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(6) Monitoring of essential generators, transformers, and circuits is provided in the main control room.

1.2.1.2.5.3 Power Conversion Systems 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.

1.2.1.2.6 Power conversion Systems Criteria

Components of the power conversion systems shall be designed to perform the following basic objectives:

(1) produce electrical power from the steam coming from the reactor, condense the steam into water, and return the water to the reactor as heated feedwater with a major portion of its gases and particulate impurities removed; and

(2) assure 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.

1.2.2 Plant Description

1.2.2.1 Site Characteristics

1.2.2.1.1 Site Location

The plant is located on a site adjacent to or close to a body of water with sufficient capacity for either once-through or recirculated cooling or a combination of both methods.

1.2.2.1.2 Description of Plant Environs

1.2.2.1.2.1 Meteorology

The safety-related structures and equipment are designed to retain required functions for the loads resulting from any tornado with characteristics not exceeding the values provided in Table 2.0-1.

Tornado missiles are discussed in Section 3.5.

1.2.2.1.2.2 Hydrology

The safety design basis of the plant provides that structures of safety significance will be unaffected by the hydrologic parameter envelope defined in Chapter 2.

1.2.2.1.2.3 Geology and Seismology

The structures of safety significance for the plant are designed to withstand a safe shutdown earthquake (SSE) which results in a freefield peak acceleration of 0.3g.

1.2.2.1.2.4 Shielding

Shielding is provided throughout the plant, as required to maintain radiation levels to operating personnel and to general public within the applicable limits set forth in 10CFR20 and 10CFR100. It is also designed to protect certain plant components from radiation exposure resulting in unacceptable alterations of material properties or activation.

1.2.2.2 Nuclear Steam Supply Systems

The nuclear steam supply system includes a direct-cycle forced-circulation boiling water reactor that produces steam for direct use in the steam turbine. A heat balance showing the major parameters of the nuclear steam supply system for the rated power conditions is shown in Figure 1.1-2.

1.2.2.2.1 Reactor Pressure Vessel System

The reactor pressure vessel system (RPVS) contains the reactor pressure vessel with the reactor internal pump (RIP) casings; core and

supporting structures; the steam separators and dryers; the control rod guide tubes; the spargers for the feedwater, RHR and core flooder system; the control rod drive housing; the in-core instrumentation guide tubes and housings; and other components. The main connections to the vessel include steamlines, feedwater lines, reactor internal pumps, control rod drives and in-core nuclear instrument detectors, core flooder lines, residual heat removal lines, head spray and vent lines, core plate differential pressure lines, internal pump differential pressure lines, and water level instrumentation.

A venturi-type flow restrictor is a part of the reactor pressure vessel nozzle configuration for each steamline. These restrictors limit the flow of steam from the reactor vessel before the main steamline isolation valves are closed in case of a main steamline break outside the containment

Control rod drive housing supports are located internal to the reactor vessel and the control rod drive. The supports limit the travel of a control rod in the event that a control rod housing is ruptured.

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 1040 psia. The vessel is fabricated of low alloy steel and is clad internally with stainless steel or Ni-Cr-Fe Alloy (except for the top head, RIP motor casing, nozzles other than the steam outlet nozzle, 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 main steamlines. Each steamline is provided with two isolation valves in series, one on each side of the containment barrier.

1.2.2.2.2 Nuclear Boiler System

1.2.2.2.2.1 Main Steamline Isolation Valves

All pipelines that both penetrate the containment and offer a potential release path for radioactive material are provided with redundant isolation capabilities. Automatic isolation valves are provided in each main steamline. Each is powered by both steam pressure and spring force. These valves fulfill the following objectives:

(1) 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 containment or a malfunction of the pressure control system resulting in excessive steam flow from the reactor vessel;

(2) limit the release of radioactive materials by isolating the reactor coolant pressure boundary in case of the detection of high steam line radiation.

1.2.2.2.2.2 Main Steamline Flow Instrumentation

The steam flow instrumentation is connected to the venturi type steam nozzle of the RPV. The instrumentation provides high nozzle flow isolation signals in case of a main steam line break.

1.2.2.2.2.3 Nuclear System Pressure Relief System

A pressure relief system consisting of safety/relief valves mounted on the main steamlines is provided to prevent excessive pressure inside the nuclear system as a result of operational transients or accidents.

