1.3-1

Comparison of Nuclear Steam Supply System Design Characteristics[a]

	CGS[b] BWR 5 251-764	HATCH 1[c] BWR 4 218-560	ZIMMER[c] BWR 5 218-560
Thermal and Hydraulic Design (see Section 4.4)			
Rated power (MWt)	3323	2436	2436
Design power (MWt) (ECCS design basis)	3468	2550	2550
Steam flow rate (1b/hr)	14.295×10^6	10.03×10^6	10.477×10^6
Core coolant flow rate (1b/hr)	108.5×10^6	78.5×10^6	78.5×10^6
Feedwater flow rate (1b/hr)	14.256×10^6	10.445×10^6	10.477×10^6
System pressure, nominal in steam dome (psia)	1020	1020	1020
Average power density (KW/liter)	49.15	51.2	50.51
Maximum thermal output (KW/ft)	13.4	13.4	13.4
Average thermal output (KW/ft)	5.38	7.11	5.45
Maximum heat flux (Btu/hr-ft^2)	428,360	428,300	354,000
Average heat flux (Btu/hr-ft^2)	145,060	164,700	143,900
Maximum UO_2 temperature (°F)	4380	4380	3325
Average volumetric fuel temperature (°F)	1100	1100	1100
Average cladding surface temperature (°F)	558	558	558
Minimum critical power ratio (MCPR)	1.24	1.9[d]	1.21
Coolant enthalpy at core inlet (Btu/1b)	527.6	526.2	527.4
Core maximum exit voids within assemblies	79	79	75
Core average exit quality (% steam)	13.5	12.9	13.6
Feedwater temperature (°F)	420	387.4	420
Design power peaking factor			
Maximum relative assembly power	1.40	1.40	1.40
Local peaking factor	1.15	1.24	1.24
Axial peaking factor	1.40	1.5	1.4
Total peaking factor	2.51	2.6	2.43

Table 1.3-1

Comparison of Nuclear Steam Supply System Design Characteristics[a] (Continued)

	CGS[b] BWR 5 251-764	HATCH 1[c] BWR 4 218-560	ZIMMER[c] BWR 5 218-560
Nuclear Design (First Core) (see Section 4.3)			
Water/UO_2 volume ratio (cold)	2.55	2.53	2.41
Reactivity with strongest control rod out (k_{eff})	< 0.99	< 0.99	< 0.99
Moderator void coefficient			
Hot, no voids (Δk/k - %void)	-1.0×10^{-3}	-1.0×10^{-3}	-1.0×10^{-3}
At rated output (Δk/k - %void)	-1.6×10^{-3}	-1.6×10^{-3}	1.6×10^{-3}
Fuel temperature doppler coefficient			
At 68°F (Δk/k - °F fuel)	-1.3×10^{-5}	-1.3×10^{-5}	-1.3×10^{-5}
Hot, no voids (Δk/k - °F fuel)	-1.2×10^{-5}	-1.2×10^{-5}	-1.2×10^{-5}
At rated output (Δk/k - °F fuel)	-1.3×10^{-5}	-1.3×10^{-5}	-1.3×10^{-5}
Initial average ^{235}U enrichment wt (%)	1.91	2.23	1.90
Fuel average discharge exposure (MWd/short ton)	10,300	19,000	15,053
Core Mechanical Design (see Sections 4.2 and 7.6)			
Fuel assembly			
Number of fuel assemblies	764	560	560
Fuel rod array	8 x 8	7 x 7	8 x 8
Overall dimensions (in.)	176	176	176
Weight of UO_2 per assembly (1b) (pellet type)	458 (chamfered)	490.4 (undished) 483.4 (dished)	465.15
Weight of fuel assembly (1b)	600	681 (undished) 675 (dished)	698

Table 1.3-1

Comparison Of Nuclear Steam Supply System Design Characteristics[a] (Continued)

	CGS[b] BWR 5 251-764	HATCH 1[c] BWR 4 218-560	ZIMMER[c] BWR 5 218-560
Core Mechanical Design (see Sections 4.2 and 7.6) (Continued)			
Fuel rods (NEDE-20944P)			
Number per fuel assembly	62	49	63
Outside diameter (in.)	0.483	0.563	0.493
Cladding thickness (in.)	0.032	0.032	0.034
Cap. pellet to cladding (in.)	0.0045	0.006	0.0045
Length of gas plenum (in.)	10	16	14
Cladding material[e]	Zircaloy-2	Zircaloy-2	Zircaloy-2
Fuel pellets			
Material	UO_2	UO_2	UO_2
Density (% of theoretical)	95	95	95
Diameter (in.)	0.410	0.487	0.416
Length (in.)	0.410	0.5	0.420
Fuel channel			
Overall dimension, length (in.)	166.9	166.9	166.9
Thickness (in.)	0.100	0.080	0.100
Cross section dimensions (in.)	5.494 x 5.494	5.44 x 5.44	5.48 x 5.48
Material	Zircaloy-4	Zircaloy-4	Zircaloy-4
Core assembly			
Fuel weight as UO_2 (1b)	349,900	272,850	260,538
Core diameter (equivalent) (in.)	187.1	160.2	160.2
Core height (active fuel) (in.)	150	144	146

Table 1.3-1

Comparison of Nuclear Steam Supply System Design Characteristics[a] (Continued)

	CGS[b] BWR 5 251-764	HATCH 1[c] BWR 4 218-560	ZIMMER[c] BWR 5 218-560
Core Mechanical Design (see Sections 4.2 and 7.6) (Continued)			
Reactor control system			
Method of variation of reactor power	*Movable control rods and variable forced coolant flow*	*Movable control rods and variable forced coolant flow*	*Movable control rods and variable forced coolant flow*
Number of movable control rods	*185*	*137*	*137*
Shape of movable control rods	*Cruciform*	*Cruciform*	*Crucifonn*
Pitch of movable control rods	*12.0*	*12.0*	*12.0*
Control material in movable rods	*B_4C granules compacted in SS tubes*	*B_4C granules compacted in SS tubes*	*B_4C granules compacted in SS tubes*
Type of control rod drives	*Bottom entry locking piston*	*Bottom entry locking piston*	*Bottom entry locking piston*
Type of temporary reactivity control for initial core	*Burnable poison; gadoliniaurania fuel rods*	*Burnable poison; gadoliniaurania fuel rods*	*Burnable poison; gadoliniaurania fuel rods*
In-core neutron instrumentation			
Number of in-core neutron detectors (fixed)	*172*	*124*	*124*
Number of in-core detector assemblies	*43*	*31*	*31*
Number of detectors per assembly	*4*	*4*	*4*
Number of flux mapping neutron detectors	*5*	*4*	*4*

Table 1.3-1

Comparison of Nuclear Steam Supply System Design Characteristics[a] (Continued)

	CGS[b] BWR 5 251-764	HATCH 1[c] BWR 4 218-560	ZIMMER[c] BWR 5 218-560
<u>Core Mechanical Design</u> (see Sections 4.2 and 7.6) (Continued)			
In-core neutron instrumentation (Continued)			
Range (and number) of detectors			
Source range monitor	Source to 0.001% power (4)[f]	Source to 0.001% power (4)[f]	Source to 0.001% power (4)[f]
Intermediate range monitor	0.001% to 10% power (8)[f]	0.001% to 10% power (8)[f]	0.001% to 10% power (8)[f]
Local power range monitor	5% to 125% power (172)[f]	5% to 125% power (124)[f]	5% to 125% power (124)[f]
Average power range monitor	2.5% to 125% power (6)[f]	2.5% to 125% power (6)[f]	2.5% to 125% power (6)[f]
Number and type of in-core neutron sources	7 Sb-Be	5 Sb-Be	5 Sb-Be
<u>Reactor Vessel Design</u> (see Section 5.3)			
Material	Carbon steel stainless clad	Carbon steel stainless clad	Carbon steel stainless clad
Design pressure (psig)	1250	1265	1250
Design temperature (°F)	575	575	575
Inside diameter (ft-in.)	20-11	18-2	18-2
Inside height (ft-in.)	72-11	69-4	69-4
Minimum base metal thickness (cylindrical section) (in.)	6.75	5.53	5.375
Minimum cladding thickness (in.)	1/8	1/8	1/8

Table 1.3-1

Comparison Of Nuclear Steam Supply System Design Characteristics[a] (Continued)

	CGS[b] BWR 5 251-764	HATCH 1[c] BWR 4 218-560	ZIMMER[c] BWR 5 218-560
<u>Reactor Coolant Recirculation Design</u> (see Sections 5.1, 5.2, and 5.4)			
Number of recirculation loops	2	2	2
Design pressure:			
Inlet leg (psig)	1250	1148	1250
Outlet leg (psig)	1650;[g] 1550[h]	1274	1675;[g] 1575[h]
Design temperature (°F)	575	562	575
Pipe diameter (in.)	24	28	20
Pipe material (ANSI)	304/316	304/316	304/316
Recirculation pump flow rate (gpm)	47,200	42,200	33,880
Number of jet pumps in reactor	20	20	20
<u>Main Steam lines</u> (see Section 5.4)			
Number of steam lines	4	4	4
Design pressure (psig)	1250	1146	1250
Design temperature (°F)	575	563	575
Pipe diameter (in.)	26	24	24
Pipe material	Carbon steel	Carbon steel	Carbon steel

[a] *Parameters are related to rated power output for a single plant unless otherwise noted.*
[b] *See Section 1.3.1 regarding the status of the data presented here.*
[c] *Values correspond to original licensing.*
[d] *For Hatch, minimum critical heat flux ratio (MCHFR) was used.*
[e] *Free-standing loaded tubes.*
[f] *Channels of monitors from LPRM detectors.*
[g] *Pump and discharge piping to and including discharge block valve.*
[h] *Discharge piping from discharge block valve to vessel.*

Table 1.3-2

Comparison of Power Conversion System Design Characteristics

	CGS BWR 5 251-764	HATCH I[a] BWR 4 218-560	ZIMMER[a] BWR 5 218-560
Turbine Generator (see Sections 10.2 and 10.4)			
Rated power (MWt)	3468[b]	2550	2550
Rated power (MWe) (gross)	1205[b]	813	883
Generator Speed (rpm)	1800	1800	1800
Rated steam flow (1b/hr)	15.018 x 10^{6}[b]	10.48 x 10^6	11.0 x 10^6
Inlet pressure (psia)	955	950	950
Steam Bypass System (see Section 10.4.4)			
Capacity (% design steam flow)	25	25	25
Main Condenser (see Section 10.4.1)			
Heat removal capacity (Btu/hr)	7702 x 10^6	5720 x 10^6	7053 x 10^6
Circulating Water System (see Section 10.4.5)			
Number of pumps	3	2	3
Flow rate (gpm/pump)	186,000	185,000	150,000
Condensate and Feedwater System (see Section 10.4.7)			
Design flow rate (1b/hr)	14.26 x 10^6	10.096 x 10^6	10.971 x 10^6
Number of condensate pumps	3	3	3
Number of condensate booster pumps	3	3	3
Number of feedwater pumps	2	2	2
Number of feedwater booster pumps	None	None	None
Condensate pump drive	ac power	ac power	ac power
Booster pump drive	ac power	ac power	ac power
Feedwater pump drive	Turbine	Turbine	Turbine

[a] Values correspond to original licensing.
[b] Maximum calculated value.

Table 1.3-3

Comparison of Engineered Safety Features Design Characteristics

	CGS BWR 5 251-764	HATCH I BWR 4 218-560	ZIMMER BWR 5 218-560
Emergency Core Cooling Systems (systems sized on design power) (see Section 6.3)			
Low pressure core spray systems			
Number of loops	1	2	1
Flow rate (gpm)	6350 at 128 psid	4625 at 120 psid	4725 at 119 psid
High pressure core spray system			
Number of loops	1	1[a]	1
Flow rate (gpm)	1550 at 1130 psid 6350 at 200 psid	4250	1330 at 1110 psid 4725 at 200 psid
Automatic depressurization system			
Number of relief valves	7	7	7
Low pressure coolant injection[b]			
Number of loops	3	2	3
Number of pumps	3	4	3
Flow rate (gpm/pump)	7450 at 26 psid	7700 at 20 psid	5050 at 20 psid
Residual Heat Removal System (see Section 5.4.7)			
Reactor shutdown cooling mode:			
Number of loops	2	2	2
Number of pumps	2	4	2
Flow rate (gpm/pump)[c]	7450	7700	5050
Duty (Btu/hr/heat exchanger)[d]	41.6×10^6	32×10^6	30.8×10^6
Number of heat exchangers	2	2	2
Primary containment cooling mode:			
Flow rate (gpm)	7450[e]	30,800	5050[e]

Table 1.3-3

*Comparison of Engineered Safety Features
Design Characteristics (Continued)*

	CGS BWR 5 251-764	HATCH I BWR 4 218-560	ZIMMER BWR 5 218-560
Standby Service Water System (see Section 9.2.7)			
Flow rate (gpm/heat exchanger)	7400	8000	5000
Number of pumps	3[f]	4	4
Reactor Core Isolation Cooling System (see Section 5.4.6)			
Flow rate (gpm)	600 at 1150 psid	400 at 1120 psid	400 at 1120 psid
Fuel Pool Cooling and Cleanup System (see Section 9.1.3)			
Capacity (Btu/hr)	8.0×10^6	5.7×10^6	6.6×10^6

[a] High-pressure coolant injection system utilized.
[b] A mode of RHR system.
[c] Capacity during reactor flooding mode with more than one pump running.
[d] Heat exchanger duty at 20 hr following reactor shutdown.
[e] Flow per heat exchanger.
[f] Includes HPCS service water pumps.

Table 1.3-4

Comparison of Containment Design Characteristics

	CGS BWR 5 251-764	HATCH 1 BWR 4 218-560	ZIMMER BWR 5 218-560
Primary Containment[a] (see Sections 3.8.2 and 6.2.2)			
Type	Over and under pressure suppression	Pressure suppression	Over and under pressure suppression
Construction	Steel-free standing	Steel-free standing	Concrete pre-stressed with steel liner
Drywell	Frustum of cone upper portion	Light bulb/steel vessel	Frustum of cone upper portion
Pressure-suppression chamber	Cylindrical lower portion with eliptical bottom	Torus/steel vessel	Cylindrical lower portion
Pressure-suppression chamber internal design pressure (psig)	45	56	45
Pressure-suppression chamber external design pressure (psi)	2	2	2
Drywell internal design pressure (psig)	45	56	45
Drywell external design pressure (psi)	2	2	2
Drywell free volume (ft^3)	200,540[b]	146,240	180,000
Pressure-suppression chamber free volume (ft^3)	144,184 max	110,950	93,000
Pressure-suppression pool water volume (ft^3)	112,197 min[c]	87,300	102,000
Submergence of downcomer vent pipe below pressure pool surface (ft)	12 max. 11.67 min.	3.67	10
Design temperature of drywell (°F)	340	281	340
Design temperature of pressure-suppression chamber (°F)	275	281	275
Downcomer vent pipe pressure loss factor	1.9	6.21	2.17
Break area/total vent area	0.105	0.0194	0.008

Table 1.3-4

Comparison of Containment Design Characteristics (Continued)

	CGS BWR 5 251-764	HATCH 1 BWR 4 218-560	ZIMMER BWR 5 218-560
Primary Containment[a] (see Sections 3.8.2 and 6.2.2) (Continued)			
Calculated maximum pressure after blowdown to dwell (no pre-surge) (psig)	34.7	46.5	40.4
Pressure-suppression chamber (psig)	27.6	28	35.6
Initial pressure-suppression pool temperature rise (°F)	35	50	35
Leakage rate (% free volume/day at 45 psig and 200°F)	0.5	1.2 at 59 psig	0.635
Secondary Containment (see Sections 3.8.4 and 6.2.3)			
Type	Controlled leakage, elevated release	Controlled leakage, elevated release	Controlled leakage, elevated release
Construction			
Lower levels	Reinforced concrete	Reinforced concrete	Reinforced concrete
Upper levels	Steel super-structure and siding	Steel super-structure and siding	Steel super-structure and siding
Roof	Steel decking	Steel decking	Steel decking
Internal negative design pressure (in. H_2O)	0.25	0.25	0.25
Design inleakage rate (% free volume/day at 0.25 in. H_2O)	100	100	100

[a] Where applicable, containment parameters are based on design power.
[b] Maximum water level in suppression pool.
[c] Does not include the water within the reactor pedestal (10,065 ft^3) or the 12 ft of water below the downcomer vent pipe exits (15,000 ft^3).

Table 1.3-5

Radioactive Waste Management Systems
Design Characteristics

	CGS BWR 5 251-764	HATCH 1 BWR 4 218-560	ZIMMER BWR 5 218-560
Gaseous Radwaste (see Section 11.3)			
Design Bases (noble gases μCi/sec)	*100,000 at 30 minutes*	*100,000 at 30 minutes*	*100,000 at 30 minutes*
Process treatment	*Low temperature charcoal*	*Recombiner ambient charcoal*	*Chilled charcoal*
Number of beds	*8*	*12*	*5*
Design condenser in-leakage (cfm)	*30*	*40*	*12.5*
Release point - height above ground (ft)	*230*	*394*	*172*
Liquid Radwaste (see Section 11.2)			
Treatment of			
1. Floor drains[a]	*F, D, and R*	*F, D, and R*	*F, E, and R*
2. Equipment drains[a]	*F, D, and R*	*F, D, and R*	*F, D, and R*
3. Chemical drains[a]	*Neutralized, E, D, and R*	*F, discharged E, solid to radwaste*	*E, D, concentrates to solid radwaste distillate R*
4. Detergent drains[a]	*Chemical addition, F, E, and sent to circulating water discharge[b]*	*Diluted and sent to circulating water discharge*	*Reverse osmosis discharge*
5. Expected annual average release (μCi) (excluding tritium)	*170*	*2000*	*1.09*

[a] *Legend:*
D = demineralized.
F = filtered.
E = evaporator/concentrator.
R = recycled, i.e., returned to condensate storage.

[b] *Laundry will be processed offsite by authorized contractor.*

Table 1.3-6

Comparison of Structural Design Characteristics

	CGS BWR 5 251-764	HATCH 1 BWR 4 218-560	ZIMMER BWR 5 218-560
Seismic Design (see Section 3.7)			
Operating basis earthquake (horizontal g)	0.125	0.08	0.10
Safe shutdown earthquake (horizontal g)	0.250	0.15	0.20
Wind Design (see Section 3.3)			
Maximum sustained (mph)	100	105	90
Tornados			
Translational (mph)	60	60	60
Tangential (mph)	300	300	300

Table 1.3-7

Comparison of Electrical Systems Design Characteristics

	CGS[a] BWR 5 251-764	HATCH 1 BWR 4 218-560	ZIMMER BWR 5 218-560
<u>Transmission System</u> (see Section 8.2)			
Outgoing lines (number - rating)	1 - 500 kV	4 - 230 kV	3 - 345 kV
Normal auxillary ac power			
Incoming lines (number - rating)	1 - 230 kV 1 - 115 kV	4 - 230 kV	1 - 69 kV 1 - 345 kV
Normal auxiliary transformers	2	2	1 (unit auxiliary)
Startup/backup auxiliary transformers	2	2	2
Standby ac power supply			
Number of diesel generators	3[b]	3[c]	3
Number of 4160-V shutdown (Class 1E) buses	3[b]	3	3
Number of 480-V shutdown (Class 1E) buses	5[b]	2 (600 V)	5
<u>Power Supply (dc)</u> (see Section 8.3.2)			
Number of 24-V batteries	4	2 (48 V)	
Number of 125-V batteries	6[d]	3	3
Number of 250-V batteries	1	2	1
Number of 24-V buses	2	2 (24/48 V)	
Number of 125-V buses	6[d]	3	3
Number of 250-V buses	1	2	1

[a] Does not include 450-V dc security system.
[b] HPCS system included.
[c] Total of five for two units.
[d] HPCS battery and bus included.

LDCN-99-000

Table 1.3-8

Significant Design Changes from PSAR to FSAR

Item	*Change*	*Reason for Change*	*FSAR Portion in Which Change is Discussed*
Offgas system class change	*The offgas system components are Quality Group C, whereas the system components were described in the PSAR as being Quality Group D.*	*Improve assurance of system integrity.*	*11.3.1*
Control rod drive position indication	*Changed to 11 wire probe and solid state.*	*Improved reliability and increased frequency of checking actual rod position.*	*7.7.1*
Control rod drive system	*Deleted CRD return line and pump test bypass, revised cooling and exhaust water headers, added relief valves interconnecting cooling water and exhaust headers, redirected system exhaust flow through the multiple solenoid valves in each HCU.*	*GE recommendation.*	*4.6.1.1.2.4*
Recirculation pump and motor	*The flow rate and horsepower required has been reduced; voltage has changed from 4160 V to 6600 V. A low-frequency motor generator set was added to provide 25% speed.*	*Detailed system.*	*5.4.1*
Jet pumps	*The jet pump design was changed to improve five-hole type.*	*Design improvement, increased efficiency.*	-
Recirculation flow measurement	*The recirculation flow measurement design was changed from a flow element to an elbow-tap type.*	*To improve flow measurement accuracy.*	*7.3.1*
Recirculation system	*The pressure interlock for RHR injection was changed.*	*IEEE-279 requirements.*	*7.3.1, 7.6.1*
Recirculation system	*Bypass line around reactor recirculation system flow control valve was eliminated.*	*Reduce the possibility of cavitation and cracking of piping in the recirculation system. Need eliminated by addition of low frequency motor generator set.*	

Table 1.3-8

Significant Design Changes from PSAR to FSAR (Continued)

Item	Change	Reason for Change	FSAR Portion in Which Change is Discussed
Nuclear fuel	The number of fuel pins in each fuel bundle has been changed from 7 x 7 to 8 x 8 (including two water rods).	Improved fuel performance by increasing safety margins.	4.2
Nuclear boiler	A turbine building high temperature trip for MSIVs was added.	Improve leak detection capability.	7.3.1
Nuclear boiler	An additional test mode was added for closing MSIVs one at a time to 90% of full open in the fast mode (close in slow mode already existed).	Verifies that the spring force on the valves will cause them to close under loss-of-air conditions.	5.4.5
Main steam line isolation	A main condenser low vacuum initiation of the main steam line isolation was added.	NRC requirement.	7.3.1
Main steam line isolation	Reactor isolation was deleted for reactor high water level.	To provide improved plant availability.	5.4.5
Main steam line drain system	A main steam line drain system was improved.	Prevent accumulation of condensate in an idle line outboard of MSIV.	5.1.1
RPV code	The RPV code was updated to ASME 1971 and Summer 1971 addenda.	Update to applicable code as much as possible.	5.2.1
Level instrumentation	The RPV level instrumentation was revised to eliminate Yarway columns and replace them with a conventional condensing chamber type; also, separation and redundancy features were added.	Improve ECCS separation per IEEE-279 and improve reliability.	7.3.1
Turbine seal setpoint pressure	The turbine seal setpoint pressure was changed from 50 psia to 125 psia.	Ensures that main turbine condenser can extract reactor steam at temperature above cooling capability of RWCU system.	-
Leak detection system	The leak detection system was revised to upgrade the capability and incorporate the requirements of IEEE-279.	To meet IEEE-279 requirements.	7.6.1

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Table 1.3-8

Significant Design Changes from PSAR to FSAR (Continued)

Item	*Change*	*Reason for Change*	*FSAR Portion in Which Change is Discussed*
Reactor vibration monitoring	*A confirmatory vibration monitoring test was added.*	*NRC requirement.*	*14.2.12.3.34*
RWCU system sample station	*The P&IDs were changed to provide continuous monitoring.*	*Technical Specifications requirements.*	-
LPCS system	*Valve F011 was changed from air-operated to motor-operated control.*	*To provide Seismic Category I rated control power to this essential active component.*	*7.3.1.1.1.3*
LPCS system	*Direct connection to condensate storage replaced by removable spool piece connection to RHR.*	*Condensate used only for system commissioning tests.*	*Figure 6.3-5*
PRT replaced by RPT	*Prompt relief trip (PRT) was replaced by recirculation pump trip (RPT) for quick insertion of negative reactivity.*	*Increased reliability.*[a]	*7.6.1.5*
Main steam system	*Relief valve augmented bypass (REVAB) was deleted.*	*Licensing requirement.*[a]	-
Feedwater sparger	*The thermal sleeve was changed to provide welded design of sparger to nozzle.*	*To eliminate vibration, failure, and leakage.*	*5.3*
Standby liquid control (SLC) system	*Interlocks on the SLC system were revised.*	*To prevent inadvertent boron injection during system testing.*	*7.4.1, 9.3.5*
Standby liquid control (SLC) system	*RCPB extended to explosive valves*	*To meet isolation criteria.*	-
RClC steam supply	*A warmup bypass line and valve was added.*	*Permits pressurizing and prewarming of the steam supply line downstream to the turbine during reactor vessel heatup.*	*5.4.6*
RCIC vacuum breaker system	*A vacuum breaker system was added to the RCIC turbine exhaust line into the suppression pool.*	*To prevent backup of water in the pipe and consequential high dynamic pipe loads and reactions.*	*5.4.6*

Table 1.3-8

Significant Design Changes from PSAR to FSAR (Continued)

Item	Change	Reason for Change	FSAR Portion in Which Change is Discussed
RCIC system	Each component has been made capable of functional testing during normal plant operation.	Improved testability.	5.4.6
Automatic depressurization system (ADS)	The interlocks on the automatic depressurization system were revised.	To meet IEEE-279 requirements.	7.3.1
RPV support	The support for the RPV was changed from a ring girder to a bearing plate.	Provides a better seismic and alignment capability.	5.3.3.1.4.1
Plant service water pumps	Upon loss of offsite power without a LOCA, the normal 4160 V service buses (SM-75, SM-85), are connected to SM-7 and SM-8 to provide automatic starting of a plant service water pump for drywell cooling.	Provides service water for drywell cooling automatically after loss of offsite power without a LOCA.	Figure 8.1-2, Tables 8.3-1 and 8.3-2
Reactor building cooling system	ESF cooling units have been added to critical electric equipment areas in the reactor building.	To provide suitable ambient temperature conditions for essential electrical and control equipment located in the reactor building in the event of a LOCA.	9.4.9
Standby gas treatment system	Added second fan (powered from alternate power bus) to each standby gas treatment system.	To remove need for crosstie between the two systems.	6.5.1.2
Standby gas treatment system	Added facility to recirculate air from SGTS back into reactor building.	So that potential decay heat in filter can be removed without discharge to atmosphere in event of divisional power failure.	6.5.1.2
Standby gas treatment system	Added second electric preheater (powered from alternate power bus) to each SGTS unit.	To provide means of controlling relative humidity of air entering charcoal filter in event of primary heater or divisional power failure.	6.5.1.2

Table 1.3-8

Significant Design Changes from PSAR to FSAR (Continued)

Item	*Change*	*Reason for Change*	*FSAR Portion in Which Change is Discussed*
Control room HVAC system	*Added two remote air intakes for pressurizing control room in event of a LOCA.*	*To limit doses to operating personnel to limits of 10 CFR 50.*	*9.4.1.2*
Ventilation system for areas in which essential cable is routed	*Added to ESF ventilation system to ventilate corridors and cable chases through which essential cable is routed (diesel generators to control room).*	*To provide suitable ambient temperatures for essential cable in the event of a LOCA*	*9.4.8*
Offgas system charcoal vault	*Added a refrigeration system to the vault in which the offgas system charcoal adsorber filters are housed.*	*To maintain charcoal adsorbers at a temperature of 0°F.*	*9.4.5, 11.3.2.1*
Makeup water pumps transformer vault ventilation	*Added a ventilation system to makeup water pump transformer rooms powered from the emergency buses.*	*To ensure suitable ambient temperatures for transformers in the event of a loss of offsite power caused by a tornado.*	*9.4.6*
Radioactive waste solidification process	*Cement-sodium silicate solidification process to be used in lieu of urea-formaldehyde process.*	*To eliminate the generation of free water in solidified containers, a problem inherent to the urea-formaldehyde process.*	*11.4*
Air ejector	*Three-stage air ejector to two-stage air ejector.*	*Manufacturer offered a two-stage unit that meets the same operating conditions.*	*10.4.2*
Sealing steam supply	*The gland steam evaporator will produce sealing steam using main steam on its tube side during startup and shutdown modes. PSAR stated auxiliary boiler would be used.*	*Adequate sealing steam can be produced with main steam pressure down to 125 psig.*	*10.4.3*
Containment instrument air	*The CIA air compressors were removed and the system is now supplied with nitrogen during reactor operation. Redundant bottled gas supply utilized for supplying ADS valve accumulators for accident conditions.*	*The purpose of the safety related bottled gas supplies is to back up the non-safety-related cryogenic nitrogen supply.*	*9.3.1.1.2*

Table 1.3-8

Significant Design Changes from PSAR to FSAR (Continued)

Item	*Change*	*Reason for Change*	*FSAR Portion in Which Change is Discussed*
Offgas holdup line	*Radiography of circumferential welds was not done.*	*A partial section of the line was buried before radiography was done. Welds were magnetic particle tested and line was hydro-tested at 1200 psig and then helium pressure decay leak tested with a sensitivity of 10^{-2} cm^3/sec.*	-
Wet solid radwastes	*Packaged in 50 ft^3 containers rather than 50-gal drums.*	*Reduce handling time and operator exposure.*	*11.4.2.10*
Turbine bypass valve system	*Four bypass valves are used rather than three.*	*Solution to operating problems with bypass valves on Cooper Nuclear station.*	*10.4.4*
Main steam isolation valve leakage control system	*Added to plant.*	*NRC requirement.*	*6.7*
Main steam line from outermost isolation valve to turbine stop valve	*Piping has been upgraded from Code Group D to Code Group B.*	*NRC requirement.*	*10.3.2*
Radwaste tank sizes			
1. Waste sludge phase separator	*From 12,500 to 13,000 gal.*	*To increase capacity.*	*Table 11.4-4*
2. Chemical waste tank	*From 13,000 to 15,000 gal.*	*To increase capacity.*	*Table 11.2-13*
3. Decontamination solution concentrated waste tank	*From single 700-gal to two 700-gal tanks.*	*To provide spare tank.*	*Table 11.4-4*

Table 1.3-8

Significant Design Changes from PSAR to FSAR (Continued)

Item	Change	Reason for Change	FSAR Portion in Which Change is Discussed
4. Concentrated waste measuring tank	From 100 to 400 gal.	Due to increase in shipment container size from 50 gal to 50 ft^3.	Table 11.4-4
5. Condensate phase separators	From 12,500 to 23,500 gal.	To increase capacity in event of higher than normal backwash requirements.	Table 11.4-4
6. Chemical addition tank	From single 1000-gal tank to two 200-gal tanks.	To provide capability for both acid and caustic addition from separate tanks. Original tank oversized.	Table 11.2-13
Floor drain system	Influent waste radionuclide concentration changed from range of 10^{-4} to 10^{-2} μCi/ml to on order of 10^{-1} μCi/ml.	Reevaluation of source terms.	11.2.2.2.2
Liquid radwaste system	GALE code was used to calculate radioactive discharges with 2500-gpm blowdown. Blowdown of 4000 gpm was used in the PSAR.	NRC requirement to use GALE Code. Change in blowdown results in more conservative (higher) radionuclide concentrations.	11.2.3.2
Cleaning of filters	Changed from steam cleaning connections to chemical cleaning system.	Design improvement.	Figure 10.4-5
Missiles from tornadoes	Selection of credible missiles.	For FSAR, followed specific missiles identified in NRC Standard Review Plan.	3.5.1.4
Cleaning of filters	Changed from steam cleaning connections to chemical cleaning system.	Design improvement.	Figure 10.4-5
Missiles from tornadoes	Selection of credible missiles.	For FSAR, followed specific missiles identified in NRC Standard Review Plan.	3.5.1.4
Primary containment vessel	New loads due to hydro-dynamic effects of safety/relief valve actuation and LOCA (neither in PSAR or FSAR; see Dynamic Analysis Report).	To accommodate new GE load requirements.	3.8.2

Table 1.3-8

Significant Design Changes from PSAR to FSAR (Continued)

Item	Change	Reason for Change	FSAR Portion in Which Change is Discussed
Diesel generator building fire protection system	Changed from CO_2 system to dry pipe preaction system. after a fire.	To provide accessibility to the diesel immediately. Also availability of unlimited water supply	Appendix F
Cable chase fire protection system	Added dry pipe preaction system for cable chase and diesel generator building corridor.	To protect divisional cable concentrations in these areas.	Appendix F
500-kV line	Hookstick changed to motor-operated switch.	Available standard switches are supplied with motor operators.	Fig. 8.1-2
500-kV line	Line terminates at H. J. Ashe Swtichyard rather than Hanford Switching Station.	BPA revisions to 500 kV grid.	8.1.2
230-kV line	Deleted hookstick and 230-kV OCB at plant switchyard.	OCB relocated to H. J. Ashe Switchyard.	Fig. 8.1-2
115-kV line	Replace circuit interrupter with 115-kV OCB at plant switchyard.	Equipment availability.	Fig. 8.1-2
Backup source	Utilized to supply essential loads during diesel generator testing.	PSAR did not consider particulars of diesel generator testing.	8.3.1.1.7.1.7
Diesel generator starting	Deleted automatic starting due to startup or backup transformer undervoltage.	Class 1E bus undervoltage is the only undervoltage condition requiring diesel generator start	8.3.1.1.7.1.7 8.3.1.1.7.2.7
Diesel generator trips during emergency operation	Added incomplete sequence trip to Division 1 and 2 diesel generators.	Incomplete sequence indicates a diesel generator malfunction having an imminent possibility of unit damage.	8.3.1.1.7.1.8
125-V, 250-V-dc battery capability	Revised supply capability from 4 hr to 2 hr.	Increased dc load	8.3.2
125-V, 250-V-dc charger capability	Revised recharge capability from 8 hr to 24 hr.	Increased dc load	8.3.2

Table 1.3-8

Significant Design Changes from PSAR to FSAR (Continued)

Item	Change	Reason for Change	FSAR Portion in Which Change is Discussed
Spare 125-V-dc charger	*Spare charger serves as a backup for Divisions 1 and 2 only.*	*Spare charger is too large to provide backup to Division 3.*	*8.3.2*
Communication systems	*The commercial telephone exchange system is not redundant.*	*Redundancy not required.*	*8.2.1.5*
Fuel pool cooling and cleanup system	*Upgraded cooling portion of system to Seismic Category I to provide long-term cooling and safety grade makeup water capability for coolant of spent fuel following refueling.*	*To prevent fuel pool boiling and resultant adverse environmental conditions which could affect safety-related electrical equipment in the reactor building.*	*9.1.3*

[a] PRT and REVAB were proposed at the CP stage as non-safety-related power generation type systems to reduce the thermal-hydraulic effects of transient events in the core. However, during experiments in the MK-11 suppression pool dynamics test program, it was decided that less frequent relief valve cycling during plant operation was desirable. Consequently, the recirculation pump trip (RPT) system was developed to perform functions previously assigned to PRT and REVAB.

1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

The italicized information is historical and was provided to support the application for an operating license.

1.4.1 APPLICANT/OPERATOR

Energy Northwest is a municipal corporation and a joint operating agency of the State of Washington, organized in January 1957, pursuant to Chapter 43.52 of the Revised Code of Washington, as amended. Energy Northwest assumes the responsibility for safe operation and maintenance of the plant and for providing related services as described in Chapter 13.

1.4.2 ENGINEER AND CONSTRUCTION MANAGEMENT - BURNS & ROE, INC.

Burns and Roe, Inc. (B&R) provides engineering and initial construction management and quality assurance services for the design and construction of the plant, integrating the major plant items furnished by the General Electric Company (GE) and Westinghouse Electric Corporation.

Burns & Roe was founded in 1932 and incorporated in 1935 as Burns and Roe, Inc. Burns & Roe has been active in the fields of power generation and distribution, sea water and brackish water desalination, waste water renovation, and engineering, design, and/or construction management services for over 50 thermal power generating units representing more than 11,400,000 kW of new generating capacity, of which more than 4,800,000 kW is nuclear. Burns & Roe, Inc., has been continuously engaged in construction of engineering activities since 1935.

1.4.3 NUCLEAR STEAM SYSTEM SUPPLIER - GENERAL ELECTRIC COMPANY

General Electric designed, fabricated, and delivered the direct-cycle boiling water nuclear steam supply system (NSSS) for Columbia Generating Station (CGS). General Electric also fabricated the first core of nuclear fuel and provided technical direction of installation and startup of this equipment.

General Electric has engaged in the development, design, construction, and operation of boiling water reactors (BWRs) since 1955. Table 1.4-1 lists GE reactors completed, under construction, or ordered (several later canceled). Thus, GE has substantial experience, knowledge, and capability to design, manufacture, and furnish technical assistance for the installation and startup of reactors.

General Electric continues to provide technical support for the operation of CGS as requested by Energy Northwest. This includes providing support for the CGS Megawatt Improvement Program (see Section 1.1).

1.4.4 TURBINE GENERATOR SUPPLIER - WESTINGHOUSE ELECTRIC CORP.

Westinghouse Electric Corporation designed, fabricated, delivered, and installed the turbine generator for CGS. They also provided technical assistance for the startup of this equipment.

Westinghouse Electric Corporation has a long history in the application of turbine generators in nuclear power stations going back to the inception of commercial electrical power production using nuclear facilities. Westinghouse furnished the turbine generator unit for Shippingport No. 1. This unit was shipped in 1956. Westinghouse also furnished the turbine generator unit for Yankee Atomic Power Company Rowe No. 1. This unit was shipped in 1959. San Onofre No. 1 and Connecticut Yankee, Haddam Neck No. 1 unit went into commercial operation in 1968. Westinghouse nuclear turbine generators produced over 300 billion kW hr of electricity through May 1976, when 25 nuclear turbine generators totaling over 16,500 MW were in service. By 1984, 75 Westinghouse nuclear turbine generators should be in service producing over 61,319 MW. Inlet steam pressures of these units vary between 750 psig and 1000 psig and electrical outputs vary from 500,000 kW to 1,090,000 kW.

Westinghouse continues to provide technical and maintenance support for the turbine generator on an as-requested basis. They also provided replacement for the three low-pressure turbine rotors installed in the Spring 1992 refueling outage.

1.4.5 SYSTEM COMPLETION CONTRACTOR - BECHTEL

As System Completion Contractor, Bechtel provides field and home office services in project planning and control, engineering, construction completion, startup support, and quality verification for CGS. The Bechtel organization was founded in 1898, in the midwest, by Warren A. Bechtel. In 1940, Bechtel went international, working on a pipeline system in Venezuela; and then vastly diversified its activities during World War II, becoming involved in naval bases, shipyards, pipelines, refineries, and aircraft modification. Next, Bechtel pioneered in the nuclear power field, constructing the first reactor to produce useful electricity in 1949, and building Dresden I, the first commercial nuclear power station. Today, Bechtel is recognized as one of the world's leading engineering and construction firms.

1.4.6 MAJOR CONTRACTORS

1.4.6.1 Fischbach/Lord

Fischbach/Lord is responsible for the major electrical installation at CGS, consisting of raceways, conduit, cable, terminators, and electrical equipment. They were formed as a joint venture, solely for this project, in 1974.

1.4.6.2 Pittsburgh-Des Moines Steel Company

Pittsburgh-Des Moines Steel Company is responsible for engineering, fabrication, and installation of materials in the Primary Containment Vessel.

1.4.6.3 Wright-Schuchart- Harbor/Boecon (Boeing Construction)/General Energy Resources, Inc.

Wright-Schuchart-Harbor/Boecon/General Energy Resources, Inc. (WBG) was formed as a joint venture October 1, 1977, to be responsible for installation of major mechanical equipment, power, and process piping for CGS.

1.4.6.4 Bechtel

During plant construction, Bechtel served as the Construction Manager. During the operating phase Bechtel, as the Site Support Services contractor, is providing field engineering and installation support for plant modifications. Also, as Technical Services contractor, they are providing engineering support under Energy Northwest direction and under the Energy Northwest quality assurance program as requested by Energy Northwest. Under these contracts Bechtel is providing support to the Megawatt Improvement Program (see Section 1.1).

1.4.6.5 AREVA NP

The initial fuel core was fabricated by GE. Reload fuel is being provided by AREVA NP. Their contract provides for the supply of uranium concentrates and fuel fabrication services. Other fuel in the core was provided by Westinghouse (ABB/Combustion Engineering).

1.4.6.6 Westinghouse Electric

Westinghouse provided the turbine generator. They provided replacement of the three low-pressure rotors which were installed in 1992. Westinghouse also provided a new plant simulator which was installed in 1995.

1.4.7 CONSULTING ENGINEER - R. W. BECK AND ASSOCIATES

The independent consulting firm of R. W. Beck and Associates is the consulting engineer for Energy Northwest's Columbia Generating Station. This firm was also a consulting engineer for WNP-1. Having extensive experience in preparing engineering feasibility and financing studies and reports necessary for the success of utility and civic improvement projects, the firm is well qualified for employment as a consulting engineer and was chosen as a result of its experience.

The duties of the consulting engineer are briefly summarized as follows: prepare estimates of plant capability, energy potential, usability within area loads and resources, the cost of power and energy output of the project, and generally determine the feasibility of the project. These duties will include assisting in preparation of a Bond Resolution, preparation of an engineering report, schedules for investment of funds, schedules for debt service payments, and other engineering services necessary to facilitate the financing of the project.

Table 1.4-1

Commercial Nuclear Reactors Completed, Under Construction, or in Design by General Electric

Station	Utility	Rating (MWe)	Year of Order	Year of Startup
Dresden 1 [a]	Commonwealth Edison	207	1955	1960
Humboldt Bay [a]	Pacific G&E	63	1958	1963
Kah1 [a]	Germany	15	1958	1961
Garigliano [a]	Italy	150	1959	1964
Big Rock Point	Consumers Power	71	1959	1965
JPDR	Japan	11	1960	1963
KRB [a]	Germany	237	1962	1967
Tarapur 1	India	190	1962	1969
Tarapur 2	India	190	1962	1969
GKN	Holland	52	1963	1968
Oyster Creek	JCP&L	620	1963	1969
Nine Mile Point 1	Niagara Mohawk	610	1963	1969
Dresden 2	Commonwealth Edison	794	1965	1970
Pilgrim 1	Boston Edison	655	1965	1972
Millstone 1	NUSCo	660	1965	1970
Tsuruga	Japan	340	1965	1970
Nuclenor	Spain	440	1965	1971
Fukushima 1	Japan	439	1966	1971
BKW KKM	Switzerland	306	1966	1972
Dresden 3	Commonwealth Edison	794	1966	1971
Monticello	Northern States	536	1966	1971
Quad Cities 1	Commonwealth Edison	789	1966	1972
Browns Ferry 1	TVA	1065	1966	1974
Browns Ferry 2	TVA	1065	1966	1975
Quad Cities 2	Commonwealth Edison	789	1966	1972
Vermont Yankee	Vermont Yankee	514	1966	1972
Peach Bottom 2	Philadelphia Electric	1065	1966	1974
Peach Bottom 3	Philadelphia Electric	1065	1966	1974
James A. FitzPatrick	New York Power Authority	821	1966	1975
Bailly [b]	NIPSCo	660	1966	----
Shoreham [b]	LILCo	819	1967	1985
Cooper	Nebraska PPD	778	1967	1974
Brown Ferry 3	TVA	1065	1967	1977
Limerick 1	Philadelphia Electric	1055	1969	1985
Hatch 1	Georgia	786	1967	1975
Fukashima 2	Japan	762	1967	1974
Brunswick 1	Carolina P&L	790	1968	1977
Brunswick 2	Carolina P&L	790	1968	1975
Arnold	Iowa ELP	545	1968	1975
Fermi 2	Detroit Edison	1056	1968	1984
Limerick 2	Philadelphia Electric	1055	1969	----

Table 1.4-1

Commercial Nuclear Reactors Completed, Under Construction, or in Design by General Electric (Continued)

Station	Utility	Rating (MWe)	Year of Order	Year of Startup
Hope Creek 1	PSE&G	1067	1969	1986
Hope Creek 2[b]	PSE&G	1067	1969	----
Zimmer[b]	CCDPP	810	1969	----
Chinshan	Taiwan	610	1969	1977
Caorso 1	Italy	827	1969	1975
Hatch 2	Georgia	795	1970	1979
LaSalle County 1	Commonwealth Edison	1078	1970	1983
LaSalle County 2	Commonwealth Edison	1078	1970	1984
Susquehanna 1	Pennsylvania P&L	1050	1968	1983
Susquehanna 2	Pennsylvania P&L	1050	1968	1984
Chinshan 2	Taiwan	610	1970	1978
Columbia Generating Station	Energy Northwest	1103	1971	1984
Nine Mile Point 2	Niagara Mohawk	1090	1971	1986
Grand Gulf 1	Midsouth	1250	1972	1985
Kaiseraugst[b]	Switzerland	915	1971	----
Fukushima	Japan	1135	1971	1976
Takai 2	Japan	1135	1971	1976
River Bend 1	Gulf States	940	1971	1985
River Bend 2[b]	Gulf States	940	1971	----
Perry 1	Cleveland Electric	1205	1971	1985
Perry 2[b]	Cleveland Electric	1205	1971	----
Hartsville A-1[b]	TVA	1233	1972	----
Hartsville B-1[b]	TVA	1233	1972	----
Hartsville A-2[b]	TVA	1233	1972	----
Hartsville B-2[b]	TVA	1233	1972	----
Laguna Verde 1	Mexico	660	1972	1977
Leibstadt	Switzerland	940	1972	1978
Kuosheng 1	Taiwan	992	1972	1978
Kuosheng 2	Taiwan	992	1972	1979
Clinton 1	Illinois Power	950	1973	1986
Clinton 2[b]	Illinois Power	950	1973	----
Montague 1[b]	NUSCO	1150	1973	----
Allens Creek 1[b]	Houston L&P	1200	1973	----
Skagit 1[b]	Puget SD	1288	1973	----
Skagit 2[b]	Puget SD	1288	1973	----
Barton 3[b]	Alabama	1220	1973	----
Blackfox 1[b]	Oklahoma	1150	1973	----
Blackfox 2[b]	Oklahoma	1150	1973	----
Cofrentes	Spain	975	1973	1977
Laguna Verde 2	Mexico	660	1973	1978
Enel 6[b]	Italy	982	1974	----
Enel 8[b]	Italy	982	1974	----

[a] *Retired*
[b] *Discontinued*

1.5 REQUIREMENTS FOR FURTHER TECHNICAL INFORMATION

The italicized information is historical and was provided to support the application for an operating license.

1.5.1 GENERIC ISSUES

NUREG-0933, "A Prioritization of Generic Safety Issues" presents the generic issues as follows:

a. TMI action plan items

In NUREG-0933, these follow the content and format of NUREG-0660 and NUREG-0737.

b. Task action plans

These include both the unresolved safety issues (USIs) previously included in NUREG-0606 and the Category A Generic Activities previously included in NUREG-0371 and the Category B, C and D Generic Activities previously included in NUREG-0471.

c. Human factors

These are the human factors considerations of NUREG-0660 and NUREG-0737.

d. Chernobyl Issues

This part addresses the recommendations of NUREG-1251.

In the sections below, these issues are addressed as unresolved safety issues (USIs), generic safety issues (GSIs), and TMI Task Action Plans. Human Factors considerations are included as part of the TMI Task Action Plans. Chernobyl is not addressed below or on the CGS docket as NUREG-1251 lead to the conclusion that no immediate changes in NRC regulations regarding the design or operation of U.S. commercial reactors were required. However, NUREG-1251 and INPO SER 34-86, "Chernobyl Unit 4 Accident," and INPO SOER 87-1, "Core Damaging Accident Following an Improperly Conducted Test," were reviewed by Energy Northwest to identify the need for any changes to hardware, procedures, or training at CGS.

1.5.1.1 Unresolved Safety Issues

1.5.1.1.1 Unresolved Safety Issues Introduction

Unresolved safety issues are issues identified by the NRC that affect a number of plants, question the adequacy of existing requirements, have no current resolution and are judged to be unacceptable if left unresolved for the life of the plant.

A December 20, 1977, amendment to the Energy Reorganization Act required that the NRC develop a plan providing for specification and analysis of USIs and take action as necessary to implement corrective measures with respect to such issues. In a joint Explanatory Statement of the House - Senate Conference Committee for the FY 1978 Appropriations Bill this was explained to mean that a plan was to be developed to resolve the USIs. In September 1989, the NRC achieved resolutions of all of the identified USIs.

On October 19, 1989, the NRC issued Generic Letter 89-21, "Request for Information Concerning Status of Implementation of Unresolved Safety Issue (USI) Requirements." This generic letter requested that licensees and construction permit holders review and report on the status of the implementation of USIs for which final technical resolution had been achieved.

Energy Northwest responded to this request in Reference 1.5-1. The NRC responded to this submittal by Reference 1.5-2 and identified anticipated transient without scram (ATWS), Station Blackout and Safety Implications of Control Systems (A-9, A-44, and A-47, respectively) as not being implemented. (Subsequently, these have been resolved as discussed below.)

1.5.1.1.2 Implementation of Specific Unresolved Safety Issues

A-8 Mark II Containment Pool Dynamic Loads

Resolution of A-8 for CGS is documented in NUREG-0892 (the SER for CGS) and Supplements 4 and 5 in Sections 6.2.1.8 and 3.9.3.1, respectively.

A-9 Anticipated Transients Without Scram

In the safety evaluation transmitted with Reference 1.5-7, the NRC stated that the standby liquid control (SLC) flow and sodium pentaborate decahydrate concentration for CGS were in compliance with the ATWS rule.

The design requirements for resolution of ATWS for CGS were to install an alternate rod injection (ARI) system (see Section 7.4.1.6), a standby liquid control (SLC) system (see Sections 7.4.1.2 and 9.3.5), and to trip the reactor recirculation pumps automatically by a recirculation pump trip (RPT) system under conditions indicative of an ATWS (Section

7.4.1.5). In addition, ATWS equipment needed to be qualified for the environmental conditions associated with anticipated operational occurrences and to ATWS conditions up to the time the required function is completed (Reference 1.5-10). The FSAR Section 15.8 ATWS analysis also needed to be revised.

In Reference 1.5-3, the NRC stated that the CGS alternate rod injection system was in compliance with the ATWS rule. The reference also stated that the RPT system required two modifications to be in compliance with the rule. Reference 1.5-4 documents the implementation of the changes required to resolve these two issues.

In Reference 1.5-5, Energy Northwest informed the NRC that confirmation of the environmental qualifications of ATWS equipment remained to be confirmed. Reference 1.5-6 documented that the confirmation had been completed.

In FSAR Amendment 42, Section 15.8 was revised to include new ATWS analyses. Technical Specification Amendment 93 was issued on August 9, 1991 which addressed modifications to the ATWS-RPT system. With this amendment, all activities required for ATWS resolution for CGS were completed.

<u>A-10 BWR Feedwater Nozzle Cracking</u>

NRC review of CGS relative to A-10 and NUREG-0619, which Generic Letter 89-21 states resolves this USI, is documented in NUREG-0892, Sections 3.9.3.1, 5.2.3.1, and 5.2.4. While these sections address A-10, they do not specifically state that the total issue is resolved for CGS. However, as no concerns were raised in the subsequent five supplements to NUREG-0892 and as Energy Northwest was not aware of a concern of the NRC's regarding A-10 subsequent to the issuance of the operating license, in Reference 1.5-1 Energy Northwest stated that it believed A-10 to be resolved for CGS. This position was apparently accepted by the NRC by the issuance of Reference 1.5-2.

<u>A-11 Reactor Vessel Material Toughness</u>

NRC acceptance of the CGS commitment to 10 CFR 50, Appendix G, is discussed in NUREG-0892, Section 5.3.2. In NUREG-0744 and Generic Letter 82-26 issued subsequent to the publication of the original issue of NUREG-0892, a response by licensees was not required; they only provided guidance to licensees who may have been required to submit a fracture analysis to justify continued operation. This was not the case for CGS.

<u>A-17 Systems Interactions</u>

Generic Letter 89-18 issued September 6, 1989 transmitted NRC final resolution of this USI. No formal reply was required. Energy Northwest incorporated information contained and referenced in this Generic Letter into the CGS IPE program, the results of which were *submitted to the NRC by Reference 1.5-22. However, as no formal action to Generic Letter 89-18 was required, Energy Northwest considered this USI closed for CGS prior to the completion of the IPE. This was so stated in Reference 1.5-1.*

A-24 Qualification of Class 1E Safety Related Equipment

In NUREG-0892 Supplement 4, Section 3.11.5, the NRC states that CGS has demonstrated conformance to NUREG-0588. Generic Letter 89-21 states that Revision 1 to NUREG-0588 resolved A-24. By NRC memorandum, J. Knight to T. Novak, dated November 1983 (8312120370), Mr. Knight states that the CGS review was to Revision 1 of the NUREG.

A-31 Residual Heat Removal Shutdown Requirements

NUREG-0892 states in Section 5.4.2.1 that the CGS RHR system conforms to the Commission's regulations and applicable Regulatory Guides. Generic Letter 89-21 states that A-31 was resolved in May 1978 by publication of SRP 5.4.7. As NUREG-0892 was written in May 1982, Energy Northwest stated in Reference 1.5-1 that this established closure of A-31 for CGS.

A-36 Control of Heavy Loads

NUREG-0892 Supplement 4, Section 9.1.5, states that the guidelines of NUREG-0612 have been satisfied for CGS. Generic Letter 89-21 states that NUREG-0612 resolves A-36.

A-39 Determination of Safety Relief Valve Pool Dynamic Load and Temperature Limits

Section 6.2.1.8 of NUREG-0892 Supplements 1 and 4, provides NRC acceptance of the resolution of this issue for CGS.

A-40 Seismic Design Criteria

NUREG-1233 issued September 1989 states that the proposed changes that constitute the resolution of USI-40 are to apply to new applicants only. CGS is not one of the plants identified in Generic Letter 89-21 that needed to be reviewed to the new criteria.

A-42 Pipe Cracks in Boiling Water Reactors

NUREG-0892 states in Section 5.2.3.1 that CGS conforms to the requirements of NUREG-0313, Revision 1, which Generic Letter 89-21 states resolves A-42. NUREG-0892 Supplement 5, Section 5.2.3.2, provides additional information on this issue. Also see Section 5.2.3.2.3. Additional consideration for BWR pipe cracks beyond the scope of A-42 were raised by the NRC in Generic Letter 88-01. The resolution of Generic Letter 88-01 for CGS *is provided in References 1.5-21, 1.5-35, and 1.5-36, and in the Bases for CGS Technical Specifications.*

A-43 Containment Emergency Sump Performance

Generic Letter 89-21 states that resolution of A-43 only applies to new plants (i.e., those reviewed after October 1985) and, as such, does not apply to CGS.

A-44 Station Blackout

See Appendix 8A.

A-45 Shutdown Decay Heat Removal

According to guidance provided in Generic Letter 89-21 and Supplement 9 to NUREG-0933, Energy Northwest incorporated closure of A-45 into the CGS IPE program the results of which were submitted to the NRC by Reference 1.5-22.

A-46 Seismic Qualification of Equipment in Operating Plants

Generic Letter 87-03 issued February 27, 1987 which addresses A-46 resolution for CGS did not require any action or plant review. NUREG-1211, Enclosure 1, established Generic Letter 87-03 as applicable to CGS rather than Generic Letter 87-02. As such, Energy Northwest considers this USI closed for CGS. Also, NUREG-0892, Supplement 5 in Appendix C states that A-46 only applies to plants that were operating at the time.

A-47 Safety Implication of Control System

Generic Letter 89-19 provides requirements to close A-47. The overfill protection system required of BWRs is provided for in CGS. Closure of this issue was provided by Reference 1.5-9.

A-48 Hydrogen Control Measures and Effects of Hydrogen Burn on Safety Equipment

As stated in Generic Letter 89-21, A-48 is closed and implemented for Mark II BWRs such as CGS.

1.5.1.1.3 Unresolved Safety Issues Implementation Summary

The resolution of all USIs for CGS has been achieved with the NRC. Regarding Station Blackout (A-44), 10 CFR 50.63(c)(4) provides for a 2 year implementation schedule for closure of identified modifications.

1.5.1.2 Generic Safety Issues

1.5.1.2.1 Generic Safety Issues Introduction

In Generic Letter 90-04, Reference 1.5-12, the NRC requested that licensees and construction permit holders address a list of specific generic safety issues (GSIs) listed in the generic letter. Energy Northwest's response to this request for CGS was provided in Reference 1.5-13.

1.5.1.2.2 Implementation of Specific Generic Safety Issues

The following summarizes the CGS implementation of applicable GSIs listed in Generic Letter 90-04 and other GSIs that have been resolved for CGS subsequent to the issuance of the Generic Letter. The following is a summary of information provided in Reference 1.5-13 with updated information provided as appropriate.

GSI/Subject	*Status*
40/BWR Scram System Pipe Breaks	*Closed as documented in NUREG-0892 (p. 4-4) and documents listed in Reference 1.5-13*
41/BWR Scram Discharge Volume	*Closed as documented in NUREG-0892 (p. 7-6)*
43/Reliability of Air Systems	*Closed as discussed in References 1.5-13 and 1.5-15*
48/LCOs for Class 1E vital Instrumentation Buses - Generic Letter 91-11 (added subsequent to Generic Letter 90-04 response)	*Closed as documented in Reference 1.5-19*
49/Interlocks and LCOs for Class 1E Tie Breakers - Generic Letter 91-11 (added subsequent to Generic Letter 90-04 response).	*Closed as documented in Reference 1.5-19*
51/Improved Reliability of Open-Cycle Service Water Systems	*Closed subsequent to Generic Letter 90-04 as addressed by References 1.5-11, 1.5-37, and 1.5-38*

67/Improved Accident Safety Report Monitoring	*Closed as summarized in NRC Evaluation for CGS Regulatory Guide 1.97 implementation (Reference 1.5-14)*
75/Salem ATWS Events	*Closed subsequent to the Generic Letter 90-04 response by letters listed in Reference 1.5-13, Reference 1.5-17, and issuance of Technical Specification Amendment 90. Generic Letter 83-28, Supplement 1, issued October 7, 1992, did not change this status as CGS does not use reactor trip breakers.*
79/RPV Thermal Stress During Natural Convection Cooldown	*Closed subsequent to Generic Letter 90-04 by Generic Letter 92-02 as not impacting BWRs*
86/Long Range Plan for Stress Corrosion Cracking in BWR Piping	*Closed based upon documents listed in Reference 1.5-13.*
A-13/Snubber Operability Assurance	*NUREG-0933 states that this issue was resolved in 1980 by revision to the Standard Technical Specifications (STS). As the original CGS Technical Specifications were based upon Revision 3 to the BWR STS issued in 1980, this concern is resolved for CGS. In particular, for the five issues mentioned for GSI A-13 resolution in Generic Letter 90-04:* *1. The arbitrary capacity limit of 50,000 lbs that previously existed in Technical Specifications does not appear in the CGS Technical Specifications.* *2. The requirement for NRC approval of seal material does not appear in the CGS Technical Specifications.* *3, 4. Monitoring and IST programs to ensure snubber reliability do exist in the CGS Licensee Controlled Specifications. They are significantly expanded from that included in earlier programs.*

	5. *The CGS Licensee Controlled Specifications allow for an in-place snubber IST program.* *Thus, the five requirements of A-13 resolution as discussed in Generic Letter 90-04 have been implemented for CGS*
A/30 Adequacy of Safety Related DC Power Supplies - Generic Letter 91-06 (added subsequent to Generic Letter 90-04 response)	*Closed as documented in Reference 1.5-18*
A-35/Adequacy of Offsite NUREG-0892 Power Systems	*Closed as documented in NUREG-0892 (p. 8-16) and discussed in Reference 1.5-13)*
B-63/Installation of Low Pressure Systems Connected to the RCPB	*Closed as discussed in Question 040.079 (FSAR Volume 22) and Reference 1.5-13*

1.5.1.2.3 Generic Safety Issues Implementation Summary

Implementation of the applicable GSIs of Generic Letter 90-04 is complete.

1.5.1.3 TMI Task Action Plans

The CGS responses to the TMI-2 action plans as they were included in NUREG-0737 are provided in Appendix B. This Appendix agrees with Reference 1.5-16 in documenting that all TMI Task Action Plans have been implemented for CGS.

1.5.2 REFERENCES

1.5-1 Letter, GO2-89-215, G. C. Sorensen to NRC, "Response to Generic Letter 89-21 Requesting Plant Status on Implementation of Unresolved Safety Issues," dated November 30, 1989.

1.5-2 Letter, R. B. Samsworth (NRC) to G. C. Sorensen (SS), "Unimplemented Unresolved Safety Issues at WNP-2 (TAC No. 74538)," dated March 20, 1990.

1.5-3 Letter, R. B. Samworth (NRC) to G. C. Sorensen (SS), "ATWS Rule 10 CFR 50.62 relating to ARI and RPT Systems," dated November 6, 1988.

1.5-4 Letter, GO2-90-110, G. C. Sorensen to NRC, "Anticipated Transients Without Scram (ATWS) Design Modifications," dated June 22, 1990.

1.5-5 Letter, GO2-89-110, G. C. Sorensen (SS) to NRC, "Anticipated Transients Without Scram Implementation Schedule," dated June 16, 1989.

1.5-6 Letter, GO2-90-116, G. C. Sorensen (SS) to NRC, "Resolution of ATWS for WNP-2," dated June 29, 1990.

1.5-7 Letter, R. B. Samworth (NRC) to G. C. Sorensen (SS), "Issuance of Amendment No. 43," dated May 29, 1987.

1.5-8 Letter, GO2-89-062, G. C. Sorensen (SS) to NRC, "Response to Station Blackout Rule using HPCS Diversion III as Alternate AC Power," dated April 17, 1989.

1.5-9 PL Eng (NRC) to G. C. Sorensen (SS), Response to "Request for Action Related to Resolution of Unresolved Safety Issue A-47 - Safety Implications of Control System in LWR Nuclear Power Plants, pursuant to 10 CFR 50.54(f) - Generic Letter 89-19 (TAC NO. 75019)," dated November 13, 1991.

1.5-10 BWROG Topical Report NEDE-31096-P, "Anticipated Transients Without Scram; Response to NRC ATWS Rule 10 CFR 50.62," dated December 1985.

1.5-11 Letter, PL Eng (NRC) to G. C. Sorensen (SS), Evaluation of Response to NRC Generic Letter 89-13, "Service Water System Problems Affecting Safety-Related Equipment (TAC No. 74086)," dated April 26, 1992.

1.5-12 Generic Letter 90-04, "Request for Information on the Status of Licensee Implementation of Generic Safety Issues Resolved With Imposition of Requirements or Corrective Actions," dated April 25, 1990.

1.5-13 Letter, GO2-90-113, G. C. Sorensen to NRC, "Response to Generic Letter 90-04 Regarding Status of Implementation of Generic Safety Issues, (TAC No. 75993)," dated June 28, 1990.

1.5-14 Letter, G. W. Knighton (NRC) to G. C. Sorensen (SS), "Emergency Response Capability - Conformance to Regulatory Guide 1.97, Revision 2, (TAC No. 59516)," dated March 23, 1988.

1.5-15 Letter, GO2-89-128, G. C. Sorensen to NRC, "Final Response to Generic Letter 88-14, 'Instrument Air Supply Problems Affecting Safety-Related Equipment," dated July 28, 1989.

1.5-16 NUREG-1435 Supplement 2, "Status of Safety Issues at Licensed Power Plants," dated December 1992.

1.5-17 Letter, P. L. Eng (NRC) to G. C. Sorensen (SS), "Response to Generic Letter 90-03 for Washington Nuclear Plant 2 (TAC No. 76314)," dated November 8, 1990.

1.5-18 Letter, W. M. Dean (NRC) to G. C. Sorensen (SS), "Response to Generic Letter 91-06, MPA L106, Resolution of Generic Issue A-30, Adequacy of Safety Related DC Power Supplies, Pursuant to 10 CFR 50.54(f) for Washington Public Power Supply System Unit 2 (TAC NO. M81515)," dated March 27, 1992.

1.5-19 Letter, P. L. Eng (NRC) to G. C. Sorensen (SS), "Response to Generic Letter 91-11, 'Resolution of Generic Issues 48-LCOs for Class 1E Vital Instruments Buses and 49 - Interlocks and LCOs for Class 1E Tie Breakers' pursuant to 10 CFR 50.54(f) for Washington Public Power Supply System Nuclear Plant No. 2 (TAC No. M82484)," dated March 2, 1992.

1.5-20 Letter, P. L. Eng (NRC) to G. C. Sorensen (SS), "Status of TMI Item I.D.1.2, 'Detailed Control Room Design Review (DCRDR) at Washington Public Power Supply System Nuclear Project No. 2 (WNP-2) (TAC No. 56181)," dated November 13, 1991.

1.5-21 Letter, P. L. Eng (NRC) to G. C. Sorensen (SS), "Response to GL 88-01, Intergranular Stress Corrosion in Piping (TAC No. 69161)," dated December 28, 1990.

1.5-22 Letter, GO2-92-206, G. C. Sorensen (SS), "Response to Generic Letter 88-20," Individual Plant Examinations for Severe Accident Vulnerabilities 10 CFR 50.54(f)," dated August 28, 1992.

1.5-23 through 1.5-34 Deleted

1.5-35 Letter, GO2-92-004, G. C. Sorensen to NRC, "Response to NRC SER on Generic Letter 88-01 (TAC No. 69161)," dated January 8, 1992.

1.5-36 Letter, GO2-92-086, G. C. Sorensen to NRC, "Additional Response to Generic Letter 88-01 Safety Evaluation Report (TAC Nos. M80358 and M69161)," dated April 10, 1992.

1.5-37 *Letter, GO2-90-017, G. C. Sorensen to NRC, "Response to Generic Letter 89-13, Service Water System Problem Affecting Safety-Related Equipment," dated February 5, 1990.*

1.5-38 *Letter, GO2-91-041, G. C. Sorensen to NRC, "Response to Generic Letter 89-13, Service Water System Problems Affecting Safety-Related Equipment," dated February 28, 1991.*

1.6 MATERIAL INCORPORATED BY REFERENCE

Table 1.6-1 is a list of General Electric topical reports and other reports and documents which are incorporated in whole or in part by reference. These documents were filed with the NRC.