1.2.2.2.2.4 Automatic Depressurization System

The ADS rapidly reduces reactor vessel pressure in a loss-of-coolant accident, enabling the low-pressure RHR to deliver cooling water to the reactor vessel.

The ADS uses some of the safety relief valves that are part of the nuclear system pressure relief system. The safety relief valves used for ADS are set to open on detection of appropriate low reactor water level and high drywell pressure signals. The ADS will not be activated unless either RHR/low pressure flooding loop pump are operating. This is to ensure that adequate coolant will be available to maintain reactor water level after depressurization.

1.2.2.2.2.5 Reactor Vessel Instrumentation

In addition to instrumentation for the nuclear

safety systems and engineered safety features, 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 reactor vessel head inner seal ring leakage.

1.2.2.2.3 Reactor Recirculation System

The reactor internal pumps (RIPs) are internal pumps which provide a continuous internal circulation path for the core coolant flow. The RIPs are located at the bottom of the vessel. The pump motors are enclosed in casings which are a part of the vessel. A break in the casing will result in a leak flow that is less than the ECCS capacity allowing full core coverage. The internal pumps are a wet motor design with no shaft seals, thereby providing increased reliability, reduced maintenance requirements and decreased operational radiation exposure. The RIP has a low rotating inertia. Coupled with the solid state adjustable speed drives, the RIP can respond quickly to load transients and operator demands.

1.2.2.3 Control and Instrumentation Systems

1.2.2.3.1 Rod Control and Information System

The rod control and information system (RCIS) provides the means by which control rods are positioned from the control room for power control. The system operates the rod drive motors to change control rod position. For operation in the normal gang movement mode, one gang of control rods can be manipulated at a time. The system includes the logic that restricts control rod movement (rod block) under certain conditions as a backup to procedural controls.

1.2.2.3.2 Control Rod Drive System

When scram is initiated by the RPS, the control rod drive (CRD) system inserts the negative reactivity necessary to shut down the reactor. Each control rod is normally controlled by an electric motor unit. When a scram signal is received, high-pressure water stored in nitrogen charged accumulators forces the control rods into the core and the electric motor drives are signalled to drive the rods into the core. Thus, the hydraulic scram action is backed up by an electrically energized insertion of the control rods.

1.2.2.3.2.1 Control Rod Braking Mechanism

An electro-mechanical braking mechanism is incorporated in each control rod to limit the velocity at which a control rod can fall out of the core should a hydraulic line break or failure of flange bolts or a spool piece. This action limits the rate of reactivity insertion resulting from a rod drop accident.

1.2.2.3.2.2 Control Rod Ejection

A nuclear excursion is prevented in case of a housing failure and thus the fuel barrier is protected because, as discussed in Subsection 1.2.2.2.1, the housing and the drive are restrained internally to the vessel to prevent the control rod ejection.

1.2.2.3.3 Feedwater Control System

The feedwater control system automatically controls the flow of feedwater into the reactor pressure vessel to maintain the water within the vessel at predetermined levels. A fault-tolerant triplicated, digital controller using conventional three-element control scheme is used to accomplish this function.

1.2.2.3.4 Standby Liquid Control System

The standby liquid control (SLC) system provides an alternate method to bring the nuclear fission reaction to subcriticality and to maintain subcriticality 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 shutdown condition.

1.2.2.3.5 Neutron Monitoring System

The neutron monitoring (NMS) 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. There are fixed in core sensors which provide level indications during reactor startup and low power operation. The startup range neutron monitors (SRNM) and average power range monitors (APRM) allow assessment of local and overall flux conditions during power range operation. The automatic traversing in-core probe (ATIP) system provides a means to calibrate the local power range monitors. The NMS provides inputs to the rod control and information system to initiate rod blocks if preset flux limits or period limits for rod block are exceeded as well as inputs to the RPS if other limits for scram are exceeded.

Those portions of the neutron monitoring system that input signals to the RPS qualify as a nuclear safety system. The startup range neutron monitor (SRNM) and the average power range monitors (APRM) which monitor neutron flux via incore detectors provide scram logic inputs to the reactor protection system (RPS) to initiate a scram in time to prevent excessive fuel clad damage as a result of over-power transients. The APRM system also generates a simulated thermal power signal. Both upscale neutron flux and upscale simulated thermal power are conditions which provide scram logic signals.

1.2.2.3.6 Remote Shutdown System

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

1.2.2.3.7 Reactor Protection System

The reactor protection system (RPS) initiates a rapid, automatic shutdown (scram) of the reactor. It acts in time to prevent fuel cladding damage and any nuclear system process barrier damage following abnormal operational transients. The reactor protection system 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 failure occurs.

1.2.2.3.8 Recirculation Flow Control System

During normal power operation, the speed of the reactor internal pumps is adjusted to control flow. Adjusting RIP speed changes the coolant flow rate through the core and thereby changes the core power level. The system can automatically adjust the reactor power output to the load demand. The solid-state adjustable speed drives (ASD) provide variable voltage, variable frequency electrical power to the RIP motors. In response to plant needs, the recirculation flow control system adjusts the ASD power supply output to vary RIP speed, core flow, and core power.

1.2.2.3.9 Automatic Power Regulator System

The automatic power regulator system is summarized in Subsection 7.7.1.7(1).

1.2.2.3.10 Steam Bypass 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 turbine bypass system has the capability to shed 40 percent of the turbine-generator rated load without reactor trip or operation of safety relief valve. The pressure regulation system provides main turbine control valve and bypass valve flow demands so as to maintain a nearly constant reactor pressure during normal plant operation. It also provides demands to the recirculation system to adjust power level by changing reactor recirculation flow rate.

1.2.2.3.11 Process Computer (Includes PMCS, PGCS)

On-line process computers are provided to monitor and log process variables and make certain analytical computations. The performance and power generation control systems are included.

1.2.2.3.12 Refueling Platform Control Computer

The refueling platform control computer provides memory of all the fuel and platform positions, directions for the traversable area and traveling paths, directions for the speed functions for all modes of travel and control of the fuel load. The computer controls automatic or manual refueling between fuel storage and the reactor from the remote control room.

1.2.2.3.13 CRD Removal Machine Control Computer

The CRD handling machine control computer provides automatic positioning, continuous operation and prevention of erroneous operation in the step wise removal and installation of CRD's from the remote control room.

1.2.2.4 Radiation Monitoring Systems

1.2.2.4.1 Process Radiation Monitoring System

Process radiation monitoring systems monitor and control radioactivity in process and effluent streams and activate appropriate alarms and controls.

A process radiation monitoring system indicates and records radiation levels associated with selected plant process streams and effluent paths leading to the environment. All effluents form the plant which are potentially radioactive are monitored.

1.2.2.4.2 Area Radiation Monitoring System

Area radiation monitoring systems alert occupants and the control room personnel of excessive gamma radiation levels at selected locations within the plant.

1.2.2.4.3 Containment Atmospheric Monitoring System

The containment atmospheric monitoring system (CAMS) measures alarms and records radiation levels and the oxygen concentration in the primary containment under post accident conditions. It is automatically put in service upon detection of LOCA conditions.

1.2.2.5 Core Cooling System

In the event of a breach in the reactor coolant pressure boundary that results in a loss of reactor coolant, three independent divisions of ECCS are provided to maintain fuel cladding below the temperature limit as defined by 10CFR50.46. Each division contains one high pressure and one low pressure inventory makeup system.

1.2.2.5.1 Residual Heat Removal System

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

(1) removes decay and sensible heat during and after plant shutdown;

(2) injects water into the reactor vessel following a loss-of-coolant accident to reflood the core in conjunction with other core cooling systems (Subsection 1.2.2.4.8);

(3) removes heat from the containment following a loss-of-coolant accident to limit the increase in containment pressure. This is accomplished by cooling and recirculating the suppression pool water.

1.2.2.5.1.1 Low Pressure Flooder Loop

Low pressure flooding is an operating mode of each RHR system, but is discussed here because the low pressure flooder loop (LPFL) mode acts in conjunction with other injection systems. LPFL uses the pump loops of the RHR to inject cooling water into the pressure vessel. LPFL operation provides the capability of core flooding at low vessel pressure following a LOCA in time to maintain the fuel cladding below the prescribed temperature limit.