Table 1.6-1

Topical Reports

Report	Title	FSAR Section
General Electric Company		
APED-4824	Maximum Two-Phase Vessel Blowdown from Pipes (April 1965)	6.2
APED-5458	Effectiveness of Core Standby Cooling Systems for General Electric Boiling Water Reactors (March 1968)	5.4
APED-5460	Design and Performance of General Electric BWR Jet Pumps (July 1968)	3.9
APED-5555	Impact Testing on Collet Assembly for Control Rod Drive Mechanism 7RDB144A (November 1967)	4.6
APED-5640	Xenon Considerations in Design of Large Boiling Water Reactors (June 1968)	4.1
APED-5652	Stability and Dynamic Performance of the General Electric Boiling Water Reactor (April 1969)	4.1
APED-5696	Tornado Protection for the Spent Fuel Storage Pool (November 1968)	3.3, 3.5, 9.1
APED-5706	Incore Neutron Monitoring System for General Electric Boiling Water Reactors (November 1968; revised April 1969)	7.6
APED-5750	Design and Performance of General Electric Boiling Water Reactor Main Steam Line Isolation Valves (March 1969)	3.9, 5.4
GEAP-5620	Failure Behavior in ASTM A106B Pipes Containing Axial Through-Wall Flaws (April 1968)	5.2

Table 1.6-1

Topical Reports (Continued)

Report	Title	FSAR Section
GEAP-10546	Theory Report for Creep-Plast Computer Program (January 1972)	4.1
GEAP-13197	Emergency Cooling in BWRs Under Simulated Loss-of-Coolant (BWR PLECMP) Final Report (June 1971)	6.2
GE-NE-778-028-0790	GE Duralife 215 Control Rod Safety Evaluation, Revision 2 (July 1992)	4.2
GE-NE-187-24-0992	Washington Public Power Supply System Nuclear Project 2, SRV Setpoint Tolerance and Out-of-Service Analysis, Revision 2 (July 1993)	6.3
NEDC-31984-P	Generic Evaluations of General Electric Boiling Water Reactor Power Uprate - (July 1991)	5.4, 15.8
NEDC-32115-P	Washington Public Power Supply System Nuclear Project 2, SAFER/GESTR-LOCA Loss-of-Coolant Accident Analysis (September 1992)	6.3
NEDC-32141-P	Power Uprate With Extended Load Line Limit Safety Analysis for WNP-2 (June 1993)	5.4, 15.8
NEDC-32232-P	WNP-2 Reactor Recirculation Adjustable Speed Drive (ASD) System Reliability Analysis (August 1993)	7.7
NEDC-32983-P-A	General Electric Methodology for Reactor Pressure Vessel Fast Neutron Flux Evaluations (January 2006)	4.3.2.8, 4.3.4
NEDE-10169	Safe-System Analysis for Standby Core Cooling Equipment (September 1970)	3A

Table 1.6-1

Topical Reports (Continued)

Report	Title	FSAR Section
NEDE-10313-P	PDA - Pipe Dynamic Analysis Program for Pipe Rupture Movement	3.6
NEDE-11146-P	Design Basis for New Gas System (July 1971)	11.3
NEDE-13442-P-01	Mark II - Pressure Suppression Test Program (May 1976)	3A
NEDE-20943-P	Urania-Gadolinia Nuclear Fuel Physical and Irradiation Characteristics and Material Properties (January 1977)	4.2
NEDE-20944-P	BWR/4 and BWR/5 Fuel Design (October 1976)	Table 1.3-1
NEDE-21175-3-P	Fuel Assembly Evaluation of Combined Safe Shutdown Earthquake (SSE) and Loss-of-Coolant Accident (LOCA) Loadings (July 1982)	3.9
NEDE-21354-P	BWR Fuel Channel Mechanical Design and Deflection (September 1976)	3.9
NEDE-21471-P	Analytical Model for Estimating Drag Forces on Rigid Submerged Structures Caused by LOCA and Safety/Relief Valve Ramshead Air Discharges (September 1977)	3A
NEDE-21544-P	Mark II Pressure Suppression Containment System, an Analytical Model of the Pool Swell Phenomenon (December 1976)	3A, 6.2
NEDE-21821	BWR Feedwater Nozzle/Sparger Final Report (March 1978)	5.2, 5.3
NEDE-23604	Brunswick Unit 1 Reacor Internals Vibration and Temperature Measurements (June 1977)	5.3

Table 1.6-1

Topical Reports (Continued)

Report	Title	FSAR Section
NEDE-23749-P	Analytical Model for Computing Transient Pressure and Forces in the S/RVDL (February 1978)	3.9
NEDE-23806-P	MK II Main Vent Lateral Loads Summary Report (October 1978)	3A
NEDE-24010-P	Technical Bases for the Use of the SRSS Method for Combining Dynamic Loads for Mark II Plants (July 1977) with Supplement 1 (October 1978), Supplement 2 (December 1978), and Supplement 3 (August 1979)	3.9
NEDE-24011-P-A-16	General Electric Standard Application for Reactor Fuel (October 2007)	3.9, 4.1, 4.2, 4.3, 4.4, 15.1, 15.4
NEDE-24057-P	Assessment of Reactor Internals Vibration in BWR/4 and BWR/5 Plants (November 1977)	3.9
NEDE-24106-P	Dynamic Lateral Loads on a Main Vent Downcomer - Mark II Containment (March 1978)	3A
NEDE-24222	Assessment of Boiling Water Reactor Mitigation of Anticipated Transient Without Scram, Volume II (December 1979)	15.8
NEDE-24285-P	Chugging Loads - Revised. Definition and Application Methodology for Mark II Containments (July 1981)	3A
NEDE-24288-P	Generic Condensation Oscillation Load Definition Report (November 1980)	3A
NEDE-24302-P	Generic Chugging Load Definition Report (April 1981)	3A

Table 1.6-1

Topical Reports (Continued)

Report	Title	FSAR Section
NEDE-24695	RVF0R04 User's Manual, S/RVDL Clearing Transient Pressures and Forces in the S/RDL (December 1979)	3.9
NEDE-24794-P	Dynamic Lateral Loads on Mark II Main Vent Downcomer - Correlation of Independent Reference Data (March 1980)	3A
NEDE-24811-P	4T Condensation Oscillation Test Program Final Test Report (May 1980)	3A
NEDE-24822-P	Mark II Improved Chugging Methodology (May 1980)	3A
NEDE-24834	Hanford 2 Crimped Control Rod Drive Line (June 1980)	3.6
NEDE-24988-P	Analysis of Generic BWR Safety/Relief Valve Operability Test Results (October 1981)	5.2, 5.4, Table F.4-1
NEDE-25100-P	CAORSO SRV Discharge Tests Phase I Test Report (May 1979)	3A
NEDE-25118	CAORSO SRV Discharge Tests Phase II ATR Report (August 1979)	3A
NEDE-31096-P	Licensing Topical Report, Anticipated Transient Without Scram (February 1987)	4.6, 7.4, 9.3
NEDM-10320	The General Electric Pressure Suppression Containment Analytical Model (March 1971)	3A, 6.2
NEDO-10029	An Analytical Study on Brittle Fracture of GE BWR Vessel Subject to the Design Basis Accident (July 1969)	1.8
NEDO-10320	The General Electric Pressure Suppression Containment Analytical Model (April 1971)	3A 6.2

Table 1.6-1

Topical Reports (Continued)

Report	Title	FSAR Section
NEDO-10329	Loss-of-Coolant Accident and Emergency Core Cooling Models for General Electric Boiling Water Reactors (April 1971); Supplement 1, (April 1971); Addenda, (May 1971)	6.2
NEDO-10349	Analysis of Anticipated Transients Without Scram (March 1971)	15.8
NEDO-10466-A	Power Generation Control Complex Design Criteria and Safety Evaluation (September 1977)	8.3, 9.5, F.2, F.3, F.7
NEDO-10527	Rod Drop Accident Analysis for Large Boiling Water Reactors (March 1976); Supplement 1, (July 1972); Supplement 2, (January 1973)	4.2, 15.4
NEDO-10602	Testing of Improved Jet Pump for the BWR/6 Nuclear System (June 1972)	3.9
NEDO-10734	A General Justification for Classification of Effluent Treatment System Equipment as Group D (February 1973)	11.3
NEDO-10751	Experimental and Operational Confirmation of Off-Gas System Design Parameters (January 1973) (Proprietary)	11.3
NEDO-10802	Analytical Methods of Plant Transient Evaluations for General Electric Boiling Water Reactor (February 1973)	15.2
NEDO-10899	Chloride Control in BWR Coolants (June 1973)	1.8, 5.2
NEDO-10905	HPCS Power Supply	1.8, 8.3

Table 1.6-1

Topical Reports (Continued)

Report	Title	FSAR Section
NEDO-10951	Releases from BWR Radwaste Management Systems (July 1973)	11.2
NEDO-10958-A	General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation, and Design Application (January 1977)	6A
NEDO-20533	The General Electric Mark III Pressure Suppression Containment System Analytical Model (June 1974)	3A, 6.2
NEDO-20566-P-A	Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50, Appendix K (Proprietary) (January 1976)	3.9, 4.2
NEDO-20626	Studies of BWR Designs for Mitigation of Anticipated Transients without Scrams (October 1974)	6.2, 9.3
NEDO-20761	Millstone Nuclear Power Station, Refueling/Maintenance Outage (Fall 1974)	12.2
NEDO-21061	Mark II Containment Dynamics Forcing Functions Information Report (September 1976, June 1978, November 1981)	3A, 6.2
NEDO-21142	Realistic Accident Analysis for General Electric Boiling Water Reactor - The RELAC Code and User's Guide (December 1977)	15.2, 15.6
NEDO-21231	Banked Position Withdrawal Sequence (September 1976)	15.4
NEDO-21471	Analytical Model for Estimating Drag Forces on Rigid Submerged Structures Caused by LOCA and Safety/Relief Valve Ramshead Air Discharges (September 1977)	3A

Table 1.6-1

Topical Reports (Continued)

Report	Title	FSAR Section
NEDO 21667	Comparison of the 1/13 Scale Mark II Containment Multivent Pool Swell Data with Analytical Methods (August 1977)	3A
NEDO-21708	Radiation Effects in Boiling Water Reactor Vessel Steels (October 1977)	5.3
NEDO-21778-A	Transient Pressure Rises Affecting Fracture Toughness Requirements for Boiling Water Reactors January 24, 1978 (January 17, 1979)	5.3
NEDO-21985	Functional Capability Criteria for Essential Mark II Piping (September 1978)	3.9
NEDO-23678-P	Mark II Pressure Suppression Test Program Phases I, II, and III of the 4T Tests (June 1978)	3A
NEDO-24057-P	Assessment of Reactor Internals Vibration in BWR/4 and BWR/5 Plants (November 1977)	3.9
NEDO-24154-A	Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors, Volumes 1 and 2 (August 1986)	5.2
NEDO-24210	PISYS Analysis of NRC Problem (August 1979)	3.9
NEDO-24226	General Electric Company, Control Blade Lifetime With Potential B_4C Loss, with Supplement 1 (December 1979)	4.2
NEDO-24288	Mark II Containment Program - Generic Condensation Oscillation Load Definition Report (February 1981)	3A
NEDO-24548	Technical Description Annulus Pressurization Load Adequacy Evaluation (January 1979)	6.2

Table 1.6-1

Topical Reports (Continued)

Report	Title	FSAR Section
NEDO-24708-A	Additional Information Required for NRC Staff Generic Report on Boiling Water Reactors (June 1980)	7.4, B, I, Table F.4-1
NEDC-24154-P-A	Qualification of the One-Dimensional Core Transient Model for Boiling Water Reactors, Volumes 1, 2, 3 and 4 (February 2000)	15.1, 15.2, 15.3, 15.5
NEDC-32084P-A	TASC-03A A Computer Program for Transient Analysis of a Single Channel (July 2002)	6.3
NEDC-32601P-A	Methodology and Uncertainties for Safety Limit MCPR Evaluations (August 1999)	4.4
NEDC-32694P-A	Power Distribution Uncertainties for Safety Limit MCPR Evaluations (August 1999)	4.4
NEDC-32851-P-A	GEXL14 Correlation for GE14 Fuel (September 2007)	4.4
NEDC-32868P	GE14 Compliance With Amendment 22 of NEDE-24011-P-A (GESTAR II) (September 2007)	4.1, 4.2, 4.3
NEDC-32950P	Compilation of Improvements to GENE's SAFER ECCSLOCA Evaluation Model (July 2007)	6.3
NEDC-33419P	GEXL97 Correlation Applicable to ATRIUM-10 Fuel (June 2008)	4.4
NEDE-23785-1-PA	The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident. Volumes 1, 2, and 3 (October 1984)	6.3

Table 1.6-1

Topical Reports (Continued)

Report	Title	FSAR Section
NEDE-23785P-A	The GESTR-LOCA and SAFER Models for the Evaluation of the Loss-of-Coolant Accident. Volume 3 Supplement 1, Additional Information for Upper Bound PCT Calculation. (March 2002)	6.3
NEDE-24011-P-A-US	General Electric Standard Application for Reactor Fuel (GESTAR II) (Supplement for United States) (most recent approved revision)	3.9, 4.1, 4.2, 4.3, 4.4, 15.4
NEDE-30130-P-A	Steady State Nuclear Methods (April 1985)	15.1, 15.4

Exxon Nuclear Company / Advanced Nuclear Fuels Corp. / Siemens Power Corporation / Framatome ANP / Areva NP Inc.

Report	Title	FSAR Section
ANF-524 (P) (A)	Advanced Nuclear Fuels Corporation Critical Power Methodology for Boiling Water Reactors, Revision 2, and Supplements 1 and 2 (November 1990)	4.4
ANF-913(P)(A)	CONTRANSA2: A Computer Program for Boiling Water Reactor Transient Analyses, Volume 1 Revision 1 and Volume 1 Supplements 2, 3, and 4 (August 1990)	15.1, 15.2
ANF-1358(P)(A)	The Loss of Feedwater Heating Transient in Boiling Water Reactors, Revision 3 (September 2005)	15.1
ANF-89-98 (P)(A)	Generic Mechanical Design Criteria for BWR Fuel Designs, Revision 1 and Supplement 1 (May 1995)	3.9, 4.1, 4.2, 4.3, 4.4
EMF-CC-074 (P)(A)	BWR Stability Analysis Assessment of STAIF with Input from MICROBURN-B2, Volume 4, Revision 0, (August 2000)	4.1, 4.3

Table 1.6-1

Topical Reports (Continued)

Report	Title	FSAR Section
EMF-93-177 (P)(A)	Mechanical Design for BWR Fuel Channels, Revision 1 (August 2005)	3.9
EMF-2158(P)(A)	Siemens Power Corporation Methodology for Boiling Water Reactors; Evaluation and Validation of CASMO-4/MICROBURN-B2, Revision 0 (October 1999)	4.4, 15.1, 15.4
EMF-2209(P)(A)	SPCB Critical Power Correlation, Revision 2 (September 2003)	4.4
EMF-2245(P)(A)	Applications of Siemens Power Corporation Critical Power Correlations to Co-resident Fuel, Revision 0 (August 2000)	4.4
XN-NF-80-19 (P)(A)	Exxon Nuclear Methodology for Boiling Water Reactors, THERMEX: Thermal Limits Methodology Summary Description, Volume 3, Revision 2 (January 1987)	15.4
	Application of the ENC Methodology to BWR Reloads, Volume 4, Revision 1 (June 1986)	15.4
XN-NF-81-58 (P)(A)	RODEX2 Fuel Rod Thermal-Mechanical Response Evaluation Model, Revision 2 and Supplements 1 and 2 (March 1984)	6.3
XN-NF-82-07 (P)(A)	Exxon Nuclear Company ECCS Cladding Swelling and Rupture Model, Revision 1 (November 1982)	4.2, 6.3
Asea Brown Boveri (ABB) / CE Nuclear Power / Westinghouse Electric Company		
CENPD-287-P-A	Fuel Assembly Mechanical Design Methodology for Boiling Water Reactors (July 1996)	4.4

Table 1.6-1

Topical Reports (Continued)

Report	Title	FSAR Section
CENPD-300-P-A	Reference Safety Report for Boiling Water Reactor Reload Fuel (July 1996)	4.1, 4.2, 4.3, 4.4
CENPD-392-P-A	10 x 10 SVEA Fuel Critical Power Experiments and CPR Correlations: SVEA-96, Revision 0 (September 2000)	4.4
Other References		
WPPSS-74-2-R2 and Supplements WPPSS-74-2-R2A and WPPSS-74-2-R2B	Washington Public Power Supply System Sacrificial Shield Wall (March 1974) Sacrificial Shield Wall Design Supplemental Information (February 1975, August 1975)	3.8, 6.2
Report Submitted with letter GO2-80-172, August 8, 1980	Engineering Evaluation of the WNP-2 Sacrificial Shield Wall (March 1974)	3.8, 6.2
Report submitted with letter GO2-80-182, August 19, 1980	Engineering Evaluation of the WNP-2 Sacrificial Shield Wall, Supplement No. 1	3.8, 6.2
--	Plant Design Assessment Report for SRV and LOCA Loads	3A
WPPSS-74-2-R3	Burns & Roe, Inc., Protection Against Pipe Breaks Outside Containment (April 1974)	3.5
WPPSS-74-2-R5	Drywell to Wetwell Leakage Study (July 1974, February 1974) (GO2-74-17, dated May 9, 1974)	6.2, 3.8
Inservice Inspection Program Plan	Inservice Inspection Program Plan Interval - 2	5.2.4, 6.6

Table 1.6-1

Topical Reports (Continued)

Report	Title	FSAR Section
Preservice Inspection Program Plan	Preservice Inspection Program Plan	5.2.4, 6.6
CGS-FTS-0168	Columbia Generating Station Alternative Source Term (report consolidated from letters GO2-04-170 dated September 30, 2004, GO2-06-116 dated September 11, 2006, GO2-05-054 dated March 16, 2005, GO2-05-160 dated September 29, 2005, GO2-06-043 dated March 21, 2006, GO2-06-105 dated August 7, 2006 and GO2-06-108 dated August 24, 2006)	1.8, 15.4, 15.6, 15.7

1.7 ACRONYMS

The acronyms used in this FSAR follow

ACI	American Concrete Institute
ACRS	Advisory Committee on Reactor Safeguards
ADS	automatic depressurization system
AEC	Atomic Energy Commission
AISC	American Institute of Steel Construction
ALARA	as low as is reasonably achievable
ALI	annual limit on intake
ANSI	American National Standards Institute
APRM	average power range monitor
ARM	area radiation monitor
AS	auxiliary steam
ASCE	American Society of Civil Engineers
ASD	adjustable speed drive
ASME	American Society of Mechanical Engineers
ASTM	American Society for Testing and Materials
ATWS	anticipated transient without scram
AWS	American Welding Society
B&R	Burns and Roe, Inc.
BISI	bypass & inoperable status indication
BOC	beginning of cycle

BPA	Bonneville Power Administration
BPC	Bechtel Power Corporation
BWR	boiling water reactor
CAS	central alarm station, control air system
CEP	containment exhaust purge
CGS	Columbia Generating Station
CHF	critical heat flux
CIA	containment instrument air
CMFA	common mode failure analysis
COLR	Core Operating Limits Report
COND	main condensate system
CPR	critical power ratio
CRA	primary containment cooling system
CRD	control rod drive
CRDA	control rod drop accident
CREF	control room emergency filtration
CRPI	control rod position indication
CSP	containment purge supply
CST	condensate storage and transfer, condensate storage tank
CW	circulating water
DAC	derived air concentrations
DAW	dry active radioactive waste

DB	design basis
DBA	design basis accident
DBE	design basis earthquake
DG	diesel generator
DEH	digital electrohydraulic
DOE	Department of Energy
DOP	dioctylphthalate
DZO	depleted zinc oxide
ECA	engineering change authorization
ECCS	emergency core cooling system
ECN	engineering change notice
EDR	equipment drain (radioactive) processing
EFCV	excess flow check valve
EHC	electrohydraulic control
EOC	end of cycle
EOF	emergency operations facility
EPA	electrical protection assembly
EPN	equipment piece number
EPZ	emergency planning zone
ESF	engineered safety feature
EWD	electrical wiring diagram
FA	full arc (mode of TGV operation)

FANP	Framatome ANP
F-B/V	front to back/vertical
FCD	functional control diagram
FCV	flow control valve
FDDR	Field Deviation Disposition Request
FDR	floor drain (radioactive) processing system
FLECHT	full-length emergency cooling heat transfer
FMEA	failure modes effects analysis
FPC	fuel pool cooling and cleanup
FSAR	Final Safety Analysis Report
GE	General Electric Company
HAD	heat actuated device
HCA	horizontal control accelerometer
HCU	hydraulic control unit
HEPA	high-efficiency particulate air/absolute
HID	high-intensity discharge (lighting--vapor lamp)
HPCS	high-pressure core spray
H&V	heating and ventilating
HVAC	heating, ventilating, and air conditioning
HX	heat exchanger
IBA	intermediate break accident
IDC	incident detection circuitry

IDS	instrument data sheet
IED	instrument engineering diagram
IEEE	Institute of Electrical and Electronics Engineers
IGSCC	intergranular stress corrosion cracking
IHSI	Induction Heat Stress Improvement
IRM	intermediate range monitor
ISA	Instrument Society of America
LCO	Limiting Condition of Operation
LCS	leak control system
LDS	leak detection system
LHGR	linear heat generation rate
LLRT	local leak rate test
LOCA	loss-of-coolant accident
LPCI	low-pressure coolant injection
LPCS	low-pressure core spray
LPRM	local power range monitor
LPZ	low population zone
LSSS	limiting safety system setting
MAPLHGR	maximum average planar linear heat generation rate
MCC	motor control center
MCPR	minimum critical power ratio
MEL	Master Equipment List

MG	motor-generator
MLD	mean low water datum
MLHGR	maximum linear heat generation rate
MOV	motor-operated valve
MS	main steam
MSIV	main steam isolation valve
MSIV-LCS	main steam isolation valve leakage control system
msl	mean sea level
MSL	main steam line
MSLC	main steam isolation valve leakage control
MWR	mixed waste (radioactive)
NB	nuclear boiler
NBR	nuclear boiler rated (power)
NDE	nondestructive examination
NDT	nil-ductility transition
NDTT	nil-ductility transition temperature
NEC	National Electrical Code
NED	Nuclear Energy Division (GE)
NFPA	National Fire Protection Association
NEPIA	Nuclear Energy Property Insurance Association
NMS	neutron monitoring system
NPDES	National Pollutant Discharge Elimination System

NPHS	net positive suction head
NRC	Nuclear Regulatory Commission
NSOA	nuclear safety operational analysis
NSSS	nuclear steam supply system
NSSSS	nuclear steam supply shutoff system
OBE	operating basis earthquake
OQAPD	Operational Quality Assurance Program Description
ODCM	Offsite Dose Calculation Manual
OPRM	Oscillation Power Range Monitor
OSHA	Occupational Safety and Health Act
OT	operating transient
OS&Y	outside screw and yoke
OT	operating transient
PA	Public Address (System)
PABX	Private Automatic Branch Exchange
PATP	Power Ascension Test Program
PCIOMR	preconditioning cladding interim operating management recommendation
PCRVICS	primary containment and reactor vessel isolation control system
PCS	process computer system
PCT	peak cladding temperature
PDIS	plant display information system
PEC	Plant Engineering Center

PGCC	power generation control complex
P&ID	piping and instrumentation diagram
PMF	probable maximum flood
PPM	Plant Procedure Manual
PRM	power range monitor
PSAR	Preliminary Safety Analysis Report
PSF	Plant Support Facility
PVS	plant vent stack
RBM	rod block monitor
RCC	reactor building closed cooling water
RCIC	reactor core isolation cooling
RCPB	reactor coolant pressure boundary
REA	reactor building exhaust air
RFW	reactor feedwater
RHR	residual heat removal
RMC	reactor manual control
RMS	remote manual switches
ROA	reactor building outside air
RPIS	rod position information system
RPS	reactor protection system
RPT	recirculation pump trip
RPV	reactor pressure vessel

RRC	reactor recirculation system
RRS	required response spectra
RSCS	rod sequence control system
RSO	reactor system outline
RWCU	reactor water cleanup
RWM	rod worth minimizer
RWP	Radiation Work Permit
SA	service air
SACF	single active component failure
SAF	single active failure
SAR	Safety Analysis Report
SAS	Secondary Alarm Station
SBA	small break accident
SBO	station blackout
SCF	single component failure
SDC	shutdown cooling
SEF	single equipment failure
SER	Safety Evaluation Report
SF	single failure (NSOA)
SGT	standby gas treatment
SGTS	standby gas treatment system
SJAE	steam jet air ejector

SLC	standby liquid control
SLMCPR	minimum critical power ratio safety limit
SLO	single loop operation
SMS	Scheduled Maintenance System
SOE	single operator error
SPC	Siemens Power Corporation
SPV	solenoid pilot valve
SRM	source range monitor
SRO	Senior Reactor Operator
SRP	Standard Review Plan
SRV	safety/relief valve
SS	safe shutdown
SS	stainless steel
SSC	structures, systems, and components
SSE	safe shutdown earthquake
S-S/V	side-to-side/vertical
SSW	sacrificial shield wall
SW	standby service water
SWP	Site Wide Procedure
TCV	turbine control valve
TDAS	transient data acquisition system
TEDE	total effective dose equivalent

TG	turbine generator
TGV	turbine governor valve
TIP	traversing in-core probe
TLD	thermoluminescent dosimeter
TMU	tower makeup
TPM	thermal power monitor
TRS	test response spectra
TSC	Technical Support Center
TSPM	Test and Startup Program Manual
TSW	plant service water (turbine building service water)
TWG	Test Working Group
UBC	Uniform Building Code
UHS	ultimate heat sink
UPS	uninterruptable power supply
WNP-2	Washington Nuclear Project No. 2
WPPSS	Washington Public Power Supply System
ZPA	zero period acceleration

1.8 CONFORMANCE TO NRC REGULATORY GUIDES

1.8.1 INTRODUCTION

This section of the FSAR contains information on Energy Northwest's conformance assessment of CGS to Regulatory Guides, Division 1, Power Reactor Guides and revisions thereof as noted.

Since the scope of equipment responsibility is project unique and the time of equipment design, procurement, manufacture, installation, and operation varies with the supplier, a unique assessment for the nuclear steam supply system (NSSS) scope of supply and balance of plant (BOP) scope of supply is necessary and is presented.

1.8.2 NUCLEAR STEAM SUPPLY SYSTEM SCOPE OF SUPPLY EVALUATION

The following paragraphs define the nomenclature and the manner in which the NSSS scope of supply assessment is to be interpreted.

Regulatory Guides - Incorporated in the Design

This section serves to identify specific safety or regulatory guides which were included in the plant as a design commitment during the construction permit review. It also identifies those incorporated by commitment after the construction permit issuance. All of these are specifically noted as "Incorporated in the Design."

Regulatory Guides - Assessed Capability in the Design

For those other regulatory guides which have been issued before, during, or after the construction permit issuance, Energy Northwest (through his agents and/or suppliers) has performed an assessment evaluation to determine the capability of the previously approved design to accommodate and meet these new requirements. These are noted as "Assessed Capability in the Design."

Conformance to the regulatory guide falls under either one of two categories - "Full Compliance" or "Meeting Intent Through an Alternate Approach."

Regulatory Guide - Full Compliance

Any regulatory guide so noted, whether by direct conformance or by assessed capability, complies with subject requirements as described in the FSAR.

Regulatory Guide - Meeting Intent by Alternate Approach

This designation is based on NRC rules which state that "Regulatory Guides are not substitutes for regulations, and compliance with them is not required. Methods and solutions different from those set out in the guides will be acceptable if they provide a basis for the findings requisite to the issuance or continuance of a permit or license by the Commission." The description and justification of an alternate approach is provided where this method is employed.

The following evaluation represents the NSSS scope of supply regulatory guide assessment.

Regulatory Guide 1.1, Revision 0, November 1970

Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal Pumps.

Regulatory Guide Intent:

This guide prohibits design reliance on pressure and/or temperature transients expected during a loss-of-coolant accident (LOCA) for ensuring adequate net positive suction head (NPSH). The requirements of this regulatory guide are applicable to the high-pressure core spray (HPCS), low-pressure core spray (LPCS), and residual heat removal (RHR) pumps.

Applicable Assessment:

Incorporated in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design, and/or equipment used in CGS is in full compliance with this regulatory guide.

General Compliance or Alternate Approach Assessment:

The boiling water reactor (BWR) design conservatively assumes 0 psig containment pressure and maximum expected temperature of the pumped fluids; thus no reliance is placed on pressure and/or temperature transients to assure adequate NPSH.

Requirements for NPSH are available at the centerline of the pump suction nozzles for each pump.

Specific Evaluation Reference:

See Sections 6.2 and 6.3.

Similar Application Reference:

Similar application was used for LaSalle and GESSAR.

Regulatory Guide 1.2, Revision 0, November 1970

Thermal Shock to Reactor Pressure Vessels

Regulatory Guide Intent:

This regulatory guide states that potential reactor pressure vessel brittle fracture which may result from emergency core cooling systems (ECCS) operation need not be reviewed in individual cases if no significant changes in presently approved core and pressure vessel designs are proposed. Should it be concluded that the margin of safety against reactor pressure vessel brittle failure due to ECCS operation is unacceptable, and engineering solution, such as annealing, could be applied to ensure adequate recovery of the fracture toughness properties of the vessel material. This regulatory guide requires that available engineering solutions be outlined and requires that it be demonstrated that the design does not preclude their use.

Application Assessment:

Incorporated in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design, and/or equipment used in CGS is in compliance with the intent of this regulatory guide through the incorporation of the alternate approach cited.

General Compliance or Alternate Approach Assessment:

The reactor pressure vessel used for CGS employs no significant core or vessel design changes from previously approved BWR pressure vessels such as Browns Ferry, all units.

An investigation of the structural integrity of BWR pressure vessels during a design-basis accident (DBA) has been conducted (see NEDO-10029, "An Analytical Study on Brittle Fracture of GE-BWR Vessel Subject to the Design Basis Accident"). It has been determined, based on methods of fracture mechanics, that no failure of the vessel by brittle fracture as a result of a DBA will occur.

The investigation included

a. A comprehensive thermal analysis considering the effect of blowdown and the LPCI system reflooding,

b. A stress analysis considering the effects of pressure, temperature, seismic load, jet load, dead weight, and residual stresses,

c. The radiation effect on material toughness [nil ductility transition temperature (NDTT) shift and critical stress intensity], and

d. Methods for calculating crack tip stress intensity associated with a nonuniform stress field following DBA.

This analysis incorporated very conservative assumptions in all areas (particularly in the areas of heat transfer, stress analysis effects of radiation on material toughness, and crack tip stress intensity). Therefore, the results reported in NEDO-10029 provide an upper bound limit on brittle fracture failure mode studies. Because of the upper bound approach, it is concluded that catastrophic failure of the pressure vessel due to the DBA is shown to be impossible from a fracture mechanics point of view. In the case studies, even if an acute flaw does form on the vessel inner wall, it will not propagate as the result of the DBA.

The criteria of 10 CFR 50, Appendix G, are interpreted as establishing the requirement for annealing. Paragraph IV C of Appendix G requires vessels to be designed for annealing of the beltline only where the predicted value of adjusted RT_{NDT} exceeds 200°F as defined in paragraph NB2331 of the ASME Section III Code. This predicted value is not exceeded; therefore design for annealing is not required.

Specific Evaluation Reference:

See Section 5.3.1.5.

Similar Application Reference:

Similar application was used for Browns Ferry 1, 2, and 3.

Regulatory Guide 1.6, Revision 0, March 1971

Independence Between Redundant Standby (Onsite) Power Source and Between Their Distribution Systems

Regulatory Guide Intent:

The guide states the extent and nature of independence of the two onsite power divisions required by General Design Criterion (GDC) 17. Key features that ensure operation and prevent cascading single failures from disrupting both power systems are delineated.

Application Assessment:

Incorporated in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design, and/or equipment used in this facility is in compliance with the intent of this regulatory guide through the incorporation of the alternate approach cited.

General Compliance or Alternate Approach Assessment:

This regulatory guide is applicable to redundant standby (onsite) power sources and their distribution systems.

HPCS Onsite Power System (NSSS Scope of Supply)

Division 3 (HPCS) is provided with one offsite power source. Only one offsite supply is connected because no credit is given to offsite power sources in accident analysis. The diesel generator breaker can be closed automatically only if the other source breakers to the (HPCS) load group are open.

When the HPCS diesel generator breaker is closed, no other source breaker can be closed automatically. No other means exist for automatically connecting redundant load groups with each other. The HPCS diesel generator may be manually connected to either Division 1 or to Division 2 in the extended station blackout (SBO) or non-DBA loss of offsite power (LOOP) scenario described in Section 8.3.1.1.7.2.1. The source breakers are administratively controlled in the open position to prevent paralleling of standby sources.

Sufficient interlocks are provided to prevent paralleling the diesel generators manually by operator error during loss of offsite power. Division 3 diesel generator is provided with only one prime mover.

The HPCS division dc load group is fed from one battery charger and one battery.