1.2.2.5.1.2 Residual Heat Removal System Containment Cooling

The residual heat removal (RHR) system is placed in operation to: (1) limit the temperature of the water in the suppression pool and the atmospheres in the drywell and suppression chamber following a design basis LOCA; (2) control the pool temperature during normal operation of the safety/relief valves and the RCIC system; and (3) 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 the service water. The fluid is then discharged back to the suppression pool, to the drywell spray header, to the suppression chamber spray header, or to the RPV.

1.2.2.5.1.3 Wetwell/Drywell Spray

A spray system is provided for wetwell/ drywell cooling in the suppression chamber and drywell air space. The wetwell/drywell spray can be initiated

manually if a high containment pressure signal is received. Each subsystem is supplied from a separate redundant RHR subsystem.

1.2.2.5.2 High Pressure Core Flooder System

High pressure core flooder (HPCF) are provided in two divisions to maintain an adequate coolant inventory inside the reactor vessel to limit fuel cladding temperatures in the event of breaks in the reactor coolant pressure boundary. The systems are initiated by either high pressure in the drywell or low water level in the vessel. They operate independently of all other systems over the entire range of system operating pressures. The HPCF system pump motors are powered by a diesel generator if auxiliary power is not available. The systems may also be used as a backup for the RCIC system.

1.2.2.5.3 Leak Detection and Isolation System

The leak detection and isolation system consists of temperature, pressure, flow, and fission-product sensors with associated instrumentation, alarms, and isolation functions. This system detects and annunciates leakage (and closes isolation valves, as required) in the following systems:

(1) main steamlines;

(2) reactor water cleanup (RWCU) system;

(3) residual heat removal (RHR) system;

(4) reactor core isolation cooling (RCIC) system;

(5) feedwater system;

(6) emergency core cooling (ECCS) systems; and

(7) miscellaneous systems.

Small leaks generally are detected by monitoring the air coolers, condensate flow, radiation levels, and drain sump fill-up and pump-out rates. Large leaks are also detected by changes in reactor water level and changes in flow rates in process lines.

1.2.2.5.4 Reactor Core Isolation Cooling System

The RCIC system provides makeup water to the reactor vessel when the vessel is isolated and is also part of the emergency core cooling network. 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 for events defined in Section 5.4.

One division contains the RCIC system which consists of a steam-driven turbine which drives a pump assembly and the turbine and pump accessories. The system also includes piping, valves, and instrumentation necessary to implement several flow paths. The RCIC steam supply line branches off one of the main steam lines (leaving the reactor pressure vessel) and goes to the RCIC turbine with drainage provision to the main condenser. The turbine exhausts to the suppression pool with vacuum breaking protection. Makeup water is supplied from the condensate storage tank (CST) or the suppression pool with preferred source being the CST. RCIC pump discharge lines include the main discharge line to the feedwater line, a test return line to the suppression pool, a minimum flow bypass line to the suppression pool and a cooling water supply line to auxiliary equipment.

Following a reactor scram, steam generation in the reactor core continues at a reduced rate due to the core fission product delay heat. The turbine condenser, and the feedwater system supplies the makeup water required to maintain reactor vessel inventory.

In the event the reactor vessel is isolated, and the feedwater supply is unavailable, relief valves are provided to automatically (or remote manually) maintain vessel pressure within desirable limits. The water level in the reactor vessel drops due to continued steam generation by decay heat. Upon reaching a predetermined low level, the RCIC system is initiated automatically. The turbine-driven pump supplies water from the suppression pool or from the CST to the reactor vessel. The turbine is driven with a portion of the decay heat steam from the reactor vessel, and exhausts to the suppression pool.

In the event there is a LOCA, the RCIC system in conjunction with the two HPCF systems, is designed to pump water into the vessel from approximately 150 psig to full operating pressure. These high pressure systems, combined with the

RHR low pressure flooders and ADS, make up the ECCS network which can accommodate any single failure and still shut down the reactor. (See Subsection 6.3.1.1 for a detailed description of ECCS redundancy and reliability.)

During RCIC operation, the wetwell suppression pool acts as the heat sink for steam generated by reactor decay heat. This results in a rise in pool water temperature. Heat exchangers in the residual heat removal (RHR) system are used to maintain pool water temperature within acceptable limits by cooling the pool water directly.