The HPCS standby power source and distribution system is independent from the other two standby power sources and associated distribution system in the plant.

Specific Evaluation Reference:

See Section 8.3.1.2.

Similar Application Reference:

Similar application was used for LaSalle.

Regulatory Guide 1.9, Revision 0, March 1971

Selection of Diesel Generator Set Capacity for Standby Power Supplies

Regulatory Guide Intent:

This guide provides an approach for ensuring sufficient onsite power capability and for determining load requirements of diesel generator set power sources.

Application Assessment:

Incorporated in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design, and/or equipment used in this facility is in compliance with the intent of this regulatory guide through the incorporation of the alternate approach cited.

General Compliance or Alternate Approach Assessment:

This regulatory guide is applicable to the standby ac power supply for the HPCS diesel. The specific guidelines are unduly restrictive when applied to the selection of the diesel generator set dedicated to the HPCS system. This is mainly due to the unique application of the special HPCS equipment relative to normal diesel generator units.

Specific conformance and alternate positions to and with Regulatory Guide 1.9 are described in the following statements:

Regulatory Guide 1.9, Position 2 Conformance

Chapter 8 illustrates that the 2000-hr rating of the HPCS diesel generator, the 90% of 30-minute rating, and the maximum coincidental load, are in conformance with this position. Intermittent loads such as motor-operated valves are considered for long-term loads.

Regulatory Guide 1.9, Position 3 Conformance

CGS load requirements were verified as test data was completed and analyzed, following the preoperational tests.

Regulatory Guide 1.9, Position 4 Conformance

The HPCS diesel generator unit is considered as a unique application with justifiable departure from the strict conformance to Regulatory Guide 1.9, Revision 0, regarding voltage and frequency limits during the initial loading transient. The HPCS system consists of one large pump and motor combination which represents more than 90% of the total load; consequently, limiting the momentary voltage drop to 25% and the momentary frequency drop to 5% would not significantly enhance the reliability of HPCS operation. To meet the specific regulatory guide requirements, a diesel generator unit approximately two to three times as large as that required to carry the continuous rated load would be necessary. The specific diesel engine-electric generator-pump assembly was designed specifically for this integral operation. The frequency and voltage over-shoot requirements of Regulatory Guide 1.9, Revision 0, are met. A factory testing program on a prototype unit has verified the following functions:

a. System fast-start capabilities,

b. Load-carrying capability,

c. Load shedding capability,

d. Ability of the system to accept and carry the required loads, and

e. The mechanical integrity of the diesel-engine generator unit and all of the major system auxiliaries.

GE Licensing Topical Report, HPCS Power Supply, NEDO-10905, describes the theoretical analytical aspects of the unique application including prototype and reliability test considerations.

The design of the HPCS diesel generator conforms with the applicable sections of IEEE criteria for Class 1E "Electrical Systems for Nuclear Power Generation Stations," IEEE 308-1971.

The generator has the capability of providing power for starting the required loads with operationally acceptable voltage and frequency recovery characteristics. A partial or complete load rejection will not cause the diesel engine to trip on overspeed.

A special prototypic test conducted at the LaSalle facility verified the hardware and load aspects of the HPCS power supply concept. This test is described in topical report NEDO-10905, Revision 3.

Specific Evaluation Reference:

See Section 8.3.1.2.1.4.

Similar Application Reference:

Similar application was used for LaSalle; for comparison see Table 8.3-6.

Regulatory Guide 1.13, Revision 0, March 1971

Fuel Storage Facility Design Basis

Regulatory Guide Intent:

This guide delineates design criteria that are appropriately applied to the fuel storage facility of the CGS plant.

Application Assessment:

Incorporated in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design, and/or equipment used in this facility is in compliance with the intent of this regulatory guide through the incorporation of the alternate approach cited.

General compliance or Alternate Approach Assessment:

This regulatory guide is applicable to the refueling platform within NSSS scope of supply.

The refueling platform is designed to prevent it from toppling into the pools during a safe shutdown earthquake (SSE). Redundant safety interlocks are provided as well as limit switches to prevent accidentally running the grapple into the pool walls.

Specific Evaluation Reference:

See Section 9.1.4.3.

Similar Application References:

Similar application was used for Nine Mile Point 2.

Regulatory Guide 1.20, Revision 2, May 1976

Comprehensive Vibration Assessment Program for Reactor Internals During Preoperational and Initial Startup Testing

Regulatory Guide Intent:

Regulatory Guide 1.20 describes a comprehensive vibration assessment program for reactor internals during preoperational and initial startup testing. The vibration assessment program meets the requirements of Criterion 1, "Quality Standards and Records," of Appendix A to 10 CFR Part 50 and Section 50.34, "Contents of Applications; Technical Information," of 10 CFR Part 50.

Application Assessment:

Incorporated in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design, and/or equipment used in this facility is in compliance with the intent of this regulatory guide through the incorporation of the alternate approach cited.

General or Alternate Approach Statement:

This regulatory guide is applicable to the core support structures and other reactor internals.

A vibration measurement program has been defined for the confirmatory testing of this plant during initial startup tests.

CGS *reactor internals were tested in accordance with provisions of Regulatory Guide 1.20, Revision 2, Category IV, using Tokai-2 as the limited valid prototype.*

Specific Evaluation Reference:

See Sections 3.9.2.1, 3.9.2.3, and 3.9.2.4.

Similar Application Reference:

Similar application was used for Browns Ferry 1.

Regulatory Guide 1.21, Revision 1, June 1974

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

Regulatory Guide Intent:

Regulatory Guide 1.21 describes programs for measuring, reporting, and evaluating releases of radioactive materials in liquid and gaseous effluents and guidelines for classifying and reporting the categories and curie content of solid wastes.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design, and/or equipment used in this facility is in compliance with the intent of this regulatory guide through the incorporation of the alternate approach cited.

General Compliance or Alternate Approach Assessment:

The process and effluent radiological monitoring and sampling system is designed to provide the monitoring and sampling capability required to make the measurements, evaluations, and reports recommended by this guide.

The radiation monitoring systems (RMS) provided to meet these objectives are

a. For gaseous effluent streams

   Reactor building ventilation exhaust plenum RMS

b. For liquid effluent streams

   1. Radwaste effluent RMS, and
   2. Service water RMS

c. For gaseous process streams

1. Offgas pretreatment RMS,
2. Offgas posttreatment RMS, and
3. Carbon bed vault RMS

d. For liquid process streams

1. RHR service water RMS, and
2. Reactor building closed cooling water RMS

These systems have the capability for alarm and initiation of automatic closure of waste treatment discharge valves in the affected systems prior to exceeding the normal operation limits specified in Technical Specifications thereby satisfying the intent of the regulatory guide.

Specific Evaluation Reference:

See Sections 7.6.1.1 and 11.5.1.

Similar Application Reference:

Similar application has not been used for other projects.

Regulatory Guide 1.22, Revision 0, February 1972

Periodic Testing of Protection System Actuation Function

Regulatory Guide Intent:

This guide describes acceptable design approaches that facilitate the periodic testing, during reactor operation, of actuation devices/equipment incorporated into the reactor protection system design. This regulatory guide is applicable to the systems within NSSS scope of supply listed in this regulatory guide.

Application Assessment:

Incorporated in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design, and/or equipment used for this facility is in full compliance with this regulatory guide.

General Compliance or Alternate Approach Assessment:

Compliance for each system is discussed for this plant in the listed references.

	Section
Reactor protection system	7.2.2.3
Emergency core cooling system	
HPCS	7.3.2.1.3
Automatic depressurization system (ADS)	7.3.2.1.3
LPCS	7.3.2.1.3
LPCI (RHR)	7.3.2.1.3
Primary containment and reactor vessel isolation control system (PCRVICS)	7.3.2.1.3
Reactor core isolation cooling (RCIC)	7.4.2.3
Leak detection system	7.6.2.4
HPCS standby power supply	8.1.3
RHR system containment spray cooling system	7.3.2.1.3
Suppression pool cooling system	7.3.2.1.3
Reactor shutdown cooling system	7.4.2.3
Standby liquid control system	7.4.2.3
Process radiation monitoring system	7.6.2.4

Specific Evaluation Reference:

See above.

Similar Application Reference:

Similar application has not been used for other projects.

Regulatory Guide 1.26, Revision 3, February 1976

Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste-Containing Components of Nuclear Power Plants

Regulatory Guide Intent:

Regulatory Guide 1.26 describes a quality classification system for determining acceptable quality standards for safety-related components containing water, steam, or radioactive material other than those components addressed in 10 CFR 50.55a.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design, and/or equipment used in this facility is in compliance with the intent of the subject regulatory guide through the incorporation of the alternate approach cited.

General Compliance or Alternate Approach Assessment:

The definition of quality group classifications for this plant was made initially and recorded in the Preliminary Safety Analysis Report (PSAR) in accordance with ASME Boiler and Pressure Vessel Code (B&PV), Sections III and VIII. Quality group classifications were maintained during design and construction and are actively maintained during plant operations and modifications commensurate with the safety functions performed by the safety-related components.

This regulatory guide is applicable to Quality Groups B through D pressure parts including piping, pumps, valves, and vessels. Section 3.2 shows the quality groups classifications of these parts. The safety-related RCPB of the RWCU system meets the quality grouping requirements of Regulatory Guide 1.26. Non-safety-related portions of the RWCU system are maintained as Quality Group D vice C as delineated by Regulatory Guide 1.26.

Specific Evaluation Reference:

See Section 3.2 and the Operational Quality Assurance Program Description (OQAPD).

Similar Application Reference:

Similar application was used for LaSalle.

The italicized information is historical and was provided to support the application for an operating license.

Regulatory Guide 1.28, Revision 0, June 1972

Quality Assurance Program Requirements (Design and Construction).

Regulatory Guide Intent:

This guide describes an acceptable method of complying with the NRC's regulations with regard to overall quality assurance program requirements.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

The identified BWR Quality Assurance Program used in this facility reflects compliance with provisions of NRC regulations and regulatory guides or NRC-approved alternate positions.

General Compliance or Alternate Approach Assessment:

The General Electric BWR Quality Assurance Program has been developed over the years such that at any point in time it has been in compliance with mandatory regulatory requirements such as 10 CFR 50, Appendix B, and the ASME Code. Implementation of the applicable ANSI-N45.2 series standards and the associated NRC regulatory guides (or NRC-approved GE alternate positions) has been an evolutionary process and although partial implementation has always been effected before the date of issue of the regulatory guide or "AEC Guidance on Quality Assurance," which recognized applicable ANSI standards, full implementation was not necessarily in place until the GE commitment date (see Attachment A for complete listing of GE commitment dates and extent of commitment).

Since GE operates under a single quality assurance (QA) program, quality system improvements, such as more formalized audits or certification programs, are generally implemented across the board on all active projects with no opportunity for retrofit of completed work; therefore, work performed later in a project is typically subject to more quality assurance effort as a result of additional requirements. Attachment B gives a graphic representation of the time relation of some of the major project activities with the date of issue of regulatory guides and the GE commitment dates. Because of the long generation cycle of the related ANSI Standard, GE had already *upgraded its QA program to at least partially implement each of the related ANSI Standards, where applicable, prior to the date of issue of the regulatory guide.*

Attachment B also shows approximate dates of NRC and utility customer/architect-engineer QA audits. These audits have been performed frequently enough and over a long enough time period to establish confidence that GE has been following a QA program which has kept current with customer and regulatory requirements. Obviously, where most equipment is ordered years in advance of shipment, the QA program at the time of shipment will necessarily be somewhat different from that which was in effect at the time of ordering; however, at any point in time the GE QA program has been equal or better than the requirements in effect at that time.

Specific Evaluation Reference:

Information was provided at the PSAR stage.

Similar Application Reference:

Similar application has not been used for other projects.

Regulatory Guide 1.29, Revision 3, September 1978

Seismic Design Classification

Regulatory Guide Intent:

Regulatory Guide 1.29 describes an acceptable method of identifying and classifying those features of light-water-cooled nuclear power plants that should be designed to withstand the effects of the SSE.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design, and/or equipment used in this facility is in compliance with the intent of this regulatory guide through the incorporation of the alternate approach cited.

General Compliance or Alternate Approach Assessment:

This regulatory guide is used as a basis for defining the systems and components which must meet Seismic Category I requirements.

For the purpose of defining equipment that should be described to withstand the SSE, NSSS equipment conforms to the guide. The regulatory guide needs to be more specifically integrated in the following areas:

C.1(b)

Application of this guide is limited to those reactor vessel internals which use engineered safety features, such as core spray piping, core spargers, and hardware, etc.

C.1(h)

The component cooling water portions of the reactor recirculation pumps are not required to be Seismic Category I since the pumps do not perform a safety function.

Specific Evaluation Reference:

See Section 3.2, Table 3.2-1, and the OQAPD.

Similar Application Reference:

Similar application was used for LaSalle.

Regulatory Guide 1.30, Revision 0, August 1972

Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment.

Regulatory Guide Intent:

This guide describes an acceptable method of complying with the NRC's regulations with regard to overall QA program requirements.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

The identified BWR Quality Assurance Program used in this facility reflects compliance with the provisions of NRC regulatory guide or NRC regulations and NRC-approved alternate positions.

General Compliance or Alternate Approach Assessment:

Reference compliance assessment for Regulatory Guide 1.28.

Specific Evaluation Reference:

Information was provided at the PSAR stage. Compliance is discussed in the OQAPD.

Similar Application Reference:

Similar application has not been used for other projects.

Regulatory Guide 1.31, Revision 1, June 1973

Control of Stainless Steel Welding

Regulatory Guide Intent:

Regulatory Guide 1.31 describes an acceptable method of implementing requirements with regard to the control of welding when fabricating and joining austenitic stainless steel components and systems.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design, and/or equipment used in this facility is in compliance with the intent of this regulatory guide through the incorporation of the alternate approach cited.

General Compliance or Alternate Approach Assessment:

All austenitic stainless steel weld filler materials were supplied with a minimum of 5% delta ferrite. This amount of ferrite is considered adequate to prevent microfissuring in austenitic stainless steel welds.

An extensive test program performed by GE, with the concurrence of the NRC, has demonstrated that controlling weld filler metal ferrite at 5% minimum produces production welds which meet the requirements of this regulatory guide.

A total of approximately 400 production welds in five BWR plants were measured and all welds met the requirements of the Interim Regulatory Position.

Specific Evaluation Reference:

See Section 5.2.3.

Similar Application Reference:

Similar application was used for LaSalle.

Regulatory Guide 1.32, Revision 1, March 1976

Use of IEEE 308-1974, "Criteria for Class 1E Electric Systems for Nuclear Power Generating Stations"

Regulatory Guide Intent:

This guide describes a method for implementation of electrical safety related equipment design relative to GDC 17 and 18. This guide does contain some conflicts between GDC 17 and IEEE 308-1974 that of course will require resolution by plant design implementation. This regulatory guide is applicable to the battery and battery charger of the HPCS standby power system.

Applicable Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design, and/or equipment used in this facility is in compliance with the intent of this regulatory guide through incorporation of the alternate approach cited.

General Compliance or Alternate Approach Assessment:

The HPCS battery charger has sufficient capacity to restore its battery to full charge under the maximum steady-state load within a 24-hr period. A period of 24 hr is considered to be adequate to restore the battery from the design minimum charge state to the fully charged state irrespective of the status of the plant.

Specific Evaluation Reference:

See Section 8.3.1.2.

Similar Application Reference:

Similar application was used for LaSalle.

Regulatory Guide 1.34, Revision 0, December 1972

Control of Electroslag Weld Properties.

Regulatory Guide Intent:

Regulatory Guide 1.34 describes an acceptable method of implementing requirements regarding control of weld properties when fabricating electroslag welds for nuclear components made of ferritic or austenitic materials.

Application Assessment:

Not applicable.

Compliance or Alternative Approach Statement:

Not applicable.

General Compliance or Alternate Approach Assessment:

The electroslag welding process is not used on components within the NSSS scope of supply. Therefore this regulatory guide is not applicable.

Specific Evaluation Reference:

Not applicable.

Similar Application Reference:

Not applicable.

Regulatory Guide 1.37, Revision 0, March 1973

Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants.

Regulatory Guide Intent:

This guide describes an acceptable method of complying with the NRC's regulations with regard to overall QA program requirements.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

The identified BWR Quality Assurance Program used in this facility reflects compliance with the provisions of NRC regulations and NRC regulatory guide or NRC-approved alternate position.

General Compliance or Alternate Approach Assessment:

Reference compliance assessment for Regulatory Guide 1.28.

Specific Evaluation Reference:

Information was provided at the PSAR stage. Compliance is discussed in the OQAPD.

Similar Application Reference:

Similar application has not been used for other projects.

Regulatory Guide 1.38, Revision 2, May 1977

Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage and Handling of Items for Water-Cooled Nuclear Power Plants.

Regulatory Guide Intent:

This guide describes an acceptable method of complying with the NRC's requirements for handling of nuclear materials.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

The identified BWR Quality Assurance Program used in this facility reflects compliance with the provisions of NRC regulations and the NRC regulatory guide or NRC-approved alternate position.

General Compliance or Alternate Approach Assessment:

Reference compliance assessment for Regulatory Guide 1.28.

Specific Evaluation Reference:

Information was provided at the PSAR stage.

Similar Application Reference:

Similar application has not been used for other projects.

Regulatory Guide 1.41, Revision 0, March 1973

Preoperational Testing of Redundant On-Site Electric Power Systems to Verify Proper Load Group Assignments.

Regulatory Guide Intent:

The requirements of this regulatory guide are applicable to the total onsite electric power systems within Energy Northwest's responsibility.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design, and/or equipment used in this facility is in full compliance with this regulatory guide.

General Compliance or Alternate Approach Assessment:

The HPCS power system is designed to be tested independently of any other redundant load group.

Specific Evaluation Reference:

See Sections 8.3 and 14.2.

Similar Application Reference:

Similar application was used for LaSalle.

Regulatory Guide 1.43, Revision 0, May 1973

Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components

Compliance or Alternate Approach Statement:

This regulatory guide is not applicable since stainless steel cladding on coarse grain low-alloy steel for safety class components is not used.

General Compliance or Alternate Approach Assessment:

Not applicable.

Specific Evaluation Reference:

Not applicable.

Regulatory Guide 1.44, Revision 0, May 1973

Control of the Use of Sensitized Steel

Regulatory Guide Intent:

The purpose of Regulatory Guide 1.44 is to address GDC 1 and 4 and 10 CFR 50 Appendix B requirements to control "the application and processing of stainless steel to avoid severe sensitization could lead to stress corrosion cracking." The guide proposes that this should be done by limiting sensitization due to welding as measured by ASTM A 262 Practice A or E, or another method that can be demonstrated to show nonsensitization in austenitic stainless steels.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design, and/or equipment used in this facility is in compliance with the intent of this regulatory guide through the incorporation of the alternate approach cited.

General Compliance or Alternate Approach Assessment:

Tests by GE indicate that the test specified by A262 A or E (Detecting Susceptibility to Intergranular Attack in Stainless Steel) detects sensitization in a gross way, and the tests do not provide a precise method of predicting susceptibility to stress corrosion cracking in the BWR environment.

All austenitic stainless steel for CGS was purchased in the solution heat treated condition in accordance with applicable ASME and ASTM specifications. Carbon content was limited to 0.08% maximum, and cooling rates from solution heat treating temperatures were required to be rapid enough to prevent sensitization.

Welding heat input was restricted to 110,000 joules per inch maximum, and interpass temperature was restricted to 305°F. High heat welding processes such as block welding and electroslag welding were not permitted. All weld filler metal and castings were required by specification to have a minimum of 5% ferrite.

Whenever any wrought austenitic stainless steel was heated to temperatures over 800°F, by means other than welding or thermal cutting, the material was re-solution heat treated.

These controls were used to avoid severe sensitization and to comply with the intent of Regulatory Guide 1.44.

Specific Evaluation Reference:

See Section 5.2.3.

Similar Application Reference:

Similar application was used for LaSalle.

Regulatory Guide 1.45, Revision 0, May 1973

Reactor Coolant Pressure Boundary Leak Detection System.

Regulatory Guide Intent:

The guidelines are prescribed to ensure that leakage detection and collection systems provide maximum practical identification of leaks from within the reactor coolant pressure boundary (RCPB).

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design, and/or equipment used in this facility is in compliance with the intent of this regulatory guide through the incorporation of the alternate approach cited.

General Compliance or Alternate Approach Assessment:

The leak detection system consists of temperature, pressure, fission product monitoring and flow sensors with associated instrumentation and alarms. This system detects, annunciates, and isolates (in certain cases) leakages in the following systems:

a. Main steam lines,
b. Coolant systems within the drywell,
c. Reactor water cleanup (RWCU) system,
d. RHR system,
e. RCIC system,
f. Feedwater system, and
g. HPCS system.

Leakage is separated into identified and unidentified categories thus meeting position C.1 of Regulatory Guide 1.45. The affected systems and the leakage detection methods are discussed in Section 5.2.5.1.

Small unidentified leaks (5 gpm and less) inside the drywell are detected by temperature changes, pressure changes, drain sump pump activities, fission product monitoring, and floor drain flow monitoring; floor drain flow includes drywell cooler condensate flow.

Large leaks are also detected by changes in reactor water level and changes in flow rates in process lines.

The 5 gpm leakage rate is the limit on unidentified leakage inside the drywell. The leak detection system is capable of monitoring the flow rates with an accuracy of 1 gpm and is thus in compliance with paragraph C.2 of Regulatory Guide 1.45.

By monitoring drywell equipment and floor drain sump flow rates, which includes drywell coolers' condensate flow rates, and fission products (airborne particulate and gaseous radioactivity), position C.3 is satisfied.

Isolation and/or alarm of affected systems and the detection methods used are summarized in Table 5.2-12.

Monitoring of coolant for radiation in the Residual Heat Removal (RHR) and Reactor Water Cleanup (RWCU) heat exchangers satisfies position C.4 of the Regulatory Guide. (For system details see Sections 7.6.1.2 and 11.5.)

The three methods differ in sensitivity and response time. Position C.5 requires the leak detection system be able to detect a leakage rate of 1 gpm in less than 1 hour. See Section 7.6.2.4 for further discussion.

The leakage detection system instruments listed in Table 7.6-2 have been evaluated and are capable of performing their functions following an operating basis seismic event. The drywell airborne particulate monitoring channel will remain functional following a safe shutdown earthquake. This satisfies position C.6 of Regulatory Guide 1.45.

Leakage detection indicators and alarms are provided in the main control room. This satisfies C.7 for the NSSS scope of supply. Procedures are developed for converting the various indications to a common leakage equivalent for the operators to satisfy remainder of C.7.

The leakage detection systems are equipped with provisions to permit testing for operability and calibration during operation by the following methods:

a. Continuous monitoring of sump level compared to flow rates into sump,
b. Operability checked by comparing one method to another,
c. Simulation of signals into trip monitors, and
d. Channel "A" against Channel "B" of the same method.

Thus position C.8 is satisfied.

Limiting conditions for identified and unidentified leakage are established as 20 gpm and 5 gpm respectively, thus satisfying position C.9.

Specific Evaluation Reference:

See Sections 5.2.5 and 7.6.2.4.

Similar Application Reference:

Similar application was used for LaSalle.

Regulatory Guide 1.46, Revision 0, May 1973

Protection Against Pipe Whip Inside Containment

Regulatory Guide Intent:

Regulatory Guide 1.46 describes an acceptable basis for selecting the design locations and orientations of postulated breaks in fluid system piping within the reactor containment and for determining the measures that should be taken for restraint against pipe whipping that may result from such breaks.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design, and/or equipment used in this facility is in compliance with the intent of this regulatory guide through the incorporation of the alternate approach cited.

General Compliance or Alternate Approach Assessment:

This regulatory guide is applicable to the recirculation pipe lines.

The design of the containment structure, component arrangement, Class 1 pipe runs, pipe whip restraints and compartmentalization was done in consonance with the acknowledgment of protection against dynamic effects associated with postulated rupture of piping. Analytically sized and positioned pipe whip restraints were engineered to preclude damage based on the pipe break evaluation.

Pipe whip requirements for fluid system piping within the primary containment that, under normal operation, has service temperature greater than 200°F or pressures greater than 275 psig, complied with ANS N176, "Design Basis for Protection Against Pipe Whip," and Regulatory Guide 1.46 except as delineated in the following criteria for no breaks in Class 1 piping:

a. If Equation 10 of NB-365301, ASME Code Section III results in $S < 2.4\ S_m$ for ferritic or austenitic steels, no other requirements need be met. Stress range should be calculated between any two load sets (including zero load set) according to NB-3600 for upset and on operating basis earthquake (OBE) event transient;

b. If Equation 10 results in $2.4 < S < 3.0\ S_m$ for ferritic or austenitic steels, the cumulative usage factor, U, calculated on the bases of Equation 14 of NB-3653.6, must be less than 0.1; and

c. If Equation 10 results in $S > 3.0\ S_m$ for ferritic or austenitic steels, then the stress value in Equations 12 and 13 of NB-3653.6 must not be greater than $2.4\ S_m$.

Specific Evaluation Reference:

See Section 3.6.

Similar Application Reference:

Similar application was used in GESSAR.

Regulatory Guide 1.47, Revision 0, May 1973

Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems.

Regulatory Guide Intent:

This guide describes an acceptable method of complying with the requirements of IEEE 279-1971 and Appendix B to 10 CFR 50.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design, and/or equipment used in this facility is in compliance with the intent of the regulatory guide through the incorporation of the alternate approach cited.

General Compliance or Alternate Approach Assessment:

Each safety-related system described in Sections 7.2, 7.3, 7.4, and 7.6 is provided with an automatically or operator initiated system level bypass and inoperability annunciator.

The system level annunciators are located with the associated system controls and indications on main control room panels.

In addition to system level annunciation, component and channel level annunciators are provided on other panels either in the control room near system controls or locally near affected equipment, to indicate the cause of the system bypass or inoperability.

A switch is provided for manual actuation of each system level annunciator to allow display of those bypass or inoperable conditions which are not automatically indicated.

Typically, the following bypasses or inoperabilities cause actuation of system level (and component level) annunciation for the affected system:

a. Pump motor breaker not in operate position,

b. Loss of pump motor control power,

c. Loss of motor-operated valve control power/motive power,

d. Logic power failure,

e. Logic in test,

f. Position of remote manual valves which do not receive automatic alignment signals, and

g. Bypass or test switches actuated.

Auxiliary supporting system inoperability or bypass resulting in the loss of other safety-related systems will cause actuation of system level annunciators for the auxiliary supporting system as well as those safety-related systems affected.

Specific Evaluation Reference:

See Section 7.1.2.4.

Similar Application Reference:

Similar application was used for LaSalle.

Regulatory Guide 1.48, Revision 0, May 1973

Design Limits and Loading Combinations for Seismic Category I Fluid System Components.

Regulatory Guide Intent:

Regulatory Guide 1.48 provides acceptable design limits and appropriate combinations of loadings associated with normal operation, postulated accidents, and specified seismic events for the design of the Seismic Category I fluid system components.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design, and/or equipment used in this facility is in compliance with the intent of this regulatory guide through the incorporation of the alternate approach cited.

General Compliance or Alternate Approach Assessment:

For a comparison of NSSS with Regulatory Guide 1.48, see the attached tabulation.

The design basis was representative of good industry practices at the time of design, procurement, and manufacture and is shown to be in general agreement with requirements of Regulatory Guide 1.48, with the following clarifications:

a. The probability of an OBE of the magnitude postulated for CGS is consistent with its classification as an emergency event. However, for design conservatism, loads due to the OBE vibration motion have been included under upset conditions; loads due to the OBE vibratory motion plus associated transients, such as a turbine trip, have been considered in the equipment design under emergency conditions consistent with the probability of the OBE occurrence; and

b. The use of increased stress levels for Class 2 components is consistent with industry practice as specified in ASME Code Section III.

Specific Evaluation Reference:

See Section 3.9.3.

Similar Application Reference:

Similar application was used for LaSalle.

COMPARISON WITH REGULATORY GUIDE 1.48

		NRC Regulatory Guide 1.48			Columbia Generating Station			
Component	Plant Condition	Loading Combination[1/]	Design Limit	Regulatory Guide Paragraph	Loading Combination[(f)]	Code allowable Stresses	ASME Section III Reference	How CGS Compares With NRC Regulatory Guide 1.48
Class 1 vessels	Upset (U)	(NPC or UPC) + 0.5 SSE	NB-3223	1.a	(NPC or UPC), 0.5 SSE	3.0Sm (includes secondary stresses)	NB-3223	Reflects industry position
	Emergency (E)	EPC	NB-3224 2/	1.b	EPC, 0.5 SSE + transient	1.8Sm	NB-3224	
	Faulted (F)	NPC + SSE + DSL	NB-3225	1.c	NPC + SEE + DSL	App.F-Sec. III	NB-3225	
Class 1 piping	U	(NPC or UPC) + 0.5 SSE	NB-3654	1.a	(NPC or UPC), 0.5 SSE	3.0Sm (includes secondary stresses)	NB-3654	Reflects industry position
	E	EPC	NB-3655 2/	1.b	EPC, 0.5 SSE + transient	2.25Sm	NB-3655	
	F	NPC + SSE + DSL	NB-3656	1.c	NPC + SSE + DSL	3.0Sm	NB-3656	
Class 1 pumps (inactive)	U	(NPC or UPC) + 0.5 SSE	NB-3223[5/]	2.a	(NPC or UPC), 0.5 SSE	1.65Sm	NB-3223	Reflects industry position
	E	EPC	NB-3224 1/	2.b	EPC, 0.5 SSE + transient	1.8Sm	NB-3224	
	F	NPC + SSE + DSL	NB-3225	2.c	NPC + SSE + DSL	App. F-Sect. III	NB-3225	
Class 1 pumps (active)	U	(NPC or UPC) + 0.5 SSE	NB-3222 5/	4.a.1	(NPC or UPC), 0.5 SSE	Not applicable	Not applicable	Not applicable
	E	EPC	NB-3222 6/	4.a.2	EPC			
	F	NPC + SSE + DSL	NB-3222 7/ 8/	4.a.3	NPC + SSE + DSL			
Class 1 valves (inactive) by analysis	U	(NPC or UPC) + 0.5 SSE	NB-3223[5/]	2a	(NPC or UPC), 0.5 SSE	Not applicable	Not applicable	Not applicable
	E	EPC	NB-3224 4/	2.b	EPC			
	F	NPC + SSE + DSL	NB-3225[2/]	2.c	NPC + SSE + DSL			
Class 1 valves (inactive) designed by either std. or alternative design rules	U	(NPC or UPC) + 0.5 SSE	1.1 Pr	3.a	(NPC or UPC), 0.5 SSE	1.1 Pr	NB-3525	Reflects industry position
	E	EPC	1.2 Pr	3.b	EPC, 0.5 SSE + transient	1.2 Pr	NB-3526	
	F	NPC + SSE + DSL	1.5 Pr	3.c	NPC + SSE + DSL	1.5 Pr	NB-3527	
Class 1 valves (active) by analysis	U	(NPC or UPC) + 0.5 SSE	NB-3222 5/	4.a.1	(NPC or UPC, 0.5 SSE	Not applicable	Not applicable	Not applicable
	E	EPC	NB-3222 6/	4.a.2	EPC			
	F	NPC + SSE + DSL	NB-3222 7/ 8/	4.a.3	NPC + SSE + DSL			
Class 1 valves (active) designed by std. or alternative design rules	U	(NPC or UPC) + 0.5 SSE	1.0 Pr	5.a.1	(NPC or UPC), 0.5 SSE	1.0 Pr	NB-3525	Reflects industry position
	E	EPC	1.0 Pr 6/	5.a.2	EPC	1.0 Pr (a)	NB-3526	
	F	NPC + SSE + DSL	1.0 Pr	5.a.3	NPC + SSE + DSL	1.0 Pr	NB-3527	

COMPARISON WITH REGULATORY GUIDE 1.48 (Continued)

		NRC Regulatory Guide 1.48			Columbia Generating Station			
Component	Plant Condition	Loading Combination[1/]	Design Limit	Regulatory Guide Paragraph	Loading Combination[(f)]	Code Allowable Stresses	ASME Section III Reference	How CGS Compares With NRC Regulatory Guide 1.48
Class 2 & 3 vessels	U	(NPC or UPC) + 0.5 SSE	1.1S	6.a	(NPC or UPC), 0.5 SSE	$\sigma m = 1.1S$	code case 1607	Faulted condition,
(Division 1) of section	E	EPC	1.1S 9/	6.b	EPC,0.5 SSE + transient	(c)	NC/NB	NRC more conservative,
VIII of the ASME Code	F	NPC + SSE + DSL	1.5S	6.c	NPC + SSE + DSL	$\sigma m = 2.0S$	3321.1(b)	reflects industry position
Class 2 vessels	U	(NPC or UPC) + 0.5 SSE	NB-3223	7.a	(NPC or UPC), 0.5 SSE	Not applicable	Not applicable	Not applicable
(Division 2) of section	E	EPC	NB-3224 2/	7.b	EPC			
VIII of the ASME Code	F	NPC + SEE + DSL	NB-3225	7.c	NPC + SSE + DSL			
Class 2 & 3	U	(NPC or UPC) + 0.5 SSE	NC3611.1(b)(4)(c)(b)(1)	8.a	(NPC or UPC), 0.5 SSE	1.2 Sh	NC/ND 3611.3(b)	NRC more conservative,
piping	E	EPC	NC3611.1(b)(4)(c)(b)(1) 10/	8.a	EPC,0.5 SSE + transient	1.8 Sh	NC/ND 3611.3(c)	Reflects industry
	F	NPC + SSE + DSL	NC3611.1(b)(4)(c)(b)(2)	8.b	NPC + SSE + DSL	2.4 Sh	(4)(b) (b) code case1606, NC/ND 3611.3(d) [see note (b)]	position
Class 2 & 3 pumps (inactive)	U	(NPC or UPC) + 0.5 SSE	$\sigma m \le 1.1S \ge \frac{\sigma m + \sigma b}{1.5}$	9.a	(NPC or UPC), 0.5 SSE	Not applicable	Not applicable	Not applicable
	E	EPC	$\sigma m \le 1.1S \ge \frac{\sigma m + \sigma b}{1.5}$	9.a	EPC			
	F	NPC + SEE + DSL	$\sigma m \le 1.2S \ge \frac{\sigma m + \sigma b}{1.5}$	9.b	NPC + SEE + DSL			
Class 2 & 3 pumps (inactive)	U	(NPC or UPC) + 0.5 SSE	$\sigma m \le 1.1S \ge \frac{\sigma m + \sigma b}{1.5}$	10.a	(NPC or UPC), 0.5 SSE	$\sigma m = 1.1S$	Code case 1636, NC/ND3423	Reflects industry position
	E	EPC	$\sigma m \le 1.1S \ge \frac{\sigma m + \sigma b}{1.5}$ 11/	10.a	EPC,0.5 SSE + transient	(a) (c)	[see note (b)]	
	F	NPC + SSE + DSL	$\sigma m \le 1.1S \ge \frac{\sigma m + \sigma b}{1.5}$	10.a	NPC + SSE + DSL	$\sigma m = 1.2S$		
Class 2 & 3 valves	U	(NPC or UPC) + 0.5 SSE	1.1 Pr	11.a	(NPC or UPC), 0.5 SSE	$\sigma m = 1.1S$	Code case1636,	Equally conservative
(inactive)	E	EPC	1.1 Pr	11.a	EPC,0.5 SSE + transient	(c)	NC/ND3621	
	F	NPC + SSE + DSL	1.2 Pr	11.b	NPC + SSE + DSL	$\sigma m = 2.0S$	[see note (b)]	
Class 2 & 3 valves	U	(NPC or UPC) + 0.5 SSE	1.0 Pr	12.a	(NPC or UPC), 0.5 SSE	$\sigma m = 1.1S$	Code case1636,	Equally conservative
(active)	E	EPC	1.0 Pr 11/	12.a	EPC,0.5 SEE + transient	(a)	NC/ND3621	(e)
	F	NPC + SSE + DSL	1.0 Pr	12.a	NPC + SSE + DSL	$\sigma m = 1.2S$ (c)	[see note (b)]	

COMPARISON WITH REGULATORY GUIDE 1.48 (Continued)

NOTES

Numerical indicators (e.g., 1/) in the regulatory guide portion of the table correspond to the footnotes of Regulatory Guide 1.48. Alphabetical indicators in CGS portion of table (or comparative column) correspond to the following:

[a]In addition to compliance with the design limits specified, assurance of operability under all design loading combinations shall be in accordance with Section 3.9.3.2.

[b]Referenced paragraphs of code currently in course of preparation.

[c]The design limit for local membrane stress intensity or primary membrane plus primary bending stress intensity is 150% of that allowed for general membrane (except as limited to 2.4S for inactive components under faulted condition). See Section 3.9.5.2.

[d]Not used.

[e]Inactive limits may be used since operability will be demonstrated in accordance with Section 3.9.3.2.

[f]When selecting plant events for evaluation, the choice of the events to be included in each plant condition is selected based on the probability of occurrence of the particular load combination. The combination of loads are those identified in Table 3.9-2.

LEGEND:

UPC = upset plant conditions
NPC = normal plant conditions
EPC = emergency plant conditions
DSL = dynamic system loading
SSE = safe shutdown earthquake

Regulatory Guide 1.49, Revision 1, December 1973

Power Levels of Nuclear Power Plants.

Regulatory Guide Intent:

Regulatory Guide 1.49 requires that the proposed licensed power level be restricted to a reactor core power level of 3800 MWt or less and that analyses and evaluations in support of the application should be made at 1.02 times the proposed licensed power level.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design, and/or equipment used in this facility is in full compliance with this regulatory guide.

General Compliance or Alternate Approach Assessment:

The rated thermal power for the CGS reactor is 3486 MWt. The safety analyses and evaluations were made for a CGS power level of 3556 MWt which is 1.02 times the rated power. This complies with the subject guide requirements.

Specific Evaluation Reference:

See Section 1.1.

Similar Application Reference:

Similar applications were used for Browns Ferry Units 1, 2, and 3.

Regulatory Guide 1.50, Revision 0, May 1973

Control of Preheat Temperature for Welding of Low-Alloy Steel

Regulatory Guide Intent:

This guide delineates preheat temperature control requirements and welding procedure qualifications supplementing those in ASME Sections III and IX.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design, and/or equipment used in this facility is in compliance with the intent of this regulatory guide through the incorporation of the alternate approach cited.

General Compliance or Alternate Approach Assessment:

The use of low-alloy steel is restricted to the reactor pressure vessel. Other ferritic components in the RCPB are fabricated from carbon steel materials.

Preheat temperatures employed for welding of low-alloy steel meet or exceed the requirements of ASME Section III. Components were either held for an extended time at preheat temperature to ensure removal of hydrogen, or preheat was maintained until postweld heat treatment. The minimum preheat and maximum interpass temperature were specified and monitored.

All welds were nondestructively examined by radiographic methods. In addition, a supplemental ultrasonic examination was performed.

By meeting and/or exceeding the recommendation of the ASME Code, the intent of the regulatory guide is satisfied even though the design was significantly developed prior to issuance of the specific guide wording.

Specific Evaluation Reference:

See Section 5.2.3.

Similar Application Reference:

Similar application was used for LaSalle.

Regulatory Guide 1.53, Revision 0, June 1973

Application of the Single-Failure Criterion to Nuclear Power Plant Protection Systems

Regulatory Guide Intent:

Regulatory Guide 1.53 requires that protection systems meet the requirements of Section 4.2 of IEEE 279-1971, which is also required by ANSI-N 42.7-1972 in that any single failure within the protection systems shall not prevent proper protective action at the system level when required. This guide provides guidance on an acceptable method of complying with this requirement.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design, and/or equipment used in this facility is in full compliance with this regulatory guide.

General Compliance or Alternate Approach Assessment:

Compliance is achieved by specifying, designing, and constructing the engineered safeguards systems to meet the single failure criterion, Section 4.2 of IEEE 279-1971, "Criteria for Protection Systems for Nuclear Power Generating Stations," and IEEE 379-1972, "IEEE Trial-Use Guide for the Application of the Single-Failure Criterion to Nuclear Power Generating Station Protection Systems."

This regulatory guide applies to the following NSSS supplied protection systems: reactor protection system (RPS), ECCS, and PCRVICS.

The reactor protection system has separate and redundant instrument channels, logic, and actuation circuits to ensure that the single failure criterion is met. The PCRVICS is similarly designed.

The ECCS is divided into the ADS, HPCS, LPCS and RHR (LPCI) which meets the single failure criterion on a network basis.

Specific Evaluation Reference:

See Sections 7.2.2.2 and 7.3.2.1.2.

Similar Application Reference:

Similar application was used for LaSalle.

Regulatory Guide 1.54, Revision 0, June 1973

Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants.

Regulatory Guide Intent:

This guide describes an acceptable method of complying with QA requirements for protective coatings.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

The identified BWR Quality Assurance Program used in this facility reflects compliance with the provisions of NRC regulations and the NRC regulatory guide or NRC-approved alternate position.

General Compliance or Alternate Approach Assessment:

Reference compliance assessment for Regulatory Guide 1.28.

Specific Evaluation Reference:

Information was provided at the PSAR stage.

Similar Application Reference:

Similar application has not been used for other projects.

Regulatory Guide 1.56, Revision 0, June 1973

Maintenance of Water Purity in Boiling Water Reactors

Regulatory Guide Intent:

This guide describes an acceptable method of implementing GDC 13, 14, 15, and 31 with regard to minimizing the probability of corrosion-induced failure of the RCPB in BWRs by maintaining acceptable purity levels in the reactor coolant and acceptable instrumentation to determine the condition of the reactor coolant.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design, and/or equipment used in this facility is in compliance with the intent of this regulatory guide through the incorporation of the alternate approach cited.

General Compliance or Alternate Approach Assessment:

Materials in the primary system are primarily Type 304 stainless steel and Zircaloy cladding. The reactor water chemistry limits have been established to provide an environment favorable to these materials. Design and Licensee Controlled Specifications (LCS) limits are placed on conductivity and chloride concentrations. Operationally, the conductivity is limited because it can be continuously and reliably measured and gives an indication of abnormal conditions and the presence of unusual materials in the coolant. Chloride limits are specified to prevent stress corrosion cracking of stainless steel.

The water quality requirements are further supported by GE topical report NEDO-10899.

Specific Evaluation Reference:

See Section 5.2.3.

Similar Application Reference:

Similar application was used for LaSalle.

Regulatory Guide 1.58, Revision 0, August 1973

Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel

Regulatory Guide Intent:

This guide describes an acceptable method of complying with the NRC's regulations on qualification of nuclear power plant inspection, examination and testing personnel.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

The identified BWR Quality Assurance Program used in this facility reflects compliance with the provisions of NRC regulations and the NRC regulatory guide or NRC-approved alternate position.

General Compliance or Alternate Approach Assessment:

Reference compliance assessment for Regulatory Guide 1.28.

Specific Evaluation Reference:

Information was provided at the PSAR stage. Compliance is discussed in the OQAPD.

Similar Application Reference:

Similar application has not been used in other plants.

Regulatory Guide 1.60, Revision 1, December 1973

Design Response Spectra for Seismic Design of Nuclear Power Plants.

Regulatory Guide Intents:

This guide delineates procedures for defining response spectra for designing Seismic Category I structures, systems, and components.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design, and/or equipment used in this facility is in compliance with the intent of this regulatory guide through the incorporation of the alternate approach cited.

General Compliance or Alternate Approach Assessment:

The input loadings for the seismic analysis of the CGS plant structures were given in terms of response spectra based on data available on earthquake acceleration time history records which was accepted industry practice at the time of the CGS design. This method was acceptable to the NRC prior to the issuance of this regulatory guide because no other guidance was available.

Specific Evaluation Reference:

See Section 3.7.1.1.

Similar Application Reference:

Similar application was used for LaSalle.

Regulatory Guide 1.61, Revision 0, October 1973

Damping Values for Seismic Design of Nuclear Power Plants

Regulatory Guide Intent:

This guide delineates damping values that should be applied to modal dynamic analysis of Seismic Category I elements.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design, and/or equipment used in this facility is in compliance with the intent of this regulatory guide through the incorporation of the alternate approach cited.

General Compliance or Alternate Approach Assessment:

The damping values used in the seismic analysis conform to the data available on this at the time the analysis was performed which was the practice accepted by industry and the NRC at the time of the CGS design.

The values used in Table 3.7-1 are less than those given by the regulatory guide. The calculated responses are therefore conservative.

Specific Evaluation Reference:

See Section 3.7.1.3.

Similar Application Reference:

Similar application was used for LaSalle.

Regulatory Guide 1.62, Revision 0, October 1973

Manual Initiation of Protective Actions.

Regulatory Guide Intent:

Regulatory Guide 1.62 requires that manual initiation of each protective action at the system level be provided, that such initiation accomplishes all actions performed by automatic initiation, and that protective action at the system level go to completion once manually initiated. In addition, manual initiation should be by switches readily accessible in the control room, and a minimum of equipment should be used in common with automatically initiated protective action.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design, and/or equipment used in this facility is in full compliance with this regulatory guide.

General Compliance or Alternate Approach Assessment:

Means are provided for manual initiation of primary containment and reactor vessel isolation control system (NSSS only), ECCS, and reactor protection system scram at the system level through the use of armed push buttons, as described below:

Action Initiated	Number of Switches
Primary containment and reactor vessel isolation (NSSS Only)	Four, two in Division 1 and two in Division 2
ADS	Four, two in Division 1 and two in Division 2
HPCS	One switch in Division 3
RHR (loop A)/LPCS	One switch in Division 1
RHR (loop B)/RHR (loop C)	One switch in Division 2
Reactor protection system (SCRAM)	Four, two in Division 1 and two in Division 2

Operation of these switches accomplishes the initiation of all actions performed by the automatic initiation circuitry.

The amount of equipment common to both manual and automatic initiation of the above function is kept to a minimum through implementation of manual activation as close as possible to the final devices actuators (relays, scram contractor) of the protection system. No failure in the manual, automatic or common portions of the protection system will prevent initiation of a given function by manual or automatic means.

Manual initiation of any of the above functions, once initiated, goes to completion as required by IEEE 279-1971, Section 4.16.

Specific Evaluation Reference:

See Sections 7.2.2.3 and 7.3.2.1.3.

Similar Application Reference:

Similar application has not been used for other projects.

Regulatory Guide 1.64, Revision 2, June 1976

Quality Assurance Requirements for the Design of Nuclear Power Plants

Regulatory Guide Intent:

This guide describes an acceptable method of complying with the NRC's QA requirements for the design of the nuclear power plants.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

The identified BWR Quality Assurance Program used in this facility reflects compliance with the provisions of NRC regulations and the NRC regulatory guide or NRC-approved alternate position.

General Compliance or Alternate Approach Assessment:

Reference compliance assessment for Regulatory Guide 1.28.

Specific Evaluation Reference:

Information was provided at the PSAR stage. Compliance is discussed in the OQAPD.

Similar Application Reference:

Similar application has not been used for other projects.

Regulatory Guide 1.65, Revision 0, October 1973

Materials and Inspection for Reactor Vessel Closure Studs.

Regulatory Guide Intent:

Regulatory Guide 1.65 defines acceptable materials and testing procedures with regard to reactor vessel closure stud bolting for light-water-cooled reactors.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design, and/or equipment used in this facility is in compliance with the intent of this regulatory guide through the incorporation of the alternate approach cited.

General Compliance or Alternate Approach Assessment:

The reactor pressure vessel closure studs are SA-540 Grade B23 or 24 (AISI4340) and have a maximum ultimate tensile strength of 170 ksi. Additionally, specified bolting material must have Charpy V notch impact properties of 45 ft-lb minimum with 25 mils lateral expansion. Nondestructive examination before and after threading is specified to be in accordance with subarticle NB-2580 ASME Section III, which complies with regulatory position C.2. Subsequent to fabrication, the studs are manganese phosphate coated and are lubricated with a graphite/alcohol or a nickel powder base lubricant.

In relationship to regulatory position C.2.b, the bolting materials were ultrasonically examined after final heat treatment and prior to threading, as specified. The specified requirement for examination according to ASME Section II Recommended Practice SA-388 was complied with. The specific procedures approved for use in practice are judged to ensure comparable material quality and, moreover, are considered adequate on the basis of compliance with the applicable requirements of ASME Section III paragraph NB-2585.

Additionally, straight beam examination was performed on 100% of cylindrical surfaces, and from both ends of each stud using a 3/4 maximum diameter transducer. In addition to the code required notch, the reference standard for the radial scan contained a 0.5-in. diameter flat bottom hole with a depth of 10% of the thickness, and the end scan standard contained a 0.25-in. diameter flat bottom hole 0.5-in. deep. Also, angle beam examination was performed on the outer cylindrical surface of nuts and washers per ASME SA-388 in both an axial and circumferential direction. Any indication greater than the indication from the applicable calibration feature is unacceptable. A distance-amplitude correction curve per NB-2585 is used for the longitudinal wave examination. Surface examinations were performed on the studs and nuts after final heat treatment and threading, as specified in the Regulatory Guide, in accordance with NB-2583 of ASME Code Section III, 1971 Edition through November 1971 Addenda.

In relationship to regulatory position C.2, GE practice allows exposure of stud bolting surfaces to high purity fill water; nuts and washers are stored dry during refueling.

Specific Evaluation Reference:

See Section 5.3.1.7.

Similar Application Reference:

Similar application was used for LaSalle.

Regulatory Guide 1.66, Revision 0, October 1973

Nondestructive Examination of Tubular Products.

Regulatory Guide Intent:

This guide describes a method of implementing requirements acceptable to NRC regarding nondestructive examination requirements of tubular products used in the RCPB.

Applicable Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design, and/or equipment used in this facility is in compliance with the intent of this regulatory guide through the incorporation of the alternate approach cited.

General Compliance or Alternate Approach Assessment:

Wrought tubular products were supplied in accordance with applicable ASTM/ASME material specifications. These specifications require a hydrostatic test on each length of tubing. Additionally, the specification for the tubular product used for CRD housings specified ultrasonic examination to paragraph NB-2550 of ASME Code Section III.

These RCPB components met the requirements of ASME Codes existing at time of placement of order which predated Regulatory Guide 1.66. At the time of the placement of the orders, 10 CFR 50, Appendix B requirements and ASME code requirements assured adequate control of quality for the products.

This regulatory guide was withdrawn on September 28, 1977, by the NRC because the additional requirements imposed by the guide were satisfied by the ASME Code Section III.

Specific Evaluation Reference:

See Sections 4.5.2.3 and 5.2.3.3.

Similar Application Reference:

Similar application was used for LaSalle.

Regulatory Guide 1.67, Revision 0, October 1973

Installation of Overpressure Protection Devices

Regulatory Guide Intent:

This regulatory guide describes a method acceptable to the NRC staff for implementing GDC 1 with regard to the design of piping for safety valve and relief valve stations which have open discharge systems with limited discharge pipes and which have inlet piping that neither contains a water seal nor is subject to slug flow of water on discharge of the valves.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

Identified RHR shutdown suction line thermal relief piping is located between the containment isolation valves. However, the intent of the regulatory guide does not apply due to the very short duration and small discharge of the thermal relief function.

General Compliance or Alternate Approach Assessment:

This regulatory guide is not considered to be applicable to this piping due to the small size and very short operation time of the valve (0.75 in. x 1 in.). The only purpose of the valve is to relieve the excess pressure caused by the difference of thermal expansion between the pipe and the water contained between the containment isolation valves.

Specific Evaluation Reference:

See Section 3.9.3.1.14.

Similar Application Reference:

Not applicable.

Regulatory Guide 1.68, Revision 0, November 1973

Preoperational and Initial Startup Test Programs for Water-Cooled Power Reactors

Regulatory Guide Intent:

Regulatory Guide 1.68 describes the requirements for the initial startup test programs. This regulatory guide is applicable to such activities as precritical tests and low-power tests.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design, and/or equipment used in this facility is in compliance with the intent of this regulatory guide through the incorporation of the alternate approach cited.

General Compliance or Alternate Approach Assessment:

The following discussion describes the alternate acceptable approaches for specific conformance to this regulatory guide.

The format of the CGS test procedures is different from that of the guide, but since the content specifies the required elements, the procedures are in compliance.

The reference sections refer to those of the regulatory guide. Those sections not listed are in compliance.

Section C.2.b: Operational limitations for the protection of public health and safety are included in the Technical Specifications for the plant. The General Electric startup instructions contain notes of caution which supplement the Technical Specifications. The Technical Specifications should be the instrument for describing operational (including testing) limitations. Therefore, the identification of "safety precautions" in test procedures should be limited to those items which, if not observed, could lead to reduction of system safety performance below expected levels and not the minor procedural and test details which would not cause such a reduction.

Section C.2.c: The generic simulation test appearing in Chapter 14 should appear by reference in preoperational and initial startup test programs where onsite full *simulation tests are not possible. The guide wording would change to "… less than full simulation should be provided or referenced for test where full…"*

Appendix A, Section C.2.h: The comparison of critical control rod pattern with predicted patterns (Appendix A, Section C.2.d) provides required knowledge of effective overall rod worth. Individual control rod calibrations cannot be performed in a meaningful manner in a large multirodded BWR. Therefore, this part of the guide is not applicable to BWRs.

Appendix A, Section C.2.i: The functional requirement of the reactor head cooling system design is required at operating pressures less than or equal to 135 psig. Therefore, for this paragraph to be applicable "(135 psig)" should be part of last sentence.

Appendix A, Section D.2.a: The high-pressure coolant injection (HPCI) has been replaced by an HPCS system. Due to the configuration of the sprays directly on the core, this system cannot be operated at power. The HPCS injection/core spray is demonstrated during the preoperational test program.

Appendix A, Section D.2.b: Friction tests are performed on four drives at rates pressure.

Appendix A, Section D.2.f: It is necessary to make more than two calibrations and, therefore, it is not appropriate to limit the test to 50% and 100% power levels.

Appendix A, Section D.2.g: At least six chemical analyses of fluid system are necessary; therefore, the limitations of 25%, 50%, 75%, and 100% are not appropriate.

Appendix A, Section D.2.1: Since this plant design does not include an emergency condenser, this section is not appropriate.

Appendix A, Section D.2.n: Control rod calibration in a large multirodded BWR has not been found to provide meaningful data. Any safety-related problems associated with control rods would be discovered during safety related testing, and therefore, this section is not appropriate.

Appendix A, Section D.2.p: Since the main steam valve function tests are conducted at a minimum of six power and flow conditions, the limitations of 25%, 50%, and 75% are not appropriate.

Appendix A, Section D.2.s and t: Turbine trip and generator trip have essentially the same effect on the reactor and safety related system actuation. Sections D.2.s and D.2.t should be combined into one test.

Appendix A, Section D.2.y: Minimum critical heat flux ratio (MCHFR) is an obsolete limit that has been replaced with minimum critical power ratio (MCPR). Core performance evaluation tests must be performed at every test condition.

Appendix A, Section D.2.aa: Comparison tests are made throughout the test program, and therefore, limitations of 25%, 50% and 100% are not appropriate.

Appendix C, Section B.2.d: Functionally testing the associated control rod immediately following installation of each fuel cell is not appropriate. Functional testing of all control rods after fuel loading and prior to startup to critical procedures is applicable.

Appendix A, Section A.5.a: The "demonstration of water injection for a LOCA" is an ECCS test. Therefore, "demonstration of water injection for a loss-of-coolant accident" is not within the scope of the reactor coolant makeup system test.

Appendix A, Section C.2.c: The "calibration of intermediate range monitor with power" is not meaningful due to local control rod effects.

Appendix A, Section D.2.w: Feedwater pump trip should be performed to check recirculation pump runback.

Appendix C, Section B.1.b: Poison curtains are not applicable since they are not used in this plant.

Appendix C, Section B.2.a: Poison curtains are not applicable.

Appendix C, Section B.3.c: The insertion of locked control rods is excluded in any withdrawal sequence.

Appendix D, Section D.2.0: The rod pattern exchange is not a part of the Startup Power Ascension Program since it does not involve the approach of any safety margin or operating limit. The rod pattern exchange procedure at power is part of the Nuclear Performance Evaluation Procedure and will be performed during the fuel cycle as necessary. The simultaneous trip of both recirculation pumps is not performed at 100% of rated power. The analysis of this event (see Section 15.3.1) indicates there is no decrease in the MCPR and therefore, it does not involve the approach of any safety margin or operating limit.

Specific Evaluation Reference:

See Section 14.2.

Similar Application Reference:

Similar application was used for Brunswick 1 and Browns Ferry 3.

Regulatory Guide 1.70, Revision 2, September 1975

Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants

Regulatory Guide Intent:

This guide describes the minimum acceptable requirements for format and content of Safety Analysis Reports.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design, and/or equipment used in this facility is in full compliance with this regulatory guide or through the incorporation of the NRC approved alternate approach cited.

General Compliance or Alternate Approach Assessment:

The NSSS scope of supply inputs include all the appropriate scope responsibilities and information required in Regulatory Guide 1.70, Revision 2, in both format and content, except as described below. Appendix A of NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel (GESTAR II) (most recent approved revision referenced in the COLR), provides a road map for incorporating nuclear fuel design and analysis characteristics described in GESTAR II into the FSAR. GESTAR II is consistent with Regulatory Guide 1.70, Revision 3.

Specific Evaluation Reference:

For Regulatory Guide 1.70, Revision 2, see NSSS scope of supply portions of this FSAR.

For Regulatory Guide 1.70, Revision 3, see Sections 4.1, 4.2, 4.3 and 4.4.

Similar Application Reference:

Similar application was used for Grand Gulf 1 and 2 and Susquehanna 1 and 2.

Regulatory Guide 1.71, Revision 0, December 1973

Welder Qualification for Areas of Limited Accessibility

Regulatory Guide Intent:

Regulatory Guide 1.71 requires that weld fabrication and repair for wrought low-alloy and high-alloy steels or other materials such as static and centrifugal castings and bimetallic joints should comply with fabrication requirements of Section III and Section IX of the ASME B&PV Code. It also requires additional performance qualifications for welding in areas of limited access.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design, and/or equipment used in this facility is in compliance with the intent of this regulatory guide through the incorporation of the alternate approach cited.

General Compliance or Alternate Approach Assessment:

All ASME Section III welds were fabricated in accordance with the requirements of Section III and IX of the ASME B&PV Code. There are few restrictive welds involved in the fabrication of BWR components. Welder qualification for welds with the most restrictive access was accomplished by mock-up welding. Mock-ups were examined with radiography or sectioning.

All reactor pressure boundary welding was performed in accordance with ASME Section IX. Reactor internal component welding was performed in accordance with ASME Section IX or appropriate AWS requirements.

Specific Evaluation Reference:

See Section 5.2.3.

Similar Application Reference:

Similar application was used for Zimmer and LaSalle.

Regulatory Guide 1.73, Revision 0, January 1974

Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants.

Regulatory Guide Intent:

Regulatory Guide 1.73 endorses the requirements of IEEE 382-1972, "Trial-Use Guide for Type Test of Class 1 Electric Valve Operators for Nuclear Power Generating Station." Regulatory position stipulations are also included.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design, and/or equipment used in this facility is in full compliance with this regulatory guide.

General Compliance or Alternate Approach Assessment:

This regulatory guide is applicable to the recirculation system gate valve and the HPCS injection valve motor operators.

These valve operators have been tested in accordance with the test sequence outlined in Section 4.5.2 of the IEEE 382-1972. The qualifying tests have been made under environmental conditions (temperature, pressure, humidity, radiation) that are at least as severe as those that the valve operator will be exposed to during and following a DBA (LOCA).

Specific Evaluation Reference:

See Section 3.11.

Similar Application Reference:

Similar application was used for LaSalle.

Regulatory Guide 1.74, Revision 0, February 1974

Quality Assurance Terms and Definitions

Regulatory Guide Intent:

This guide identifies quality assurance terms and acceptable definitions.

Application Assessment;

Assessed capability in design.

Compliance or Alternate Approach Statement:

The identified BWR Quality Assurance Program used in this facility reflects compliance with the provisions of NRC regulations and the NRC regulatory guide or NRC-approved alternate position.

General Compliance or Alternate Approach Assessment:

Reference compliance assessment for Regulatory Guide 1.28.

Specific Evaluation Reference:

Information was provided at the PSAR stage. Compliance is discussed in the OQAPD.

Similar Application Reference:

Similar application has not been used for other projects.

Regulatory Guide 1.75, Revision 0, February 1974

Physical Independence of Electrical Systems

Regulatory Guide Intent:

This guide presents a detailed method of ensuring physical independence of electric systems, including requirements of preparation, identification, and isolation.

Application Assessment:

Assessed capability in design

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design, and/or equipment used in this facility is in compliance with the intent of this regulatory guide through the incorporation of the alternate approach cited.

General Compliance or Alternate Approach Assessment:

When evaluating the applicability of Regulatory Guide 1.75 and its attendant IEEE Standard (IEEE-384-1971), consideration should be given to the fact that design was significantly developed prior to their issuance.

The following is a point-by-point definition of the implementation of IEEE-384 as modified by Regulatory Guide 1.75 for the CGS plant. The numbers and titles in the following see those of IEEE-384.

1. Scope

   Compliance with scope.

2. Purpose

   Compliance with purpose.

3. Definitions

   All definitions apply including Regulatory Guide 1.75 except for small nomenclature aspects in C.1 and C.2 associated within floor sections.

4. General Separation Criteria

4.1 Required Separation

4.2 Equipment and Circuits Requiring Separation

The equipment and circuits requiring separation are determined and delineated early in the plant design. Distinctive identification of those equipment and circuits were not provided on specifically noted documents and drawings but the documents and drawings are identified as applying to the "protection systems."

4.3 Methods of Separation

Barriers are used to separate divisional devices and wiring. Safety system logic is implemented with relay coil to relay contact separation of multidivisional and nondivisional signals. Distance separation was provided to the extent feasible at manufacturing time. These served the purpose or intent of requirements at that time.

4.4 Compatibility with Mechanical Systems

The Class 1E equipment and circuits are specified to be located so that a failure in the mechanical systems served by the Class 1E systems does not disable redundant portions of the Class 1E systems.*

4.5 Associated Circuits

Associated circuits are treated as non-Class 1E circuits and are separated to the extent that good electrical isolation is assured. This assurance was provided without Class 1E isolators. Some physical separation is provided.

4.6 Non-Class 1E Circuits

4.6.1 Separation from Class 1E Circuits

Same as 4.5 response above.

* Information on compliance of actual installation is provided in Section 1.8.3.

4.6.2 Separation from Associated Circuits

Same as 4.5 response above.

5. Specific Separation Criteria

5.1 Cables and Raceways

To the extent that the 5.1 series of subparagraphs might be used to critique the power generation control complex (PGCC) equipment, the physical reality of the floor sections is obviously not recognized in the IEEE-384 test. However, the floor sections are inherently in accordance with the design concepts stated in these subparagraphs and therefore comply on that basis.

5.2 Standby Power Supply

Comply as applied to the Division 3 HPCS Diesel Generator.[*]

5.3 DC System

Comply as applied to the Division 3 HPCS Diesel Generator.[*]

5.4 Distribution System

Comply as applied to the Division 3 HPCS Diesel Generator.[*]

5.5 Containment Electrical Penetrations

Not in NSSS scope of supply.

5.6 Control Switch Boards

5.6.1 Location and Arrangement[†]

Class 1E equipment and circuits are located on separate control switchboards or where operationally necessary on a single control switchboard.

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[*] Division 1 and 2 power compliance is provided in Section 1.8.3.
[†] The control room structure and location as well as local control switchboard location is discussed in Section 1.8.3.

5.6.2 Internal Separation

Most of the devices requiring separation are separated by barriers. With several divisions in one panel, and for relays which must accept multidivisional signals, 6-inch separation is impossible. Therefore, separation is done on a best effort approach. Design has used the relay coil to relay contact separation to comply with the regulatory guide.

5.6.3 Internal Wiring Identification

Panel internals wiring is not color-coded, but wires are marked with their respective Connection Diagram identify at each point of termination.

5.6.4 Common Terminations

Relay coil to relay contact separation has been used.

5.6.5 Non-Class 1E Wiring

Electrical isolation is provided, though not necessarily with Class 1E isolators. Some physical separation is provided.

5.6.6 Cable Entrance

Not in NSSS scope of supply.

5.7 Instrumentation Cabinets

Compliance

5.8 Sensors and Sensor to Process Connections

Compliance

5.9 Actuated Equipment

Not in NSSS scope of supply.

Specific Evaluation Reference:

See Section 8.3.1.4.2.7

Similar Application Reference:

Application of this regulatory guide is plant unique due to NRC agreements during the various stages of licensing and scope of responsibility of design and engineering necessary to comply with the NRC interpretation. Therefore reference plants cannot be cited.

Regulatory Guide 1.84

Design, Fabrication, and Materials Code Case Acceptability, ASME Section III

Regulatory Guide Intent:

This guide lists all Section III Code Cases that the NRC has approved for use. It is updated on a regular basis to reflect the changes to the ASME Code Cases and the current position of the NRC on acceptability for use. The guide contains tables that detail the NRC acceptance requirements for current, annulled, and superseded Code Cases. Code Cases that the NRC determined to be unacceptable are listed in Regulatory Guide 1.193, "ASME Code Cases Not Approved for Use".

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

The current version of the Regulatory Guide is utilized to determine acceptable Code Cases for all new and existing plant applications. The FSAR does not track individual Code Cases and revision numbers. Not all acceptable Code Cases listed in the regulatory guide are used. The Code Cases that are utilized for Columbia are referred to in the plant design/installation documentation.

General Compliance or Alternate Approach Assessment:

Code Cases are utilized in accordance with the requirements of the regulatory guide provisions for acceptance. Section III Code Cases that are not yet endorsed may be utilized via submittal to the NRC for approval in accordance with the regulatory guide. The plant scope of supply is in full compliance with this regulatory guide.

Specific Evaluation Reference:

See Section 3.2.

Similar Application Reference:

None.

Regulatory Guide 1.85, Revision 31, 1998*

Code Case Acceptability ASME Section III Materials

Regulatory Guide Intent:

This guide provides a list of ASME materials code cases that have been generically approved by the NRC.

Code cases on this list may be used until annulled. Annulled cases are considered "active" for equipment that has been contractually committed to fabrication prior to the annulment.