1.2.2.6 Reactor Servicing Equipment

1.2.2.6.1 Fuel Servicing Equipment

Fuel servicing equipment is summarized in Subsection 9.1.4.2.3.

1.2.2.6.2 Miscellaneous Servicing Equipment

The servicing aids equipment includes general handling fuel pool tools such as actuating poles with various end configurations. General area underwater lights and support brackets are provided to allow the lights to be positioned over the area being serviced independent of the platform. A general-purpose, plastic viewing aid is provided to float on the water surface to provide better visibility. A portable underwater closed circuit television camera may be lowered into the reactor vessel pool and/or the fuel storage pool to assist in the inspection and/or maintenance of these areas.

1.2.2.6.3 Reactor Pressure Vessel Servicing Equipment

The equipment associated with servicing the reactor pressure vessel is used when the reactor is shutdown and the reactor vessel head is being removed or installed. Tools used consist of strongbacks, racks, wrenches and protectors plus the RIP impeller accessories. Lifting tools are designed for a safety factor of 5 or better with respect to the ultimate strength of the material used.

1.2.2.6.4 RPV Internal Servicing Equipment

The majority of internal servicing equipment was designed to be attached to the refueling platform auxiliary hoist and used when the reactor is open. A variety of equipment such as grapples, guides, plugs, holders, caps, strongbacks and sampling stations are used for internal servicing. In addition to these are the RIP handling devices for repair and/or installation. Lifting tools are designed for a safety factor of 5 or better with respect to the ultimate strength of the material used.

1.2.2.6.5 Refueling Equipment

The fuel servicing equipment includes a 150-ton reactor building crane, fuel handling platform, fuel inspection stand, fuel preparation machine, jib hoist, and other related tools for reactor servicing.

The reactor building crane handles the spent fuel cask from the transport device to the spent fuel loading pit. The fuel handling platform transfers the fuel assemblies between this storage area, the reactor core, and the spent fuel shipping cask. New fuel bundles are handled by the reactor building crane. The bundles are stored in the new fuel vault on the reactor refueling floor and is transferred from the vault to the spent fuel pool with the reactor building crane auxiliary hook.

The handling of the reactor head, removable internals, and drywell head during refueling is accomplished using the reactor building crane.

1.2.2.6.5.1 Refueling Interlocks

A system of interlocks that restricts movement of refueling equipment and control rods when the reactor is in the refueling and startup modes is provided to prevent an inadvertent criticality during refueling operation. 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.

1.2.2.6.6 Fuel Storage Facility

New and spent fuel storage racks are designed to prevent inadvertent criticality and load buckling. Sufficient cooling and shielding are provided to prevent excessive pool heatup and 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 reaching 0.95 under dry or flooded conditions.

1.2.2.6.7 Under-Vessel Servicing Equipment

This equipment is used for the installation and removal work associated with the FMCRD, RIP, ICM and so on. A handling platform provides a working surface for equipment and personnel performing work in the under-vessel area. The polar platform is capable of rotating 360 degrees. All equipment is designed to minimize radiation exposure, contamination of surrounding equipment and the number of workers.

1.2.2.6.8 CRD Maintenance Facility

The CRD maintenance facility is designed and equipped to accommodate maintenance of the FMCRD, provide decontamination of the FMCRD component, perform the acceptance tests and provide storage. The facility uses manual and/or remote operation to minimize radiation exposure to the personnel and to minimize the contamination of surrounding equipment during operation. The layout of the facility is designed so as to maximize the efficiency of the personnel thereby minimizing the number of workers.

1.2.2.6.9 Internal Pump Maintenance Facility

The reactor internal pump (RIP) maintenance facility is located in the reactor building and is designed for performing maintenance work on the motor assembly and related parts. The facility is designed for one motor assembly including decontamination in assembled and disassembled states. The facility is equipped with all tools needed for inspection of motor parts and heat exchanger tube bundles. RIP handling tools are stored outside this area.

1.2.2.6.10 Fuel Cask Cleaning Facility

The fuel cask cleaning facility provides for empty casks to be checked for contamination and cleaned of road dirt, moved into the reactor building airlock, inspected for damage, opened and raised to the refueling floor cask pit. The closure head is stored in the adjacent cask wash down pit while the canal gates between the cask pit and spent fuel pool are removed and the spent fuel is transferred to fill the cask. The canal gates and closure head are replaced and the cask lifted to the washdown pit. The cask is decontaminated with high pressure water sprays, chemicals and hand scrubbing to the level required for offsite transport. Smear tests are performed to verify cleaning before the filled cask is lowered to the airlock, mounted on the transport vehicle and moved out of the reactor building.