This guide and later revisions require NRC approval of code cases for Class 1, 2, and 3 components.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design, and/or equipment used in this facility is in compliance with the intent of this regulatory guide through the incorporation of the alternate approach cited.

General Compliance or Alternate Approach Assessment:

The GE procedure is to obtain NRC approval of code cases on Class 1 components only. NRC approval of Class 2 and 3 code cases was not required by 10 CFR 50.55(a).

All Class 2 and 3 equipment has been designed to ASME Code or ASME approved Code Cases. This provision together with quality control requirements provide adequate safety equipment functional assurances.

Specific Evaluation Reference:

See Section 5.2.1.

* Regulatory Guide 1.85 was withdrawn in 2004. See Regulatory Guide 1.84 for NRC acceptance of current Materials Code Cases.

Similar Application Reference;

Similar application was used for LaSalle.

Regulatory Guide 1.88, Revision 2, October 1976

Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records.

Regulatory Guide Intent:

This guide describes an acceptable method of complying with the NRC's regulations for collection, storage, and maintenance of quality assurance records.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

The identified BWR Quality Assurance Program used in this facility reflects compliance with the provisions of NRC regulations and the regulatory guide or NRC-approved alternate position.

General Compliance or Alternate Approach Assessment:

Reference compliance assessment for Regulatory Guide 1.28.

Specific Evaluation Reference:

Information was provided at the PSAR stage. Compliance is discussed in the OQAPD.

Similar Application Reference:

Similar application has not been used for other projects.

Regulatory Guide 1.89, Revision 1, June 1984

Qualification of Class 1E Equipment for Nuclear Power Plants

Regulatory Guide Intent:

Regulatory Guide 1.89 Rev. 1 endorses both the requirements and recommendations of IEEE 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations." Additional regulatory position stipulations are also included.

Compliance or Alternate Approach Statement:

CGS complies with this regulatory guide for equipment requiring environmental qualification procured after February 22, 1983.

General Compliance or Alternate Approach Assessment:

For equipment requiring environmental qualification installed prior to February 22, 1983, CGS follows the guidance in NUREG-0588 Cat II.

In view of the NRC Memorandum and Order (CLI-80-21), dated May 27, 1980, all environmental qualifications of Class 1E equipment within the NSSS scope of supply was reevaluated for compliance with NUREG-0588, Category II. Where significant deviation from those guidelines was found in specific equipment qualifications, additional testing and/or analysis was performed to demonstrate the adequacy of the equipment to perform its safety-related function.

Specific Evaluation Reference:

Delineation of the degree of compliance is contained in Section 3.11.

Regulatory Guide 1.92, Revision 1, February 1976

Combination of Modes and Spatial Components in Seismic Response Analysis.

Regulatory Guide Intent:

This guide describes methods acceptable to the NRC for combining the values of the response spectrum nodal dynamic analysis and in combining maximum values (in case of time history dynamic analysis) or the representative maximum values (in case of spectrum dynamic analysis).

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design, and/or equipment used in this facility is in compliance with the intent of this regulatory guide through the incorporation of the alternate approach cited.

General Compliance or Alternate Approach Assessment:

Three Components of Earthquake Motion

Response Spectrum Method

The use of three components of earthquake motion was not a design basis requirement of the construction permit for this plant. The total seismic response is predicted by combining the response calculated from analyses due to one horizontal and one vertical seismic input. For this case, where the response spectrum method of seismic analysis is used, the basis for combining the loads from the two analyses is given as follows:

a. The peak of the different modes for the same earthquake excitations do not occur at the same time,

b. The peak responses of a particular mode due to earthquake excitations from different directions do not occur at the same time, and

c. The peak stresses due to different modes and due to different excitations may not occur at the same location nor in the same direction.

To implement the above, the two translation components of earthquake excitations are combined by summing the absolute sum of all responses of interest (e.g., strain, displacement stress, moment, shear, etc.) from seismic motion, the one horizontal (x or z) and one vertical direction (y), i.e., $|x+y|$ or $|y+z|$. The design is made for the larger of the two sums $|x+y|$ or $|y+z|$.

Time History Method

The algebraic sum of contributions (to displacements, loads, stresses, etc.) due to the two earthquake components are calculated for each natural mode for each time interval of analysis. The time interval should be less than or equal to 0.2 of the smallest period of interest. The maximum values of all time intervals are the design displacements, accelerations, loads, or stresses.

It is concluded that the above method adequately demonstrates the integrity of the Seismic Category I subsystems and was found acceptable as a basis of current operating BWR plants.

Combination of Modal Responses

When the response spectra method of modal analysis is used, all modes are combined by the square root of the sum of the squares (SRSS) described as follows:

The SRSS combination of modal responses is defined mathematically as

$$R = \sqrt{\sum_{i=1}^{n} (R_i)^2}$$

where

R = Combined response

R_i = Response in the i^{th} mode

n = Number of modes considered in the analysis

Closely spaced modes are not accounted for as required by the guide because the design was significantly developed prior to issuance of the guide.

Specific Evaluation Reference:

See Sections 3.7.3.6 and 3.7.3.7.

Similar Application Reference:

Similar application was used for LaSalle.

Regulatory Guide 1.99, Revision 2, May 1988

Radiation Embrittlement of Reactor Vessel Materials

Regulatory Guide Intent:

This regulatory guide provides guidance for the prediction of irradiation damage of the reactor vessel belt line materials for the life of the vessel. This information is used to develop the pressure/temperature limit curves for the reactor pressure vessel based on material chemistry and end-of-life neutron exposure.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

The reactor pressure vessel pressure/temperature limit curves are in full compliance with the identified requirements in the regulatory guide.

General Compliance or Alternate Assessment:

Compliance is achieved by using a calculated end-of-life fluence for the CGS reactor vessel to evaluate the material damage due to this fluence. This information is used to predict the end-of-life NDT temperature for the limiting belt line material for the vessel. Using linear elastic fracture mechanics, the requirements of Welding Research Council Bulletin 175, the Standard Review Plan, and the requirements of Regulatory Guide 1.99, Revision 2, the pressure/temperature limit curves were developed for CGS. These curves will be used to evaluate the predictions determined by the regulatory guide until the submittal of new curves that incorporate the results of the surveillance capsule test data.

Specific Evaluation Reference:

See Sections 5.3.1.5.2.1 through 5.3.1.5.2.6 and the Technical Specifications.

Similar Application Reference:

Similar application is used on all reactor vessels.

Regulatory Guide 1.100, Revision 1, August 1977

Seismic Qualification of Electric Equipment for Nuclear Power Plants

Regulatory Guide Intent:

Regulatory Guide 1.100 endorses both the requirements and recommendations of IEEE 344-1975, "IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," when such qualification is performed in conjunction with Regulatory Guide 1.89, and subject to the regulatory position stipulations.

Compliance or Alternate Approach Statement:

General Compliance or Alternate Approach Assessment:

All Class 1E equipment seismic qualifications are evaluated against the requirements set forth within IEEE 344-1975 as clarified in Section 3.10.1.2. The evaluations are documented and demonstrated adequacy of the methods and results of the qualifications as equal or conservative to the requirements of IEEE 344-1975. This qualification documentation includes evaluation of seismic and hydrodynamic load combinations.

Specific Evaluation Reference:

See Section 3.10 and *"WNP-2 Dynamic Qualification Report for Safety-Related Equipment," dated September 1982.*

Regulatory Guide 1.145, Revision 1, November 1982/February 1983

Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants

Regulatory Guide Intent

This guide provides acceptable methodology to determining site-specific off-site air dispersion factors (χ/Q) for assessing the potential offsite radiological consequences of postulated accidental releases of radioactive material to the atmosphere.

Application Assessment

Assessed capability in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design, and equipment used in this facility is in full compliance with the regulatory guide.

General Compliance or Alternate Approach Assessment

Two of the procedures contained in the PAVAN code were implemented. The procedures were run with the desert sigma and with the Pasquill-Gifford sigma enabled. The most conservative χ/Q values were used in the accident analysis.

Specific Evaluation Reference:

See Section 2.3 and Chapter 15.0.

Regulatory Guide 1.183, Revision 0, July 2000

Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors

Regulatory Guide Intent:

This guide provides guidance to licensees of operating power reactors on acceptable applications of alternative source terms; the scope, nature, and documentation of associated analyses and evaluations; consideration of impacts on analyzed risk; and content of submittals. This guide establishes an acceptable alternative source term (AST) and identifies the significant attributes of other ASTs that may be found acceptable by the NRC staff. This guide also identifies acceptable radiological analysis assumptions for use in conjunction with the accepted AST.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design, and equipment used in this facility is in compliance with this regulatory guide or through the incorporation of the NRC approved alternate approach cited.

General Compliance or Alternate Approach Assessment:

This regulatory guide is applicable to the analyses for the FSAR. The Columbia analysis methods and assumptions (see Energy Northwest, "Columbia Generating Station Alternative Source Term," CGS-FTS-0168, Revision 0, August 2007) conform to position of this Regulatory Guide with the following specific considerations.

[Guide Section 3.4] Table 5 of the regulatory guide lists the elements in each radionuclide group that should be considered in design basis analyses. The intent of the guidance is met by an alternate approach. The Columbia analyses consider 66 nuclides consisting of 60 identified as being potentially important contributors to TEDE in NUREG/CR-4691 plus seven additional noble gas isotopes and Ba-137m.

[Guide Section 4.3] Columbia conforms with guide section 4.3 with the exception that the TID-14844 source term continues to be used as the radiation dose basis for equipment qualification.

[Guide Section 3.3 of Appendix A] The intent of the guidance is met by the conservative approach used in the Columbia analysis. The SRP 6.5.2 model is used. Elemental iodine is assumed to be removed at the same rate as particulate. The approach of treating elemental iodine as particulate is a conservative representation of the situation in which some elemental iodine would be removed by diffusion to spray water droplets and some elemental iodine would adsorb onto particulate. A reduction of 10 in iodine removal lambda is taken when 98% of the particulate has been removed. The method results in a conservative dose.

Specific Evaluation Reference:

See Chapter 15.4.9, 15.6.4, 15.6.5, 15.7.4.

Similar Application Reference:

Similar application was used for Grand Gulf and Brunswick.

Regulatory Guide 1.190, Revision 0, March 2001

Calculational and Dosimetry Methods for Determining Pressure Vessel Neutron Fluence

Regulatory Guide Intent:

This Regulatory Guide has been developed to provide state-of-the-art calculations and measurement procedures that are acceptable to the NRC staff for determining pressure vessel fluence.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

The methodology for the neutron flux calculation for the CGS reactor vessel conforms to Licensing Topical Report (LTR) NEDC-32983-P-A. In general, the methodology described in the LTR adheres to the guidance in Regulatory Guide 1.190 for neutron flux evaluation and was approved by the U.S. NRC in the Safety Evaluation Report (SER) for referencing in Licensing submittals.

General Compliance or Alternate Assessment:

Reference compliance assessment for Regulatory Guide 1.99.

Specific Evaluation Reference:

See Section 4.3.2.8.

Similar Application Reference:

Similar application is used for Browns Ferry Nuclear Plant, Units 2 and 3, reactor vessels.

Regulatory Guide 1.194, Revision 0, June 2003

Atmospheric Relative Concentrations for Control Room Radiological Habitability Assessments at Nuclear Power Plants

Regulatory Guide Intent:

This guide provides guidance on determining atmospheric relative concentrations (χ/Q) values in support of design basis control room radiological habitability assessments at nuclear power plants. This guide describes methods acceptable to the NRC staff for determining χ/Q values that will be used in control room radiological habitability assessments performed in support of applications for licenses and license amendment requests.

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

Identified NSSS scope of supply analysis, design, and equipment used in this facility is in compliance with this regulatory guide or through the incorporation of the NRC approved alternate approach cited.

General Compliance or Alternate Approach Assessment:

This regulatory guide is applicable to the analyses for the FSAR. The Instantaneous Puff Release alternative method provided by this guide is used to calculate χ/Q for the Main Steam Line Break accident.

Specific Evaluation Reference:

See Section 15.6.4.

Similar Application Reference:

1.8.3 BALANCE OF PLANT SCOPE OF SUPPLY EVALUATION

The following evaluations of implementation of regulatory guides are relative to BOP scope of supply. Thus, reference to CGS in the following evaluations is restricted to the BOP scope of supply portions of CGS. For NSSS scope of supply implementation of regulatory guides, see Section 1.8.2.

Conformance to the regulatory guides falls under either of the two following categories:

a. Compliance with the guidance set forth in this regulatory guide as described in this FSAR or

b. Compliance with the intent of the guidance set forth in this regulatory guide by an alternate approach.

The second category is based on NRC rules which state:

> Regulatory guides are not substitutes for regulations, and compliance with them is not required. Methods and solutions different from those set out in the guides will be acceptable if they provide a basis for the findings requisite to the assurance or continuance of a permit or license by the NRC.

Regulatory guides and their revisions are addressed in the following.

Regulatory Guide 1.6, Revision 0, March 1971

Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in this regulatory guide.

General Compliance or Alternate Approach Assessment:

The compliance assessments given below correspond numerically to the Regulatory Positions as enumerated in Section C of Regulatory Guide 1.6, Revision 0.

1. The electrically powered safety loads, both ac and dc, are separated into redundant load groups such that loss of any one group will not prevent the minimum safety function from being performed.

2. Each ac load group has a connection to the preferred offsite power source and to a standby onsite power source. The standby power sources have no automatic connection to any other redundant load groups.

3. Each dc load group is energized by a battery and battery charger. The battery-charger combination has no automatic connection to any other redundant dc load group.

4. When operating from the standby sources, redundant load groups and the redundant standby sources are independent of each other.

5. A single generator driven by two prime movers in tandem is the standby power source for the Division 1 and 2 ac load groups. The Division 3 ac load group power is supplied by a single generator driven by a single prime mover.

Specific Evaluation Reference:

See Sections 8.1.5.2, 8.3.1.1.7, 8.3.1.2.1.3, 8.3.1.2.1.4, 8.3.1.3, 8.3.1.4, 8.3.2.1.1, 8.3.2.2.1.2, 8.3.2.3, and 8.3.2.4.

Regulatory Guide 1.8, Revision 1-R, May 1977

Personnel Selection and Training

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in this regulatory guide.

General Compliance or Alternate Approach Assessment:

The minimum educational and experience qualifications for the onsite plant personnel with the exception of the Health Physics/Chemistry Supervisor are based on ANSI 18.1-1971, "Standard for Selection and Training of Personnel for Nuclear Power Plants," which is referenced by Regulatory Guide 1.8. Qualification requirements for the Health Physics/Chemistry Supervisor are as set forth in this guide.

Specific Evaluation Reference:

See Sections 13.1.3, 13.2.1, and the OQAPD.

Regulatory Guide 1.9, Revision 0, March 1971

Selection of Diesel Generator Set Capacity for Standby Power Supplies.

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in this regulatory guide.

General Compliance or Alternate Approach Assessment:

The compliance assessments given below correspond numerically to the regulatory positions as enumerated in Section C of Regulatory Guide 1.9, Revision 0.

1. Both the Division 1 and Division 2 diesel generator sets were selected to have a continuous load rating equal to or greater than the sum of the conservative estimated loads needed to be powered at any one time.

2. The predicted loads on both the Division 1 and the Division 2 diesel generator sets do not exceed the 2000-hr rating of either set, respectively, or 90% of the 30-minute rating of either set, respectively.

3. Predicted loads on Division 1 and Division 2 were verified by tests during preoperational testing.

4. The Division 1 and Division 2 diesel generator sets are capable of starting and accelerating to rated speed, in the required sequence, all the needed engineered safety feature and emergency shutdown loads.

   The Division 1 and Division 2 diesel generator sets are within the limits of undervoltage, under-frequency, overspeed and voltage and frequency restoration time limits, set forth in the regulatory guide.

5. The suitability of each diesel generator set of the standby power supply was confirmed by prototype qualification test data and preoperational tests.

Specific Evaluation References:

See Sections 8.1.5.2, 8.3.1.1.7, and 8.3.1.2.1.3.

Regulatory Guide 1.10, Revision 1, January 1973

Mechanical (Cadweld) Splices in Reinforced Bars of Category I Concrete Structures.

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in this regulatory guide.

General Compliance or Alternate Approach Assessment:

The requirements of the guide have been included in the appropriate specifications for the project construction. Compliance with the guide is ensured by testing and control procedures and reporting program. The program includes splicing crew qualifications, visual inspection of each splice, tensile testing of splice samples, tensile test frequency program, and a procedure for evaluating substandard test results. The procedure for testing and sampling of mechanical splices have been implemented.

Specific Evaluation Reference:

See Sections 3.8.3.2 and 3.8.4.2 and Table 3.8-4.

Regulatory Guide 1.11, Revision 0, March 1971

Instrument Lines Penetrating Primary Reactor Containment.

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in this regulatory guide.

General Compliance or Alternate Approach Assessment:

CGS design includes flow restriction orifices and/or excess flow check valves with position indication in instrument lines which penetrate primary reactor containment. In the event of an instrument line rupture outside primary containment, the integrity and functional performance of the secondary containment system and its associated filtration systems are maintained.

Specific Evaluation Reference:

See Sections 7.1.2.4 and 6.2.4.

Regulatory Guide 1.12, Revision 1, April 1974

Instrumentation for Earthquakes

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in this regulatory guide.

General Compliance or Alternate Approach Assessment:

Triaxial strong-motion accelerographs are installed at appropriate locations to provide data on the seismic input to containment; data on frequency, amplitude, and phase relationship of the seismic response of the containment structure; and data on the seismic input to other Category I structures, systems, and components.

Specific Evaluation Reference:

See Section 3.7.4.

Regulatory Guide 1.13, Revision 1, December 1975

Spent Fuel Storage Facility Design Basis

Compliance or Alternate Approach Statement:

CGS complies with the intent of the guidance set forth in this regulatory guide by an alternate approach.

General Compliance or Alternate Approach Assessment:

A controlled leakage building is provided enclosing the fuel pool. The building is not designed to withstand extremely high winds, but leakage is suitably controlled during refueling operations. The building is equipped with a ventilation and filtration system which is designed to limit the potential consequences of the release of radioactivity specified in Regulatory Guide 1.183 to those requirements set forth in 10 CFR 50.67.

The movement paths of heavy objects such as the reactor pressure vessel head, containment vessel head, and the spent fuel cask are designed not to pass over the spent fuel racks. Furthermore, the reactor building crane and its auxiliary hoist are prevented by means of interlocks from passing over any of the spent fuel pool except the spent fuel cask area. Bypassing of the interlocks is permitted only during fuel handling and storage operations and is administratively controlled.

The fuel pool is designed so that no pipe break will drain water from the fuel pool.

Specific Evaluation Reference:

See Section 9.1.

Regulatory Guide 1.15, Revision 1, December 1972

Testing of Reinforcing Bars for Category I Concrete Structures

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in this regulatory guide.

General Compliance or Alternate Approach Assessment:

The requirements of the guide have been included in the appropriate specifications for project construction. Compliance with the guide is assured by the implementation of qualified testing and control procedures and reporting. Included are qualified control procedures and reporting for the yield strength and tensile strength tests and deformation inspections recommended by the guide.

Specific Evaluation Reference:

See Sections 3.8.3.2, 3.8.4.2, and 3.8.5.2 and Table 3.8-4.

Regulatory Guide 1.16, Revision 4, August 1975

Reporting of Operating Information - Appendix A Technical Specifications

Compliance or Alternate Approach Statement:

This regulatory guide was withdrawn in August 2009 and is no longer applicable.

General Compliance or Alternate Approach Assessment:

Not applicable.

Specific Evaluation Reference:

Not applicable.

Regulatory Guide 1.17, Revision 1, June 1973

Protection of Nuclear Power Plants Against Industrial Sabotage

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in this regulatory guide.

General Compliance or Alternate Approach Assessment:

This information is considered proprietary and is subject to limited distribution. All specifics have been forwarded to the NRC as part of the Energy Northwest proprietary physical security plan for CGS.

Specific Evaluation Reference:

See proprietary physical security plan.

Regulatory Guide 1.18, Revision 1, December 1972.

Structural Acceptance Test for Concrete Primary Reactor Containments

Compliance or Alternate Approach Statement:

This regulatory guide is not applicable since CGS does not have a concrete primary containment.

General Compliance or Alternate Approach Assessment:

Not applicable.

Specific Evaluation Reference:

Not applicable.

Regulatory Guide 1.19, Revision 1, August 1972

Nondestructive Examination of Primary Containment Liner Welds

Compliance or Alternate Approach Statement:

This regulatory guide is not applicable since CGS does not have a concrete primary containment with a steel liner.

General Compliance or Alternate Approach Assessment:

Not applicable.

Specific Evaluation Reference:

Not applicable.

Regulatory Guide 1.21, Revision 1, June 1974

Measuring, Evaluating, and Reporting of Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants

Compliance or Alternate Approach Statement:

CGS complies with the intent of the guidance established in this regulatory guide by an alternate approach.

General Compliance or Alternate Approach Assessment:

The following categories of monitoring systems incorporated into the CGS design fulfill the requirements for monitoring in Regulatory Guide 1.21.

a. Gaseous effluents,
b. Liquid effluents, and
c. Solid Waste.

The above categories of monitoring systems adequately monitor effluent discharge paths for radioactivity that may be released from normal operations, including anticipated operational occurrences, and from postulated accidents.

Columbia Generating Station complies with Section C.11.b (Quality Controls) requirements for blind duplicate analysis by an alternate approach. An intralaboratory blind sample program is performed on selected samples. The blinds are prepared from samples sent from a cross check laboratory and split between several analysts as determined by the Chemistry Supervisor or designee. This process allows evaluation of individual analysts' performance while at the same time satisfying the blind duplicate and cross check laboratory requirements.

Section C.11.c (Calibrations) suggests that appropriate standards be used to calibrate continuous radioactivity monitors and that the relationship be established between monitor readings and concentration over the full range of the readout device. In those cases where mixed fission gases or corrosion and activation products are not available, vendor instrument performance data or calculations will be used. Subsequent inservice calibrations will be performed using the specific radionuclide analytical results from grab samples taken from the effluent release path.

Appendix A, Section A.3.a (1) and Section A.3.a (3), analytical frequencies are not consistent with standard sampling and analytical techniques. Improved sensitivities and more realistic quantity measurements can be made by performing ^{140}Ba-La, $^{89-90}$Sr, and gross alpha measurements on a monthly composite sample of weekly samples.

Exception is taken to the Appendix A, Section B.1.c, requirement for a special sample and analysis of one liquid waste batch per month for entrained fission and activation gases. The gamma spectrum analysis performed prior to the release of any waste liquid batch will identify such gases without performing a separate or special analysis.

The sensitivity slated in Appendix A, Section B.3, for gamma-emitting radionuclides (5 x 10^{-7} μCi/ml) will be applied in the case of principal gamma-emitting nuclides.

Specific Evaluation Reference:

See Section 11.5.

Regulatory Guide 1.22, Revision 0, February 1972

Periodic Testing of Protection System Actuation Functions

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in this regulatory guide.

General Compliance or Alternate Approach Assessment:

The CGS protection system and the systems whose operation it initiates are designed to permit periodic testing of the actuation devices during reactor operation. The periodic tests will duplicate, as closely as practical, the performance that is required of the actuation devices in the event of an accident. The tests will be performed in overlapping portions so that an actual reactor scram will not occur as a result of the testing.

Specific Evaluation Reference:

See Section 7.3.2.1.3.

Regulatory Guide 1.23, Revision 0, February 1972

Onsite Meteorological Program

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in this regulatory guide.

Where conflicts exist between the recommendations specified in Regulatory Guide 1.23, Revision 0 and those recommended in Regulatory Guide 1.97, Revision 2, the Columbia Generating Station will comply with the recommendations of Regulatory Guide 1.97, Revision 2 unless noted in the text discussions as meeting Regulatory Guide 1.97, Revision 3 requirements (see Section 7.5.2.2.3).

General Compliance or Alternate Approach Assessment:

The requirements of this regulatory guide for a meteorological program to provide the meteorological data required to estimate potential radiation doses to the public have been and are being implemented for CGS.

Specific Evaluation Reference:

See Sections 2.3.2, 2.3.3, 7.7.1, and the Emergency Plan.

Regulatory Guide 1.26, Revision 3, February 1976

Quality Group Classifications and Standards for Water-, Steam-, and Radioactive-Waste Containing Components of Nuclear Power Plants

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in this regulatory guide.

General Compliance or Alternate Approach Assessment:

The definition of quality group classifications for CGS was provided in the PSAR in accordance with ASME B&PV Code, Sections III and VIII. Quality group classifications have been maintained during design and construction. Quality group classifications are maintained during plant operations and modifications by plant administrative procedures and the plant modification control process. The quality group classifications are commensurate with the safety functions performed by the safety-related components.

The turbine stop valves and bypass valve, which are classified Quality Group D, are subject to an enhanced quality assurance program comparable to that of Quality Group B.

Specific Evaluation Reference:

See Section 3.2 and the OQAPD.

Regulatory Guide 1.27, Revision 2, January 1976

Ultimate Heat Sink for Nuclear Power Plants

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in this regulatory guide.

General Compliance or Alternate Approach Assessment:

Energy Northwest complies with Regulatory Guide 1.27, Revision 2, without any exceptions and with one clarification.

The clarification is that the tower makeup system (TMU) water supply is only an ultimate heat sink feature in the event of a design basis tornado. Since Regulatory Guide 1.27 states that we need not consider two or more most severe natural phenomena occurring simultaneously, the TMU was designed to be tornado proof but was not designed and constructed to withstand the effects of the operating basis earthquake (OBE) and water flow based on severe historical events in the region.

Specific Assessment Reference:

See Section 9.2.5.

Regulatory Guide 1.28, Revision 0, June 1972

Quality Assurance Program Requirements (Design and Construction)

Compliance or Alternate Approach Statement:

CGS *complies with the guidance set forth in this regulatory guide as described below.*

General Compliance or Alternate Approach Assessment:

Procurement documents issued after November 1973 required compliance with ANSI N45.2. Prior to that time, an "explanative version" of 10 CFR 50 Appendix B was used. The design and construction activities initially complied with 10 CFR 50 Appendix B. In November 1974, reference to ANSI N45.2 was added to the construction specifications.

ANSI N45.2 does not apply to the activities covered by Section III and Section XI of the ASME Code; however, the quality assurance program requirements may be extended to these activities based on project requirements.

Specific Evaluation Reference:

None

Regulatory Guide 1.29, Revision 3, September 1978

Seismic Design Classification

Compliance or Alternate Approach Statement:

CGS complies with the intent of the guidance set forth in this regulatory guide by an alternate approach.

General Compliance or Alternate Approach Assessment:

CGS classifications are consistent with Regulatory Guide 1.29 with the following clarification:

Cooling of the spent fuel storage pool is accomplished by the spent fuel cooling and cleanup system or by the seismic category RHR cross connection. The spent fuel pool cooling portion which is used normally to cool the spent fuel pool water was Seismic Category I by the first refueling outage. The cleanup portion of the system is not Seismic Category I. However, all structures, systems, and components required for maintaining water cover for the spent fuel are Seismic Category I. The spent fuel cooling system uses some common pump suction and discharge piping which is embedded in concrete. Prior to the first refueling outage, the Seismic Category I RHR system cross connection would have been used in case of core offload (see Section 9.1.3).

Specific Evaluation Reference:

See Sections 3.2.1, 3.7, 3.8, 3.9, 3.10, 9.1.3, and the OQAPD.

Regulatory Guide 1.30, Revision 0, August 1972

Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electrical Equipment.

I Design and Construction Phase

Compliance or Alternate Approach Statement:

CGS generally complies with the guidance set forth in this regulatory guide. In a few cases, CGS complied with the intent of this guidance by an alternate approach.

General Compliance or Alternate Approach Assessment:

Procurement documents require compliance with ANSI N45.2.4 for the installation, inspection, and testing activities performed, except in those isolated instances where requirements were entered directly in the specification with limited or no reference to ANSI N45.2.4 or IEEE 336.

Specific Evaluation Reference:

None

II Operational Phase

Compliance is discussed in the OQAPD.

Regulatory Guide 1.31, Revision 3, April 1978

Control of Ferrite Content in Stainless Steel Weld Metal

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in this regulatory guide.

General Compliance or Alternate Approach Assessment:

CGS complies fully with Revision 3 of this guide on all contracts initiated after the date of its publication. Prior to issuance of Revision 3, CGS conformed to Revision 2 of this regulatory guide.

Specific Evaluation Reference:

See Sections 4.5.2.4, 5.2.3.3, and 5.3.1.4.

Regulatory Guide 1.32, Revision 2, February 1977

Criteria for Safety Related Electric Power Systems for Nuclear Power Plants

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in Revision 0 of this regulatory guide.

(Revisions 1 and 2 are not applicable to CGS since they are for use in evaluations of construction permits docketed after November 1, 1976, and April 15, 1977, respectively.)

General Compliance or Alternate Approach Assessment:

The CGS design is in full compliance with both Revision 0 of this regulatory guide and with Revision 2 of this regulatory guide, with the exception of those sections of the regulatory guide which require compliance with Regulatory Guides 1.93, Revision 0, and 1.75, Revision 0. See Section 8.3.1.2.1.1 for analysis of the CGS design relative to Regulatory Guide 1.75, Revision 0.

Specific Evaluation References:

See Sections 8.1.5.1, 8.1.5.2, 8.2.2.4, 8.3.1.1.7.1, 8.3.1.2.1.3, 8.3.1.3, 8.3.1.4, 8.3.2.1.1, 8.3.2.2.1, 8.3.2.3 and 8.3.2.4.

Regulatory Guide 1.33, Revision 2, February 1978

Quality Assurance Program Requirements (Operation)

Compliance or Alternate Approach Statement:

CGS complies with the intent of the guidance set forth in this regulatory guide by an alternate approach.

Compliance or Alternate Approach Assessment:

Compliance is discussed in the OQAPD.

Specific Evaluation Reference:

See Section 13.5.1.1 and the OQAPD.

Regulatory Guide 1.34, Revision 0, December 1972

Control of Electroslag Weld Properties

Compliance or Alternate Approach Statement:

This regulatory guide is not applicable to CGS since electroslag welding has not been used for welding of Class 1 or 2 vessels or components fabricated of low alloy or austenitic steel.

General Compliance or Alternate Approach Assessment:

Not applicable.

Specific Evaluation Reference:

Not applicable.

Regulatory Guide 1.35, Revision 2, January 1976

Inservice Inspection of Ungrouted Tendons in Prestressed Concrete Containment Structures

Compliance or Alternate Approach Statement:

This regulatory guide is not applicable to CGS since CGS does not have a prestressed concrete containment structure with ungrouted tendons.

General Compliance or Alternate Approach Assessment:

Not applicable.

Specific Evaluation Reference:

Not applicable.

Regulatory Guide 1.36, Revision 0, February 1973

Nonmetallic Thermal Insulation for Austenitic Stainless Steel

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in this regulatory guide.

General Compliance or Alternate Approach Assessment:

Thermal insulation on stainless steel piping conforms to requirements of this regulatory guide.

Specific Evaluation Reference:

Not applicable.

Regulatory Guide 1.37, Revision 0, March 1973

Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants

I Design and Construction Phase

Compliance or Alternate Statement:

CGS generally complies with the guidance set forth in this regulatory guide. In a few cases, CGS complied with the intent of this guidance by an alternate approach.

General Compliance or Alternate Approach Assessment:

Procurement documents generally required compliance with ANSI N45.2.1. Whether or not reference to ANSI N45.2.1 was provided, a detailed specification section supplied comprehensive instructions on cleaning and cleanliness.

Specific Evaluation Reference:

None

II Operational Phase

Compliance is discussed in the OQAPD.

Regulatory Guide 1.38, Revision 2, May 1977

Quality Assurance Requirement for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants

I Design and Construction Phase

Compliance or Alternate Approach Statement:

CGS generally complies with the guidance set forth in Revision 0 of this regulatory guide. In a few cases, CGS complied with the intent of this guidance by an alternate approach.

The changes to the regulatory positions of Revision 1 and 2 of this regulatory guide, which specify additional detailed requirements and make certain nonmandatory sections of ANSI N45.2.2 mandatory, are not implemented.

General Compliance or Alternate Approach Assessment:

Procurement documents required compliance with ANSI N45.2.2, Revision 0, and/or contained a generic specification packaging section and/or specified directly requirements for these functions.

The regulatory positions contained in Revision 1 and 2 of this regulatory guide changed significantly from the original issue. Revision 1 and 2 contain additional detailed requirements and make nonmandatory sections of ANSI N45.2.2 mandatory. Some, but not all, of the changes to the regulatory positions are included in procurement documents. Since these changes were made after award of the applicable procurement documents, Revision 1 and 2 are not fully implemented.

Specific Evaluation Reference:

None

II Operational Phase

Compliance is discussed in the OQAPD.

Regulatory Guide 1.39, Revision 1, October 1976

Housekeeping Requirements for Water-Cooled Nuclear Power Plants

I Design and Construction Phase

Compliance or Alternate Approach Statement:

CGS generally complies with the guidance set forth in this regulatory guide. In some cases, CGS complied with the intent of this guidance by an alternate approach.

General Compliance or Alternate Approach Assessment:

Procurement documents required compliance with ANSI N45.2.3 or with selected portions of ANSI N45.2.3 or specified directly applicable housekeeping requirements.

Specific Evaluation Reference:

None

II Operational Phase

Compliance is discussed in the OQAPD.

Regulatory Guide 1.40, Revision 0, March 1973

Qualification Tests of Continuous Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in the regulatory guide.

General Compliance or Alternate Approach Assessment:

Containment fans have been qualified for in containment use in accordance with IEEE 334-1974.

Specific Evaluation Reference:

See Section 9.4.11.3.

Regulatory Guide 1.41, Revision 0, March 1973

Preoperational Testing of Redundant On-Site Electrical Power Systems to Verify Proper Load Group Assignments

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in the regulatory guide.

General Compliance or Alternate Approach Assessment:

As part of the preoperational test program, the onsite electric power systems will be tested in order to verify the existence of independence among redundant onsite power sources and their respective load groups.

Specific Evaluation Reference:

See Sections 8.1.5.2, 8.3.1.2.2 and 14.2.

Regulatory Guide 1.43, Revision 0, May 1973

Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components

Compliance or Alternate Approach Statement:

This regulatory guide is not applicable to CGS since CGS does not use stainless steel cladding on coarse grain low-alloy steel for safety class components.

General Compliance or Alternate Approach Assessment:

Not applicable.

Specific Evaluation Reference:

Not applicable.

Regulatory Guide 1.44, Revision 0, May 1973

Control of the Use of Sensitized Stainless Steel

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in this regulatory guide.

General Compliance or Alternate Approach Assessment:

CGS conforms fully to the recommended welding controls for stainless steel welding. All materials are purchased to the latest ASME and ASTM specifications at time of order, and the cleaning requirements set forth in the guide are implemented during document review of vendor cleaning procedures.

Specific Evaluation Reference:

See Sections 4.5.2.4 and 5.3.1.4.

Regulatory Guide 1.46, Revision 0, May 1973

Protection Against Pipe Whip Inside Containment

Compliance or Alternate Approach Statement:

CGS complies with the intent of the guidance set forth in this regulatory guide by an alternate approach.

Pipe break location criteria is based on guidelines provided in this regulatory guide, as well as the NRC Branch Technical Positions ASB 3-1, Appendix B, and MEB 3-1. The criteria is applicable to all piping systems inside as well as outside containment. Pipe whip protection for the recirculation system is provided by the NSSS supplier. Pipe whip protection for all other piping systems, including the NSSS-furnished main steam piping, is provided by the architect-engineer.

Specific Evaluation Reference:

See Section 3.6.2.1.

Regulatory Guide 1.47, Revision 0, May 1973

Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems

Compliance or Alternate Approach Statement:

CGS complies with the intent of the guidance set forth in this regulatory guide by an alternate approach.

General Compliance or Alternate Approach Assessment:

Each safety-related system described in Sections 7.2, 7.3, 7.4, and 7.6 is provided with an automatically or operator initiated system level bypass and inoperability annunciator.

The system level annunciators are located with the associated system controls and indications on main control room panels.

In addition to system level annunciation, component and channel level annunciators are provided on other panels either in the control room near system controls or locally near affected equipment, to indicate the cause of the system bypass or inoperability.

A switch is provided for manual actuation of each system level annunciator to allow display of those bypass or inoperable conditions which are not automatically indicated.

Typically, the following bypasses or inoperabilities cause actuation of system level (and component level) annunciation for the affected systems:

a. Pump motor breaker not in operate position,

b. Loss of pump motor control power,

c. Loss of motor-operated valve control power/motive power,

d. Logic power failure,

e. Logic in test,

f. Position of remote manual valves which do not receive automatic alignment signals, and

g. Bypass or test switches actuated.

Auxiliary supporting system inoperability or bypass resulting in the loss of other safety-related systems will cause actuation of system level annunciators for the auxiliary supporting system as well as those safety-related systems affected.

Specific Evaluation Reference:

Not applicable.

Regulatory Guide 1.48, Revision 0, May 1973

Design Limits and Loading Combinations for Seismic Category I Fluid System Components

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in this regulatory guide.

General Compliance or Alternate Approach Assessment:

Implementation of this regulatory guide is discussed in Section 3.9.3.1.1.7.

Specific Evaluation Reference:

See Section 3.9.3.1.1.7.

Regulatory Guide 1.50, Revision 0, May 1973

Control of Preheat Temperature for Welding Low-Alloy Steel

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in this regulatory guide.

General Compliance or Alternate Approach Assessment:

CGS complies with the guidance set forth in the regulatory guide by maintaining the preheat temperature of low alloy steel welds until the post-weld heat treatment has been performed. For welds which were made without this “keep hot” requirement, Regulatory Position C4 for determining the soundness of the weld by acceptable examination procedures, has been enforced.

Specific Evaluation Reference:

See Section 5.3.1.4.

Regulatory Guide 1.51, Revision 0, May 1973

In-Service Inspection of ASME Code Class 2 and 3 Nuclear Power Plant Components

Compliance or Alternate Approach Statement:

This regulatory guide has been withdrawn and is no longer applicable.

General Compliance or Alternate Approach Assessment:

Inservice inspection of CGS is based on ASME Section XI for Classes 1, 2, and 3.

Specific Evaluation Reference:

See Section 3.9.6.

Regulatory Guide 1.52, Revision 2, March 1978

Design, Testing, and Maintenance Criteria for Atmosphere Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants

Compliance or Alternate Approach Statement:

CGS complies with the intent of the guidance given in Revision 2 of this regulatory guide.

General Compliance or Alternate Approach Assessment:

Standby gas treatment filter units and the control room emergency filter units are required to perform safety-related functions. A comparison of the engineered safety feature air filtration systems with respect to the regulatory position of Regulatory Guide 1.52, Revision 2, Article C, is as follows:

Paragraph Number	SGTS	Control Room System
C-1. "Environmental Design Criteria"		
1.a	In compliance	In compliance
1.b	In compliance	In compliance
1.c	In compliance	In compliance
1.d	In compliance	In compliance
1.e	In compliance	In compliance

C-2. "System Design Criteria"

2.a	In compliance	See Note 1
2.b	In compliance	In compliance
2.c	In compliance	In compliance
2.d	See Note 2	See Note 2
2.e	In compliance	In compliance
2.f	In compliance	In compliance
2.g	See Note 3	See Note 3
2.h	In compliance	In compliance
2.i	In compliance	In compliance
2.j	See Note 4	See Note 4
2.k	In compliance	In compliance
2.1	In compliance	In compliance

C-3. "Component Design Criteria and Qualification Testing"

3.a	See Note 5	See Note 5
3.b	In compliance	In compliance
3.c	In compliance	In compliance
3.d	See Note 6	See Note 6
3.e	In compliance	In compliance
3.f	In compliance	In compliance
3.g	See Note 7	See Note 7
3.h	In compliance	In compliance
3.i	See Note 8	See Note 8
3.j	In compliance	In compliance
3.k	In compliance	In compliance
3.l	In compliance	In compliance
3.m	In compliance	In compliance
3.n	In compliance	In compliance
3.o	In compliance	In compliance
3.p	In compliance	In compliance

C-4. "Maintenance"

4.a	See Note 9	See Note 9
4.b	See Note 10	See Note 10
4.c	In compliance	In compliance
4.d	See Note 11	In compliance
4.e	In compliance	In compliance

C-5. "In-Place Testing Criteria"

5.a	In compliance	In compliance
5.b	See Note 13	In compliance
5.c	See Note 14	See Note 14
5.d	See Note 14	See Note 14

C-6. "Laboratory Testing Criteria For Activated Carbon"

6.a	See Note 12	See Note 12
6.b	See Note 12	See Note 12

Note 1 (C-2.a) Demisters are not provided in the control room filter units due to the absence of entrained moisture during normal and abnormal conditions. High-efficiency particulate air (HEPA) filters are not provided after the charcoal filter because filter unit discharges into control room air conditioning unit on intake side of medium efficiency filters.

Note 2 (C-2.d) Both units of the standby gas treatment system are located in secondary containment and are not subject to containment pressure surges during accidents. Redundant Seismic Category I valves in series isolate and protect these units from containment DBA pressures. Both units of the control room emergency filter system are not subject to containment pressure surges during accidents.

Note 3 (C-2.g) Abnormal pressure drops across critical components of the SGTS and control room filter units cause an alarm in the main control room, however, no facilities to record the pressure drops are provided. A record of pressure drop across individual components and the total SGTS system would be of no value because the SGTS is a variable flow system, with flow modulated to maintain the reactor building at a fixed negative pressure. Flow through the system, which is the pertinent parameter, is recorded in the main control room, and computer input is provided to record high pressure alarms across critical components.

Note 4 (C-2.j) SGTS filter units are not designed to be removable from the building as an intact unit. The size of the units precludes removal in one section. In the event the units become radioactively contaminated they will be permitted to decay in place until radiation levels are sufficiently low to permit the removal of all internals for disposal.

Note 5 (C-3.a) SGTS system demisters furnished by FARR Company, are not in complete conformance with ANSI N509-1976 because they were not qualified by testing in accordance with AEC report MSAR-71-45. A moisture eliminator study performed by FARR Company in 1970, which did not conform to the MSAR-71-45 test setup, indicated that the installed demisters will protect the HEPA filters in the system from blinding under conditions far more severe than those hypothesized for the SGTS system. Since, under the accident mode, entrained water droplets will not be in the inlet air stream, the FARR tests and qualification are considered adequate.

Note 6 (C-3.d)	HEPA filters are not subjected to iodine removal sprays, therefore, aluminum separators are used. An alternate approach to determine acceptable design and qualification testing of HEPA filters is the use of Regulatory Guide 1.52, Revision 3, Section 4.4.
Note 7 (C-3.g)	Access doors into SGTS units are 50 x 20 in. Vacuum breakers are not provided on doors of SGTS and control room units. Unit fans are normally off.
Note 8 (C-3.i)	Test 4, Activity (Ref. Table 5-1, ANSI N509-1976) Base carbon (unimpregnated) activity test was not previously required. Because all available carbon was of the impregnated type this was not run. Test 5, Radioiodine Removal Efficiency (Ref. Table 5-1, ANSI N509-1976) New carbon will be tested in accordance with ASTM D3803-1989. Average atmosphere resident time in each SGTS unit is greater than 0.5 sec.
Note 9 (C-4.a)	Doors provided on SGTS Units are 50 x 20 in. Access panels are provided on control room units. Vacuum breakers are not provided on any of the units since they are normally not operational.
Note 10 (C-4.b)	Control room filter units have approximately 18 in. between prefilter and HEPA filter frames, and approximately 4 ft are provided between HEPA and charcoal filter frames. SGTS filter units have a minimum of three feet provided between demister, heater, prefilter, HEPA and charcoal filter frames.
Note 11 (C-4.d)	Strip heaters are provided in the charcoal filter plenum of the SGTS units to maintain charcoal beds moisture free, therefore, operation of the fans is not required for that purpose.

Note 12 (C-6.a C-6.b)	The laboratory testing criteria for the carbon adsorber section of the SGTS and CREF System meets the objectives of this section of the guide. Twelve representative test samples of four-inch length are provided across each of the two 4 in. deep beds in each SGTS filter unit. At least once per 30 months one sample from across each SGT and CREF adsorber bed is removed and sent to a laboratory for testing. For the SGTS, samples are tested in series to represent the 8-inch total bed depth. Laboratory tests are performed in accordance with ASTM D3803-1989 with methyl iodide at 30°C and 70% relative humidity with a penetration of less than 0.5% for the SGTS and less than 2.5% for the CREF System as an acceptance level. The SGTS will also be tested at a face velocity of 75 ft per minute. In the event that a sample fails this test, the carbon adsorber in its bed will be replaced.
Note 13 (C-5.b)	The flow distribution tests developed by the designer combined with the series filter design at CGS adequately meet the intent of this test. The results of the flow distribution tests as set forth in ANSI N51 are difficult to interpret with the 'U' shaped charcoal beds installed due to air flow disturbance caused by the measuring apparatus. This is particularly true on the parallel legs of the 'U' shaped beds, where the flow measuring device must be placed in the rather narrow air passage. Flow distribution criteria was developed by the designers based on the ±20% variation criteria established in Regulatory Guide 1.52 and has been met in field tests. In addition, each of the filter trains has two separate charcoal beds in series. This allows mixing of the filtered gas between the beds and further reduces the effects of variations in charcoal packing distribution.
Note 14 (C-5.c C-5.d)	The inplace leak testing of the SGT and CREF HEPA and carbon filters meets the objectives of this section of the guide with the exception that testing is performed in accordance with ASME N510-1989, Sections 10 and 11, respectively.

Specific Evaluation Reference:

See Section 6.5.1.

Regulatory Guide 1.53, Revision 0, June 1973

Application of the Single-Failure Criterion to Nuclear Power Plant Protective Systems

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in the regulatory guide.

General Compliance or Alternate Approach Assessment:

Regulatory Guide 1.53 provides guidance for the application of the single-failure criterion as discussed in IEEE 379-1972. The regulatory guide recommends the application of IEEE 379-1972 with four supplemental conditions. The design of the CGS electrical system is in conformance with IEEE 379-1972 and the four supplemental conditions noted in Regulatory Position C.

Specific Evaluation Reference:

See Section 8.1.5.2.

Regulatory Guide 1.54, Revision 0, June 1973

Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in this regulatory guide as described below.

General Compliance or Alternate Approach Assessment:

Special decontaminable coatings in primary containment areas are manufactured and applied in accordance with quality assurance requirements of ANSI N101.4.

Specific Evaluation Reference:

See Section 6.1.2.

Regulatory Guide 1.55, Revision 0, June 1973

Concrete Placement in Category I Structures

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in this regulatory guide.

General Compliance or Alternate Approach Assessment:

The requirements of the guide have been included in the appropriate construction contract specifications. Compliance with the guide is assured by the application of appropriate concrete specifications, construction practices, codes and standards, including the documents recommended by the guide, for the placement of concrete; by the implementation of approved communications procedures between qualified design and construction forces; and by implementation of an approved QA program which ensures design control and coordinated quality control of concrete material, placement, inspection and testing between applicant, designer and constructor.

Specific Evaluation Reference:

See Sections 3.8.3.2, 3.8.3.6, 3.8.4.2, 3.8.4.6, and 3.8.5.2 and Table 3.8-4.

Regulatory Guide 1.56, Revision 0, June 1973

Maintenance of Water Purity in Boiling Water Reactors

I. Design and Construction Phase

Compliance or Alternate Approach Statement:

The design of CGS complies with the guidance set forth in this regulatory guide.

General Compliance or Alternate Approach Assessment:

CGS design complies with the guidance of this regulatory guide by providing for the following:

a. Conductivity measurement and recording of the condenser hotwell and condensate flow discharge to the condensate demineralizer system,

b. Flow measurement and recording of flow through each condensate demineralizer unit,

c. Conductivity measurement, recording, and alarming of the condensate effluent discharge from each condensate demineralizer unit and from the combined system effluent,

d. Conductivity measurement, recording, and alarming of the inlet and outlet coolant to and from the RWCU system,

e. Extensive sampling of reactor coolant and auxiliary systems,

f. Full flow condensate demineralizer system, and

g. Excess condensate demineralizer capacity to permit recharging of resin beds during normal plant operation.

Specific Evaluation Reference:

See Section 5.2.3.2.2.

II. Operations Phase

Compliance or Alternate Approach Statement:

Operation of CGS RWCU and condensate demineralizer system complies with the general guidance set forth in Revision 1, July 1978, of this regulatory guide.

General Approach or Alternate Approach Assessment:

Operation of CGS complies with the guidance of the regulatory guide by providing the following:

a. Operating limits are prescribed for condensate filter demineralizers. Plant operating conductivity limits are defined for the RWCU demineralizers. Effluent conductivity for the individual demineralizers is recorded and a main control room alarm is triggered when conductivity limits are reached or exceeded;

b. Condensate filter demineralizer conductivity and flow instrumentation are used in the general assessment of individual demineralizer unit performance and capacity;

c. An operational limit is set for hotwell conductivity which triggers a main control room alarm. Hotwell conductivity, in conjunction with precalculated assessment of condenser inleakage rates and demineralizer performance permits appropriate action to be taken on exceeding the operating limit setpoint;

d. Laboratory analyses are performed for chloride, pH, and conductivity at intervals appropriate to the plant operating status. Sampling and analysis frequency is described in the LCS and plant procedures; and

e. Not applicable exception is taken to item C.4.d which applies to bead-type, deep-bed demineralizer systems, which are not incorporated into the CGS design. The general guidance of this item will, however, be applied to the pressure precoat filter demineralizer systems. Each lot of precoat resins will be analyzed for capacity and impurity levels. Frequency of precoat changeout will be staggered and is initially dictated by pressure drop associated with suspended solids. Subsequent to pressure drop limitations, frequency of sequential precoat changeout is established based on dissolved chemical constituents and flow throughput parameters.

Regulatory Guide 1.57, Revision 0, June 1973

Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components

Compliance or Alternate Approach Statement:

CGS complies with the intent of the guidance set forth in this regulatory guide by an alternate approach.

General Compliance or Alternate Approach Assessment:

The structural design criteria for the primary containment vessel is consistent with the provisions of this regulatory guide, except with respect to the stress limits specified in Section C-1-b(2) of the guide, for the load combination of accident recovery flooding plus OBE. For this load combination, the stress limits used for CGS are within the limits set forth in the NRC Standard Review Plan Section 3.8.2, Table 3.8.2-1.

This exception has precedent as stated in GESSAR, paragraph 3.8.2.3.12, "Accident Recovery Evaluation," Page 3.8-9b, and has been accepted by the NRC, as documented in paragraph 3.8.2, page 3-14, of the NRC Safety Evaluation Report for the GESSAR-328 Nuclear Island Standard Design dated December 1975.

Specific Evaluation Reference:

See Section 3.8.2.3.10.

Regulatory Guide 1.58, Revision 1, August 1980

Qualification of Nuclear Power Plant Inspection, Examination, and Testing Personnel

I Design and Construction Phase

Compliance or Alternate Approach Statement:

As of November 1980, CGS complies with the guidance set forth in this regulatory guide via an alternate approach described below.

General Compliance or Alternate Approach Assessment:

Prior to issuance of Revision 1 of this Regulatory Guide, personnel performing quality-related activities were provided indoctrination and training in the requirements of the applicable quality assurance program, procedures, instructions and drawings affecting their work. Documented evidence of the above training was maintained. The indoctrination and training complied with the requirements of Appendix B, 10 CFR Part 50, and ANSI N45.2.

As of November 1980, in addition to the indoctrination and training requirements noted above, requirements which meet this regulatory guide were imposed on site contractors for personnel performing inspections, examinations, and tests. These requirements specify that initial evaluations of education, experience, and qualifications are to be performed and documented; however, formal certificates are not required to be issued because specific inspections, examinations, and tests are performed in accordance with approved procedures. Therefore, specific capability identification and levels of certification are not required.

Specific Evaluation Reference:

None

II Operational Phase

Compliance is discussed in the OQAPD. Also see Section 14.2.

Regulatory Guide 1.59, Revision 1, April 1976

Design Basis Floods for Nuclear Power Plants

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in this regulatory guide.

General Compliance or Alternate Approach Assessment:

All the requirements that are specified in Regulatory Guide 1.59 are followed in the design of CGS.

Based on Regulatory Guide 1.102, the plant site is classified as "Dry Site." Therefore, CGS is considered to be in compliance with Regulatory Guide 1.59 and its Appendix A.

Specific Evaluation Reference:

See Section 2.4.

Regulatory Guide 1.60, Revision 1, December 1973

Design Response Spectra for Seismic Design of Nuclear Power Plants

Compliance or Alternate Approach Statements:

CGS complies with the intent of the guidance set forth in this regulatory guide by an alternate approach.

General Compliance or Alternate Approach Assessment:

CGS meets the seismic requirements previously acceptable to the NRC as discussed in Section 3.7.1.1.

Specific Evaluation Reference:

See Section 3.7.1.1.

Regulatory Guide 1.61, Revision 0, October 1973

Damping Values for Seismic Design of Nuclear Power Plants

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in this regulatory guide.

General Compliance or Alternate Approach Assessment:

The damping values recommended by Regulatory Guide 1.61 are greater, and therefore less conservative, than the values used for CGS. The more conservative CGS design satisfies the requirements of Regulatory Guide 1.61.

Specific Evaluation Reference:

See Section 3.7.1.3.

Regulatory Guide 1.62, Revision 0, October 1973

Manual Initiation of Protective Actions

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in this regulatory guide.

General Compliance or Alternate Approach Assessment:

Means are provided in the main control room for the manual initiation of BOP engineered safety feature systems or supporting systems at the division level by the operation of a minimum of equipment.

Specific Evaluation Reference:

See Section 7.3.2.1.3.

Regulatory Guide 1.63, Revision 2, July 1978, and Revision 3, February 1987

Electric Penetration Assemblies in Containment Structures for Light-Water-Cooled Nuclear Power Plants

Compliance or Alternate Approach Statement:

Revisions 2 and 3 are not applicable to CGS since they apply to the evaluation of construction permit applications docketed after August 31, 1978 and February 28, 1987, respectively. CGS complies with the guidance set forth in IEEE 317-1972 as modified by Revision 0 of Regulatory Guide 1.63.

General Compliance or Alternate Approach Assessment:

The compliance assessment given below correspond numerically to the regulatory positions as indicated in Section C of Regulatory Guide 1.63, Revision 0, October 1973.

1. Capability of withstanding maximum fault I^2T heating in the case that overload protective devices fail:

   CGS is in compliance with this requirement. In all cases, the overcurrent protective devices in circuits subject to short circuit are backed up by other overcurrent protective devices which are also designed to limit the fault current I^2T heating experienced by the penetration conductors to levels below the conductor ratings.

2. The maximum containment pressure specified for CGS complies with the safety margins required by the ASME B&PV Code, Article N3000, footnote 1.

3. The position refers to specific applicability or acceptability of other codes, standards, and guides covered separately in other regulatory guides.

4. CGS complies with the requirement of IEEE 336 and ANSI N45.2 concerning the QA.

Specific Evaluation Reference:

See Sections 3.8.6, 7.1.2.3, and 8.1.5.2.

<u>Regulatory Guide 1.64, Revision 2, June 1976</u>

Quality Assurance Requirements for the Design of Nuclear Power Plants

I. <u>Design and Construction Phase</u>

<u>Compliance or Alternate Approach Assessment</u>:

Regulatory Guide 1.64, Revision 0, Revision 1, and Revision 2 do not apply to CGS since they apply to construction permits docketed after September 1973.

<u>General Compliance or Alternate Approach Assessment</u>:

Not applicable.

<u>Specific Evaluation Reference</u>:

Not applicable.

II. <u>Operational Phase</u>

Compliance is discussed in the OQAPD.

Regulatory Guide 1.67, Revision 0, October 1973

Installation of Overpressure Protection Devices

Compliance or Alternate Approach Statement:

This regulatory guide is not applicable to CGS since the reactor coolant system pressure boundary safety/relief valve relieves to a closed discharge system.

General Compliance or Alternate Approach Assessment:

Not applicable.

Specific Evaluation Reference:

Not applicable.

Regulatory Guide 1.68, Revision 1, January 1977

Initial Test Programs for Water-Cooled Reactor Power Plants

Compliance or Alternate Approach Statement:

> *This regulatory guide is not applicable to the CGS initial test program since Revision 0 of this regulatory guide is committed to in Section 14.2.7. However, CGS complies with the intent of the guidance set forth in this regulatory guide by an alternate approach.*

General Compliance or Alternate Approach Assessment:

> *See Section 14.2 for description of initial testing program and to Sections 14.2.7 and 1.8.2 for statements concerning compliance with Regulatory Guide 1.68, Revision 0. Revision 1 of this guide in general clarifies Revision 0 and therefore there are no exceptions to the intent of this procedure.*

Specific Evaluation Reference:

> *See Sections 14.2.7 and 1.8.2 for a discussion of Regulatory Guide 1.68, Revision 0.*

Regulatory Guide 1.68.1, Revision 1, January 1977

Preoperational and Initial Startup of Feedwater and Condensate Systems for Boiling Water Reactor Power Plants.

Compliance or Alternate Approach Statements:

CGS complies with the intent of the guidance set forth in this regulatory guide by an alternate approach.

General Compliance or Alternate Approach Assessments:

The preoperational testing and the initial Startup testing as described in Section 14.2 complies with the intent of this regulatory guide. However, due to the limitations of the auxiliary steam supply system, the confirmation that the feedwater pumps satisfy required head, flow rate and suction head will not occur until the startup phase of the initial test program when the normal steam supply is available to the feedwater pump turbines.

Specific Evaluation Reference:

See Section 14.2.12.1.1.

Regulatory Guide 1.68.2, Revision 0, January 1977

Initial Startup Test Program To Demonstrate Remote Shutdown Capability For Water-Cooled Nuclear Power Plants

Compliance or Alternate Approach Statement:

CGS complies with the intent of the guidance set forth in this regulatory guide by an alternate approach.

General Compliance or Alternate approach assessment:

The startup test described in Section 14.2.12.3.28 complies with the regulatory guide with the following exceptions:

a. The test will be initiated by scramming plant from the control room versus a location outside the control room as described in Section C.3 of the regulatory guide. This exception is made to better simulate the actual procedure which would be followed if a control evacuation were to occur. The capability to scram the reactor outside the control room exists; for example, tripping the RPS motor generator (MG) sets.

b. The cold shutdown demonstration procedure as described in Section C.4 of the Regulatory Guide may not be performed immediately following the demonstration of achieving and maintaining safe hot standby from outside the control room. Rather this cooldown portion may be performed when cooldown is required during the course of the normal power ascension test program. Although this is an exception to Regulatory Guide 1.68.2, Revision 0, Revision 1 of this Guide contains provisions for a delay in the demonstration of cooldown.

Specific Evaluation Reference:

See Sections 14.2.12.3.28 and 7.4.1.4.

Regulatory Guide 1.69, Revision 0, December 1973

Concrete Radiation Shields for Nuclear Power Plants

Compliance or Alternate Approach Statement:

CGS complies with the intent of the guidance set forth in this regulatory guide by an alternate approach.

General Compliance or Alternate Approach Assessment:

Although the regulatory guide was promulgated after design and specification implementation of the engineering criteria, the recommended design and construction practices specified in the regulatory guide are documented in codes and specifications which were used in the development of the engineering criteria and contract specifications.

Specific Evaluation Reference:

See Section 12.3.2.

Regulatory Guide 1.70, Revision 2, September 1975

Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants - LWR Edition

Compliance or Alternate Approach Statement:

This FSAR complies with the guidance set forth in this regulatory guide.

General Compliance or Alternate Approach Assessment:

The FSAR has generally been prepared to satisfy the requirements of Regulatory Guide 1.70, Revision 2. This includes both format and content.

Specific Evaluation Reference:

The balance-of-plant (BOP) portions of this FSAR.

Regulatory Guide 1.71, Revision 0, December 1973

Welder Qualifications for Areas of Limited Accessibility

Compliance or Alternate Approach Statement:

CGS complies with the intent of the guidance set forth in this regulatory guide by an alternate approach.

General Compliance or Alternate Approach Assessment:

There are few incidents where welding accessibility is limited during fabrication. Where accessibility to any weld joint was restricted to a degree which prevented the welder from direct visual observation of the arc and the puddle in any area of the weld, or which required the use of mirrors or extensions to the torch handle or electrode holder, the contractor notifies the welding engineer. All limited access welds are determined by a welding engineer. For ASME Section III, Class 1, 2, and 3 components and Subsection NF and NE, a performance qualification test that simulates the limited access condition is required by the welding engineer. For welds in the pressure retaining components the welder's test weld is radiographed in accordance with and shall conform to the acceptance standards of ASME Section VIII, Division 1, U.W.-51. Alternately, the weld may be examined ultrasonically in accordance with ASME Section VIII, Division 1, Appendix U.

Specific Evaluation Reference:

See Sections 4.5.2.4, 5.2.3.3, and 5.3.1.4.

Regulatory Guide 1.72, Revision 0, December 1973

Spray Pond Plastic Piping

Compliance or Alternate Approach Statement:

This regulatory guide is not applicable to CGS because CGS does not use plastic piping in its spray ponds.

General Compliance or Alternate Approach Assessment:

Not applicable.

Specific Evaluation Reference:

Not applicable.

Regulatory Guide 1.73, Revision 0, January 1974

Qualification Tests of Electric Valve Operators Installed Inside the Containment of Nuclear Power Plants

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in this regulatory guide.

General Compliance or Alternate Approach Assessment:

Auxiliary equipment associated with valve operators are tested in accordance with the subject standards. Designed service conditions are implemented in the tests. Conservative values of the environmental variables during and after a design basis accident are used in the tests to assure that the testing is carried out under more severe environmental conditions than those expected.

Specific Evaluation Reference:

See Sections 3.11 and 8.1.5.2.

Regulatory Guide 1.74, Revision 0, February 1974

Quality Assurance Terms and Definitions

I Design and Construction Phase

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in this regulatory guide.

General Compliance or Alternate Approach Assessment:

The terms used in describing and implementing quality assurance programs for CGS have complied with ANSI N45.2.10-1973 or were clarified at the point of application.

Specific Evaluation Reference:

None

II Operational Phase

Compliance is discussed in the OQAPD.

Regulatory Guide 1.75, Revision 1, January 1975

Physical Independence of Electric Systems

Compliance or Alternate Approach Statement:

This regulatory guide is not applicable to CGS since it applies to the evaluation of construction permit applications docketed after February 1974. However, CGS complies with the intent of the guidance set forth in this regulatory guide by an alternate approach.

General Compliance or Alternate Approach Assessment:

See Section 8.3.1.4.2.7 for an assessment of CGS relative to this regulatory guide.

Specific Evaluation Reference:

See Section 8.3.1.4.2.7.

Regulatory Guide 1.76, Revision 0, April 1974

Design Basis Tornado for Nuclear Power Plants

Compliance or Alternate Approach Statement:

CGS complies with the intent of the guidance set forth in this regulatory guide by an alternate approach.

General Compliance or Alternate Approach Assessment:

The tornado design criteria for Columbia Generating Station were revised based on design basis tornado characteristics in NUREG-1503. The design basis tornado characteristics used are less severe than those specified in Regulatory Guide 1.76 for Region III. In January 1996, the revised criteria were found acceptable by the NRC.

Specific Evaluation Reference:

See Section 3.3.2.

Regulatory Guide 1.78, Revision 0, June 1974

Assumptions for Evaluating the Habitability of a Nuclear Power Plant Control Room During a Postulated Hazardous Chemical Release

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in this regulatory guide.

General Compliance or Alternate Approach Assessment:

The main control room habitability during a postulated hazardous chemical release evaluation complies with assumptions and toxicity limits in Revision 0 of this regulatory guide. The evaluation uses toxicity limits presented in Revision 1 for those chemicals not discussed in Revision 0. The results are presented in Chapter 6.

Specific Evaluation Reference:

See Sections 2.2.3 and 6.4.

Regulatory Guide 1.80, Revision 0, June 1974

Preoperational Testing of Instrument Air Systems

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in this regulatory guide.

General Compliance or Alternate Approach Assessment:

The primary containment instrument air system preoperational test procedure incorporated the requirements of this regulatory guide.

Specific Evaluation Reference:

See Sections 14.2.7.3 and 14.2.12.1.34.

Regulatory Guide 1.82, Revision 0, June 1974

Sumps for Emergency Core Cooling and Containment Spray Systems

Compliance or Alternate Approach Statement:

This regulatory guide is not applicable to CGS since no sumps are used for ECCS and containment spray.

General Compliance or Alternate Approach Assessment:

Not applicable.

Specific Evaluation Reference:

Not applicable.

Regulatory Guide 1.84

Design, Fabrication, and Materials Code Case Acceptability, ASME Section III

Regulatory Guide Intent:

This guide lists all Section III Code Cases that the NRC has approved for use. It is updated on a regular basis to reflect the changes to the ASME Code Cases and the current position of the NRC on acceptability for use. The guide contains tables that detail the NRC acceptance requirements for current, annulled, and superseded Code Cases. Code Cases that the NRC determined to be unacceptable are listed in Regulatory Guide 1.193, "ASME Code Cases Not Approved for Use".

Application Assessment:

Assessed capability in design.

Compliance or Alternate Approach Statement:

The current version of the Regulatory Guide is utilized to determine acceptable Code Cases for all new and existing plant applications. The FSAR does not track individual Code Cases and revision numbers. Not all acceptable Code Cases listed in the regulatory guide are used. The Code Cases that are utilized for Columbia are referred to in the plant design/installation documentation.

General Compliance or Alternate Approach Assessment:

Code Cases are utilized in accordance with the requirements of the regulatory guide provisions for acceptance. Section III Code Cases that are not yet endorsed may be utilized via submittal to the NRC for approval in accordance with the regulatory guide. The plant scope of supply is in full compliance with this regulatory guide.

Specific Evaluation Reference:

See Section 3.8.2.2.

Similar Application Reference:

None.

Regulatory Guide 1.85, Revision 31, 1998*

Materials Code Case Acceptability - ASME Section III, Division 1

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in this regulatory guide as described below.

General Compliance or Alternate Approach Assessment:

The use of an ASME Section III, Division 1, code case applicable to materials use on CGS is approved by Energy Northwest only after evaluating its technical acceptability and confirming that its use is acceptable to the NRC. This confirmation is by ascertaining that the code case is listed in this regulatory guide (or applicable earlier revision) or by specific written acceptance by the NRC.

Specific Evaluation Reference:

See Section 3.8.2.2.

* Regulatory Guide 1.85 was withdrawn in 2004. See Regulatory Guide 1.84 for NRC acceptance of current Materials Code Cases.

Regulatory Guide 1.88, Revision 2 October 1976

Collection, Storage and Maintenance of Nuclear Power Plant Quality Assurance Records

I Design and Construction Phase

Compliance or Alternate Approach Statement:

I Design and Construction Phase

Prior to the original issue of this regulatory guide and construction of the CGS records facility, Project Quality Assurance complied with the intent of 10 CFR Part 50, Appendix B, by duplicate storage of records. Project Quality Assurance also complied with the original issue and revisions of this regulatory guide by duplicate storage. Since March 1977, Project Quality Assurance has complied with Revision 2 of this regulatory guide as described below.

General Compliance or Alternate Approach Assessment:

Since March 1977, the collection, storage, and maintenance of quality assurance records by Project Quality Assurance has been in compliance with ANSI N45.2.9 and NFPA No. 232-1975 for fire protection as imposed by this regulatory guide. The record facility has a minimum of a 2-hr rating.

Procurement documents directly specify requirements for collection, storage, and maintenance of records. The requirements generally meet the intent of ANSI N45.2.9 except that storage facilities or cabinets are only required to meet a 1-hr rating.

II Operational Phase

Compliance is discussed in the OQAPD.

Regulatory Guide 1.89, Revision 1, June 1984

Qualification of Class 1E Equipment for Nuclear Power Plants

Regulatory Guide Intent:

Regulatory Guide 1.89 endorses both the requirements and recommendations of IEEE 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations." Additional regulatory position stipulations are also included.

Compliance or Alternate Approach Statement:

CGS complies with this regulatory guide for equipment requiring environmental qualification procured after February 22, 1983.

General Compliance or Alternate Approach Statement:

For equipment requiring environmental qualification installed prior to February 22, 1983, CGS follows the guidance in NUREG-0588 Cat II.

In view of the NRC Memorandum and Order (CLI-80-21), dated May 27, 1980, all environmental qualifications of Class 1E equipment located in harsh environments are reevaluated for compliance with NUREG-0588, Category II. Where significant deviation from those guidelines is found in specific equipment qualifications, additional testing and/or analysis is performed to demonstrate the adequacy of the equipment to perform its safety-related function. *For equipment whose qualification program has not been completed, a justification for interim operation in accordance with 10 CFR 50.49 is performed as described in the "WNP-2 Environmental Qualification Report for Safety-Related Equipment," Reference 3.11-1.*

Specific Evaluation Reference:

See Section 3.11.

Regulatory Guide 1.90, Revision 0, November 1974

In-Service Inspection of Prestressed Concrete Containment Structures with Grouted Tendons

Compliance or Alternate Approach Statement:

This regulatory guide is not applicable because CGS does not have a prestressed concrete containment structure with grouted tendons.

General Compliance or Alternate Approach Assessment:

Not applicable.

Specific Evaluation Reference:

Not applicable.

Regulatory Guide 1.91, Revision 0, January 1975

Evaluation of Explosions Postulated to Occur on Transportation Routes Near Nuclear Power Plant Sites

Compliance or Alternate Approach Statement:

This regulatory guide is not applicable to CGS since it applies to the evaluation of construction permit applications docketed on or after March 14, 1975. However, CGS complies with the guidance set forth in this regulatory guide.

General Compliance or Alternate Approach Assessment:

It has been determined that the peak overpressures produced by postulated explosions occurring on transportation routes near the plant are no greater than the wind pressures caused by the design basis tornado. Therefore, postulated explosions will not cause an accident or prevent the safe shutdown of the plant.

Specific Evaluation Reference:

See Sections 2.2.1, 2.2.2.2, and 2.2.2.4.

Regulatory Guide 1.92, Revision 1, February 1976

Combining Modal Responses and Spatial Components in Seismic Response Analysis

Compliance or Alternate Approach Statement:

This regulatory guide is not a requirement for CGS since it applies to the evaluation of construction permit applications docketed after February 1976. CGS complies with the intent of the guidance set forth in this regulatory guide by implementing the regulatory guide criteria or by an alternate approach.

General Compliance or Alternate Approach Assessment:

The method of combining modal responses has been implemented in accordance with the guide's recommendations.

The combining of spatial components was performed prior to the issuance of the guide and follows the method presented in the PSAR. The method used is an industry-accepted alternate method. The method considers the combination of the maximum structural responses to the more critical one of the two horizontal components and the vertical component of earthquake motion, using the absolute sum method. Alternatively, when the regulatory guide is followed, two horizontal components and one vertical component of earthquake motion are combined by the square root sum of the squares method.

Specific Evaluation Reference:

See Sections 3.7.2.6 and 3.7.2.7.

Regulatory Guide 1.93, Revision 0, December 1974

Availability of Electric Power Sources

Compliance or Alternative Approach Statement:

CGS complies with the regulatory position for operating the plant whenever the available electric power sources are less than the limiting conditions for operation (LCO) as defined in the regulatory guide.

General Compliance or Alternate Approach Assessment:

Operating procedures incorporate the requirements of this guide.

Specific Evaluation Reference:

See the Technical Specifications.

Regulatory Guide 1.94, Revision 1, April 1976

Quality Assurance Requirements for Installation, Inspection, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants.

I Design and Construction Phase

Compliance or Alternate Approach Statement:

This regulatory guide is not applicable to CGS since it applies to the evaluation of construction permits docketed after October 15, 1976. However, CGS complies with the intent of the guidance set forth in the guide.

General Compliance or Alternate Approach Assessment:

The guidelines included in ANSI 45.2.5-1974 for installation, inspection and testing of structural concrete and structural steel, including nonpressure vessel elements of the primary containment vessel during the construction phase of CGS are reflected in the structural concrete and structural steel contract specifications for project construction. The QA requirements of ANSI 45.2 were incorporated in these specifications.

Specific Evaluation Reference:

See Sections 3.8.3.2, 3.8.4.2, 3.8.5.2, and Table 3.8-4.

II Operational Phase

Compliance is discussed in the Topical Report referenced in the OQAPD.

Regulatory Guide 1.95, Revision 1, January 1977

Protection of Nuclear Power Plant Control Room Operators Against an Accidental Chlorine Release

Compliance or Alternate Approach Statement:

This regulatory guide is not applicable to CGS since chlorine gas is not stored at CGS or nearby facilities and the expected quantities of chlorine shipped within five miles is less than the threshold volumes specified in Regulatory Guide 1.78.

Specific Evaluation Reference:

See Section 6.4.4.2.

Regulatory Guide 1.97, Revision 2, December 1980

Instrumentation for Light Water Cooled Nuclear Power Plants to Assess Plant Conditions During and Following an Accident

Compliance or Alternate Approach Statement:

The CGS safety-related display instrumentation meets the intent of Regulatory Guide 1.97.

General Compliance or Alternate Approach Assessment:

Instrumentation is provided in the main control room to monitor plant variables and systems during and following an accident. The instrumentation is qualified to remain functional as required by the regulatory guide.

Portable multichannel gamma-ray spectrometer instrumentation provided for use by field teams during emergencies is not used at CGS, contrary to the recommendation contained in Regulatory Guide 1.97, Revision 2, Table 2, Plant and Environs Radioactivity (portable instrumentation). Regulatory Guide 1.97, recommends the use of these instruments for release assessment and analysis. Alternative methods that produce more reliable indication of fuel failure during a radioactive release are used instead, such as air sample analysis and validation of dose projections using field team sample results.

Specific Evaluation Reference:

See Section 7.5.

Regulatory Guide 1.100, Revision 1, August 1977

Seismic Qualification of Electric Equipment for Nuclear Power Plants

Regulatory Guide Intent:

Regulatory Guide 1.100 endorses both the requirements and recommendations of IEEE 344-1975, "IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations," when such qualification is performed in conjunction with Regulatory Guide 1.89, and subject to the regulatory position stipulations.

Compliance or Alternate Approach Statement:

This regulatory guide is applicable to CGS as clarified in Section 1.8.3 for Regulatory Guide 1.89, Revision 1 and Section 3.10.1.2.

General Compliance or Alternate Approach Assessment:

All Class 1E equipment seismic qualifications are evaluated against the requirements set forth within IEEE 344-1975 as clarified in Section 3.10.1.2. The evaluations are documented and demonstrate adequacy of the methods and results of the qualifications as equal or conservative to the requirements of IEEE 344-1975. These include evaluations of seismic and hydrodynamic load combinations.

Specific Evaluation Reference:

See Section 3.10.

Regulatory Guide 1.101, Revision 1, March 1977

Emergency Planning for Nuclear Power Plants

Compliance or Alternate Approach Statement:

CGS complies with the intent set forth in this regulatory guide.

General Compliance or Alternate Approach Statement:

See NUREG-0654.

Specific Evaluation Reference:

See the CGS Emergency Plan.

Regulatory Guide 1.102, Revision 1, September 1976

Flood Protection for Nuclear Power Plants

Compliance or Alternate Approach Statement

CGS complies with the guidance set forth in this regulatory guide.

General Compliance or Alternate Approach Assessment:

The safety-related buildings and spray ponds are located far above the water level estimated for the largest historical flood. Based on the criteria stipulated in Regulatory Guide 1.102, the CGS plant site is classified as a “Dry Site.”

Specific Evaluation Reference:

See Section 2.4.

Regulatory Guide 1.103, Revision 1, October 1976

Post-Tensioned Prestressed Systems for Concrete Reactor Vessels and Containments

Compliance or Alternate Approach Statement:

This regulatory guide is not applicable since CGS does not have a concrete reactor vessel or containment.

General Compliance or Alternate Approach Assessment:

Not applicable.

Specific Evaluation Reference:

Not applicable.

Regulatory Guide 1.104, Revision 0, February 1976

Overhead Crane Handling Systems for Nuclear Power Plants

Compliance or Alternate Approach Statement:

CGS complies with the intent of the guidance set forth in this regulatory guide by an alternate approach.

General Compliance or Alternate Approach

The following safeguards are included in the design of the overhead crane:

a. Redundant low limit, main hoist,

b. Redundant equalizer bar limit switch,

c. “Critical Control Path” series of limit switches for the spent fuel cask handling mode, and

d. Main hoist “paddle” type upper limit switch to prevent the inadvertent “two-blocking” condition.

Specific Evaluation Reference:

See Sections 3.8.4.1.1.5 and 9.1.4.2.2.

Regulatory Guide 1.105, Revision 1, November 1976

Instrument Setpoints

Compliance or Alternate Approach Statement:

This regulatory guide is not applicable to CGS since it applies to the evaluation of construction permit applications docketed after December 15, 1976.

General Compliance or Alternate Approach Assessment:

Instrumentation is provided in a main control room to monitor plant variables and systems. The range of instrumentation is selected to cover the anticipated ranges of variables for the following plant conditions:

a. Normal operation,
b. Anticipated operational occurrences, and
c. Accident conditions.

To ensure adequate safety, the following plant parameters and systems are monitored and provided with appropriate controls to maintain them within prescribed operating ranges:

1. Variables and systems that affect the fission process,
2. Variables and systems that affect the reactor core,
3. Reactor coolant pressure boundary, and
4. Containment and associated systems.

Specific Evaluation References:

See Section 7.1.2.5.

Regulatory Guide 1.106, Revision 1, March 1977

Thermal Overload Protection for Electric Motors on Motor Operated Valves

Compliance or Alternate Approach Statement:

This regulatory guide is not applicable to CGS since it applies to the evaluation of construction permit applications docketed after July 15, 1976. However, CGS design complies with the intent of the guidance set forth in Section C.2 of the regulatory guide.

General Compliance or Alternate Approach Assessment:

Class 1E motor-operated valve (MOV) overloads are chosen two sizes above those which would be required based on normal full load running current. The resultant overload protection (approximately 140%) permits MOV motors to operate for extended periods at moderate overloads; tripping occurs just prior to motor damages.

Specific Evaluation Reference:

See Section 8.3.1.1.9.

Regulatory Guide 1.107, Revision 1, February 1977

Qualifications for Cement Grouting for Prestressing Tendons in Containment Structures

Compliance or Alternate Approach Statement

This regulatory guide is not applicable to CGS because CGS does not have a prestressed concrete containment structure with grouted tendons.

General Compliance or Alternate Approach Assessment:

Not applicable.

Specific Evaluation Reference:

Not applicable.

Regulatory Guide 1.108, Revision 0, August 1976

Periodic Testing of Diesel Generators Used as Onsite Electric Power Systems as Nuclear Power Plants.

Compliance or Alternate Approach Statement:

This regulatory guide is not applicable to CGS since the method described for compliance with the regulations indicated in the guide are applicable to plants having construction permit applications docketed after April 1, 1977. However, CGS complies with the intent of this regulatory guide.

General Compliance or Alternate Approach Assessment:

Preoperational and periodic testing of the diesel generators is performed as referenced in Sections 14.2.12.1.40 and the Technical Specifications. As discussed in Section 8.3, provisions for testability are included in the design of the standby power system.

Specific Evaluation Reference:

See Sections 8.3 and 14.2.12.1.

Regulatory Guide 1.109, Revision 0, March 1976

Calculation of Annual Doses to Man From Routine Releases of Reactor Effluents.

Compliance or Alternate Approach Statement:

> CGS complies with the intent of the guidance set forth in this regulatory guide using an alternate approach.

General Compliance or Alternate Approach Statement:

> CGS is meeting the guidance of this regulatory guide by using Battelle Northwest models which are acceptable to the NRC.

Specific Evaluation Reference:

> See Sections 11.2.3.3, 11.3.3.3, and 5.2 of the Environmental Report.

Regulatory Guide 1.110, Revision 0, March 1976

Cost-Benefit Analysis for Radwaste Systems for Light-Water-Cooled Nuclear Power Reactors

Compliance or Alternate Approach Statement:

This regulatory guide is not applicable to CGS since a cost-benefit analysis, as described in Appendix I of 10 CFR 50 Section II-D is not required for CGS.

General Compliance or Alternate Approach Assessment:

Not applicable.

Specific Evaluation Reference:

See Section 11.2.3.4.

Regulatory Guide 1.111, Revision 1, July 1977

Method for Estimating Atmospheric Transport and Dispersion of Gaseous Effluents in Routine Releases from Light-Water Cooled Reactors

Compliance or Alternate Approach Statement:

CGS complies with the intent of the guidance set forth in this regulatory guide by an alternate approach.

General Compliance or Alternate Approach Assessment:

Analyses of atmospheric transport and dispersion of gaseous effluents at CGS are performed using the standard NRC diffusion models in NUREG/CR-2919, XOQ/DOQ: Computer Program for the Meteorological, Evaluation of Routine Effluent Releases at Nuclear Power Stations, September 1982.

Specific Evaluation References:

See Section 2.3.5.

Regulatory Guide 1.112, Revision 0-R, May 1977

Calculation of Releases of Radioactive Materials in Gaseous and Liquid Effluents from Light-Water Cooled Power Reactors.

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in this regulatory guide.

General Compliance or Alternate Approach Assessment:

The methods for calculating annual average releases of radioactive material in liquid and gaseous effluents from the plant were originally based on the GALE Code as suggested in this regulatory guide. See the sections referenced below for discussions of the methods currently used.

Specific Evaluation Reference:

See Sections 11.2.3.2 and 11.3.3.3.

Regulatory Guide 1.113, Revision 1, April 1977

Estimating Aquatic Dispersion of Effluents From Accidental and Routine Reactor Releases For the Purpose of Implementing Appendix I.

Compliance or Alternate Approach Statement:

CGS complies with the intent of the guidance set forth in this regulatory guide using an alternate approach.

General Compliance or Alternate Approach Assessment:

Routine and accidental releases of radioactive liquid, heat, and chemical discharges to the Columbia River via the CGS cooling tower blowdown line are discussed in Section 2.4.12. CGS final Environmental Report (ER) 6.1.1.1 describes in detail the advection/diffusion equations used in the near-field thermal analysis. This analysis provides dispersion characteristics, presented in ER 5.1, to 500 ft below the point of discharge. A simplified and conservative approach to estimating the far-field concentrations of routine releases is presented in ER 5.2.2. The affects of an accidental release of radioactive liquid to the ground within the CGS site area were investigated and are discussed in Section 2.4.13.3.

Specific Evaluation Reference:

See Sections 2.4.12 and 2.4.13.3 and Environmental Report Sections 5.1, 5.2.2, and 6.1.1.1.

Regulatory Guide 1.114, Revision 1, November 1976

Guidance to Operator at the Controls of a Nuclear Power Plant.

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in this regulatory guide.

General Compliance or Alternate Approach Assessment:

Plant administrative procedures implement the requirements of this regulatory guide.

Specific Evaluation Reference:

Not applicable.

Regulatory Guide 1.115, Revision 0, March 1976

Protection Against Low-Trajectory Turbine Missiles

Compliance or Alternate Approach Statement:

This regulatory guide is not applicable to CGS since it applies to the evaluation of construction permit applications docketed after November 15, 1976.

General Compliance or Alternate Approach Assessment:

Extensive amounts of concrete used in the construction of CGS serve as radiation shielding and formidable barriers protecting essential systems from low trajectory missiles.

Specific Evaluation Reference:

See Section 3.5.1.3.

Regulatory Guide 1.116, Revision 0, June 1976

Quality Assurance Requirements for Installation, Inspection, and Testing of Mechanical Equipment and Systems

I Design and Construction Phase

Compliance or Alternate Approach Statement:

CGS complied with the intent of the guidance set forth in this regulatory guide by an alternate approach.

General Compliance or Alternate Approach Assessment:

The requirements for installation, inspection, and testing are specified in procurement documents which require a quality assurance program in compliance with ANSI N45.2.

Specific Evaluation Reference:

None

II Operational Phase

Compliance is discussed in the OQAPD.

Regulatory Guide 1.117, Revision 0, June 1976

Tornado Design Classification

Compliance or Alternate Approach Statement:

This regulatory guide is not applicable to CGS since it applies to the evaluation of construction permit applications docketed after February 15, 1977.

General Compliance or Alternate Approach Assessment:

Essential systems are protected from tornadoes by structures designed for design basis tornadoes (DBT). See Regulatory Guides 1.27 and 1.76.

Specific Evaluation Reference:

See Sections 3.3.2.4 and 9.2.5.

Regulatory Guide 1.118, Revision 0, June 1976

Periodic Testing of Electric Power and Protection Systems.

Compliance or Alternate Approach Statement:

This regulatory guide is not applicable to CGS since the construction permit for CGS was issued prior to February 15, 1977.

General Compliance or Alternate Approach Assessment:

Electric power and protection systems are tested periodically as specified in the Technical Specifications. As described in Section 13.5.2, surveillance procedures have been prepared for periodic testing of these systems.

Specific Evaluation Reference:

See the Technical Specifications.

Regulatory Guide 1.120, Revision 0, June 1976

Fire Protection Guidelines for Nuclear Power Plants

Compliance or Alternate Approach Statement:

This regulatory guide is not applicable to CGS since it applies to the evaluation of construction permit applications docketed after February 28, 1977. However, the NRC requested a reevaluation of the fire protection program of CGS and a comparison with the guidelines in Appendix A to Branch Technical Position APCSB 9.5-1, "Guidelines for Fire Protection For Nuclear Power Plants, Docketed Prior to July 1, 1976." CGS complies with the intent of the guidance set forth in Appendix A to Branch Technical Position APCSB 9.5-1.

General Compliance or Alternate Approach Assessment:

Appendix F includes the fire hazard analysis and compares in detail the fire protection provisions for CGS with the guidelines in Appendix A to Branch Technical Position APCSB 9.5-1.

Regulatory Guide 1.122, Revision 0, September 1976

Development of Floor Design Response Spectra for Seismic Design of Floor Supported Equipment or Components

Compliance or Alternate Approach Statement:

CGS complies with the intent of the guidance set forth in this regulatory guide by an alternate approach.

General Compliance or Alternate Approach Assessment:

CGS complies with some of the regulatory positions and where not in compliance, alternate methods are used as discussed in Sections 3.7.2.5 and 3.7.2.6.

Specific Evaluation Reference:

See Sections 3.7.2.5 and 3.7.2.6.

Regulatory Guide 1.123, Revision 0, October 1976

Quality Assurance Requirements for Control of Procurement of Items and Services for Nuclear Power Plants

I Design and Construction Phase

Compliance or Alternate Approach Statement:

CGS complied with the intent of the guidance set forth in this regulatory guide by an alternate approach.

General Compliance or Alternate Approach Assessment:

ANSI N45.2.13-1976, the subject of this regulatory guide, requires certain supplier selection, evaluation, and pre- and post-award activities.

Prequalification of suppliers was generally not performed. The procurement documents required prospective suppliers to submit information pertaining to experience, facilities, personnel, and quality program with their bids for evaluation prior to award of a contract.

Pre-award evaluations were restricted to the information submitted with bid and selected clarifications when an adequate evaluation could not be accomplished with the information supplied. Post-award evaluations were performed in conjunction with the quality assurance program evaluation and approval after award of a contract.

Inspection and hold points were not established through agreement with the bidder but through contract requirements to notify Energy Northwest of all inspections and tests which were selectively witnessed by Energy Northwest.

Specific Evaluation Reference:

None

II Operational Phase

Compliance is discussed in the OQAPD.

Regulatory Guide 1.124, Revision 0, November 1976

Design Limits and Loading Combinations for Class 1 Linear Type Component Supports

Compliance or Alternate Approach Statement:

This regulatory guide is not applicable to CGS since it applies to the evaluation of construction permit applications docketed after July 1, 1977. However, CGS complies with the intent of the guidance set forth in this regulatory guide by an alternate approach.

General Compliance or Alternate Approach Assessment:

Design and fabrication requirements for CGS, including those requirements for linear type components supports, are in accordance with the ASME Code Section III Subsection NF, Winter 1973 Addenda. The actual design criteria were established prior to Winter 1973 Addenda and are conservative with respect to the Winter 1973 Code. Regulatory Guide 1.124 provides design limits and appropriate combinations of loadings which reflect the requirements set forth in the 1974 Edition of the ASME Code Section III, Subsection NF, along with additional requirements. Although the detailed requirements of the regulatory guide have not been incorporated as project criteria, review of the design criteria used for CGS indicates that the intent of this regulatory guide is met.

Specific Evaluation Reference:

See Sections 3.9.3.4 and 5.4.14.

Regulatory Guide 1.125, Revision 0, March 1977

Physical Models for Design and Operation of Hydraulic Structures and Systems for Nuclear Power Plants

Compliance or Alternate Approach Statement:

The guide is not applicable to CGS since it applies to the evaluation of construction permit application docketed on or after November 1, 1977. Furthermore, the guide is not applicable to CGS for reasons stated below.

General Compliance or Alternate Approach Assessment:

Physical hydraulic model testing is not used for CGS for predicting the performance of hydraulic structures, systems, and components located outside the primary containment vessel or provided for the prevention of accidents and the mitigation of the consequences of accidents. Therefore, the details and documentation of data and studies required by the guide to support such testing is not applicable.

Specific Evaluation Reference:

Not applicable.

Regulatory Guide 1.127, Revision 0, April 1977

Inspection of Water-Control Structures Associated With Nuclear Power Plants

Compliance or Alternate Approach Statement:

This regulatory guide is not applicable to CGS since water-control structures as defined in this regulatory guide do not exist.

General Compliance or Alternate Approach Assessment:

Not applicable.

Specific Evaluation Reference:

Not applicable.

Regulatory Guide 1.128, Revision 0, April 1977

Installation Design and Installation of Large Lead Storage Batteries for Nuclear Power Plants.

Compliance or Alternate Approach Statement:

This regulatory guide is not applicable to CGS since it applies to the evaluation of construction permit applications docketed after December 1, 1977. However, CGS complies with the intent of the guidance set forth in this regulatory guide by an alternate approach.

General Compliance or Alternate Approach Assessment:

Safety-related battery installation design criteria conforms to IEEE 484-1975.
A Class 1E ventilation system is also provided which is capable of limiting hydrogen concentrations to 1%.

Storage prior to installation was not in strict compliance with Section 5.1.3 of this regulatory guide. However, preoperational tests established whether or not any damage or loss of capacity resulted from storage.

Specific Evaluation Reference:

See Sections 8.3.2.1.5, 8.3.2.1.6, 8.3.2.2.1.1, and 8.3.2.2.1.2.

Regulatory Guide 1.129, Revision 0, April 1977

Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Nuclear Power Plants

Compliance or Alternate Approach Statement:

Although Regulatory Guide 1.129 is not directly applicable to CGS, Energy Northwest's maintenance procedures conform to IEEE 450- 2002, "IEEE Recommended Practice for Maintenance, Testing, and Replacement of Large Lead Storage Batteries for Generating Stations and Substations." The frequency for "service" testing is in accordance with Technical Specifications or Licensee Controlled Specifications.

General Compliance or Alternate Approach Assessment:

See Section 8.3.2.1.7.

Specific Evaluation Reference:

See Section 8.3.2.1.7.

Regulatory Guide 1.137, Revision 1, October 1979

Fuel Oil Systems for Standby Diesel Generators

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in this guide with the exception of the following:

Piping on the engine skid is ANSI B 31.1, Seismic Category I, Quality Class I, as noted in Section 9.5.4.1.

Item 11, cathodic protection surveillance. The standby diesel fuel oil storage tanks are protected with cathodic protection by anodes which are located in the near vicinity, but there are no pigtails connected to the fuel oil system piping, thus no leads to maintain. CGS does not perform the 90% distillation test before putting the fuel in the tanks as noted in Section 9.5.4.4 and the Technical Specifications.

The diesel fuel oil supply is gravity feed down to the low fuel oil alarm level. The pump suction, however, is 2.3 ft higher than the bottom of the tank. Therefore, if the transfer pump fails, the last few hours of running before the day tank is empty would be at a pump suction lift of up to 2.3 ft.

The auxiliary boiler storage tank is considered part of the diesel fuel oil system in that it is an additional diesel fuel oil storage tank. This deviates from the ANSI N195-1976 standard because of the permanent interconnection between the standby power system and the auxiliary boiler system. The auxiliary boiler storage tank and its connective piping are not Safety Class 3. The auxiliary boiler storage tank and its connecting auxiliary boiler system are not in a vital area, although ANSI N195-1976 specifies that the fuel oil system is a vital system and shall be located in a vital area. However, loss of the stored fuel oil in the auxiliary boiler storage tank or its connective piping will not affect the safety function of the diesel fuel oil system.

The diesel storage minimum required volume does not include volume for testing, as specified by ANSI N195-1976. Instead, Energy Northwest procedurally provides for makeup, as needed, during testing activities to ensure that the minimum required volume is maintained.

Specific Evaluation Reference:

See Section 9.5.4.4.

Regulatory Guide 1.143, Revision 1, October 1979

Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in this regulatory guide.

General Compliance or Alternate Approach Assessment:

CGS began implementing the guidance set forth in this regulatory guide in July 1982. Prior to this time the solid, liquid, and gaseous radioactive waste systems were being designed and fabricated as ASME Section III, Class 3, systems. Therefore, although the guidance in the regulatory guide does not call for N-stamped components, in many cases N-stamped components are found in the radwaste systems. To avoid the confusion which may result from the implementation of this regulatory guide these systems, and components which follow the guidance found in the regulatory guide are indicated as Quality Class II+ and Code Group D+.

Specific Evaluation Reference:

See Sections 3.2.4 and 3.2.6.

Regulatory Guide 1.144, Revision 1, September 1980

Auditing of Quality Assurance Programs for Nuclear Power Plants

I Design and Construction Phase

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in this regulatory guide as described below.

General Compliance or Alternate Approach Assessment:

Contractors and suppliers complied with the requirements imposed by procurement documents.

Energy Northwest, the architect-engineer (Burns and Roe), and the construction manager (Bechtel) complied with the guidance set forth in this regulatory guide except for the following.

The requirements of ANSI N45.2.12-1977 as modified and interpreted by the regulatory position were applied to the Bechtel quality program for safety-related items except as modified or interpreted below:

a. Reference: Standard Sections 4.3.2.4 and 4.5.1 (Investigation). As an equivalent alternative to the requirement for the audited organization to investigate any adverse audit finding to determine and schedule appropriate corrective action, Bechtel's auditing organization may determine the investigatory action and corrective action including action to prevent recurrence pertinent to adverse audit finding. These actions are agreed to by the audited organization. Further, in Section 4.5.1, as equivalent alternative to the 30-day response time, a response time appropriate to the finding is agreed to by the audited and auditing organizations.

b. Reference: Regulatory Section C.7, Standard Section 5.2 (Audit Records). Audit records shall include documents as defined in the standard and other documents if necessary to support audit findings.

Early project procurements specified audit program requirements in terms of Appendix B to 10 CFR 50 and ANSI N45.2. As appropriate, future procurements required that audit programs comply with ANSI Standard N45.2.12.

II Operational Phase

Compliance is discussed in the OQAPD.

Regulatory Guide 1.145, Revision 1, November 1982/February 1983

Atmospheric Dispersion Models for Potential Accident Consequence Assessments at Nuclear Power Plants

Regulatory Guide Intent

This guide provides acceptable methodology to determining site-specific relative concentrations for assessing the potential offsite radiological consequences of postulated accidental releases of radioactive material to the atmosphere.

Application Assessment

Assessed capability in design.

Compliance or Alternate Approach Statement:

Identified BOP scope of supply analysis, design, and/or equipment used in this facility is in full compliance with the regulatory guide.

General Compliance or Alternate Approach Assessment

Two of the procedures contained in the PAVAN code were implemented. The procedures were run with the desert sigma and with the Pasquill-Gifford sigma enabled. The most conservative χ/Q values were used in the accident analysis.

Specific Evaluation Reference:

See Section 2.3 and Chapter 15.0.

Regulatory Guide 1.146, Revision 0, August 1980

Qualification of Quality Assurance Program Audit Personnel for Nuclear Power Plants

I Design and Construction Phase

Compliance or Alternate Approach Statement:

CGS complied with the guidance set forth in this regulatory guide as described below.

General Compliance or Alternate Approach Assessment:

Energy Northwest, the architect-engineer (Burns and Roe), and the construction manager (Bechtel) complied with the guidance set forth in this regulatory guide.

Contractors and suppliers comply with the requirements imposed by procurement documents.

Early project procurements specified audit program requirements in terms of Appendix B 10 CFR 50 and ANSI N45.2. Where appropriate, future procurements required that auditor qualification comply with ANSI Standard N45.2.23.

II Operational Phase

Compliance is discussed in the OQAPD.

Regulatory Guide 1.147

Inservice Inspection of Code Case Acceptability ASME Section XI Division I.

By the reference below, the NRC approved application of Code Case N416 for CGS which at that time was not addressed in Regulatory Guide 1.147. The approval letter required that Energy Northwest document application of the code case in the FSAR.

The code case was first used for CGS in 1988 for deferral of hydrostatic testing of main steam drip line modifications.

As the code case has now been accepted by Regulatory Guide 1.147, Energy Northwest does not plan to document future use of the code case.

Reference:

Letter from T. M. Novak (NRC) to G. C. Sorensen (SS), "Use of ASME Code Case N-416 for the WNP-2, WPPSS Nuclear Project No. 2 (WNP-2)," dated August 8, 1985.

Regulatory Guide 1.155, Reissued August 1988

Station Blackout

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in this regulatory guide.

Compliance or Alternate Approach Assessment:

Regulatory Guide 1.155 was issued to describe a method acceptable to the NRC staff for complying with the NRC regulation that requires nuclear power plants to be capable of coping with a station blackout for a specified duration. The NRC acceptance of the CGS proposed plan for providing this capability is provided in the reference.

Specific Evaluation Reference:

See Appendix 8A.

Reference:

Letter from R. R. Assa to G. C. Sorensen, "Supplemental Safety Evaluation (SSE) of the Washington Public Power Supply System Nuclear Project No. 2 (WNP-2) Station Blackout Analysis (TAC M68626)," dated June 26, 1992.

Regulatory Guide 1.160, Revision 1, January 1995

Monitoring the Effectiveness of Maintenance at Nuclear Power Plants

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in this regulatory guide.

General Compliance or Alternate Approach Assessment:

Compliance with the guidance provided is ensured by the implementation of a maintenance program and implementing procedures at CGS.

Specific Evaluation References:

Not applicable.

Regulatory Guide 1.196, May 2003

Control Room Habitability at Light-Water Nuclear Power Reactors

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in this regulatory guide.

General Compliance or Alternate Approach Assessment:

Compliance with the guidance provided is ensured by the implementation of a Control Room Envelope Habitability (CREH) Program and implementing procedures at CGS.

Specific Evaluation References:

Not applicable.

Regulatory Guide 1.197, May 2003

Demonstrating Control Room Envelope Integrity at Nuclear Power Reactors

Compliance or Alternate Approach Statement:

CGS complies with the guidance set forth in this regulatory guide.

General Compliance or Alternate Approach Assessment:

Compliance with the guidance provided is ensured by the implementation of a Control Room Envelope Habitability (CREH) Program and implementing procedures at CGS.

Specific Evaluation References:

Not applicable.