1.2.2.6.11 Plant Start-up Test Equipment

Plant start-up test equipment is a combination of strain gages, accelerometers, temperature detectors, photo cells, pressure transducers and other associated instrumentation for conducting special startup and reactor internal vibration tests.

1.2.2.6.12 Inservice Inspection Equipment

Inservice inspection equipment are coordinated ultrasonic, eddy current and visual systems needed for incore housing, stub tube, feedwater nozzle, RPV inside and outside diameters (GERIS 2000), RPV internals and head studs, shroud head bolts, and piping (SMART 2000) inspections and examinations.

1.2.2.7 Reactor Auxiliary Systems

1.2.2.7.1 Reactor Water Cleanup System

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

1.2.2.7.2 Fuel Pool Cooling and Cleanup System

The fuel pool cleanup (FPC) system maintains acceptable levels of temperature and clarity and minimizes radioactivity levels of the water in the spent fuel pool, reactor well and dryer/separator pit on top of the containment. The FPC system also maintains the temperature and water level in the service pool and equipment pool. The system includes two heat exchangers, each capable of removing the decay heat generated from an average discharge of spent fuel, and two filter/demineralizers, each unit having the capacity to process the system flow or greater to maintain the desired purity level.

1.2.2.7.3 Suppression Pool Cleanup System

Suppression pool cleanup (SPCU) system provides a continuous purification of the suppression pool water. The system removes impurities by filtration, adsorption, and ion exchange processes. The system consists of a recirculation loop with a pump and isolation valves. Suppression pool water through the Fuel Pool Cooling and Cleanup (FPC) system filter demineralizers for treatment. Treated water may be diverted to refill the reactor well and the upper pool during refueling outage or provide makeup water to the fuel pool and reactor cooling water (RCW) surge tanks following a seismic event.

1.2.2.8 Control Panels

1.2.2.8.1 Main Control Room Panels

The main control room is summarized in Subsection 18.4.1.1.

1.2.2.8.2 Control Room Back Panels

The control room back panels are located in an area adjacent to the main control panels and convenient to the control room crew.

1.2.2.8.3 Radioactive Waste Control Panel

The radioactive waste control panel system provides the operator interface to the consolidated automatic and remote manual controlling of radioactive waste system mechanical, electrical, and chemical process components. It consists of one or more control panels including panel-mounted meters and displays, CRT displays, status indicating lights, mode and display selector switches, actuating mechanical and electrical components, controllers, and control logic elements and signal conditioning devices and processors. It does not include equipment or process sensors, local panels or equipment-mounted actuators or power controllers.

It is expected that most of the panels of this system will be located in the radioactive waste control room; panels performing the above functions which become located in the main control room shall also belong to this system.

1.2.2.8.4 Local Control Panels

The local control panels provide facilities for the installation and operation of electrical equipment and interconnecting wiring which supports no primary man-machine interface during normal plant operations. Included within the scope of the local control panels shall be the physical panel structure and the wiring associated with the components installed within the panels. The local control panels do not include the major electrical components, installed within the panels, which are instead defined and provided as part of the interfacing plant systems.

1.2.2.8.5 Instrument Racks

The instrument racks provide facilities for the installation and operation of locally mounted instrumentation. Included within the scope of the instrument racks shall be the physical structure upon which the instrumentation is mounted and the wiring associated with the instrument installations. The instrument racks do not include the locally mounted instrumentation which is instead defined and provided as part of the interfacing plant systems.

1.2.2.8.6 Multiplexing System

The multiplexing system provides redundant and distributed control and instrumentation data communications networks to support the monitoring and control of interfacing plant systems. The system provides all electrical devices and circuitry (such as multiplexing units, bus controllers, formatters and data busses) between sensors, display devices, controllers and actuators which are defined and provided by other plant systems. The multiplexing system also includes the associated data acquisition and communication software required to support its function of plant-wide data and control distribution.

1.2.2.8.7 Local Control Boxes

Local control boxes are uniquely identified to provide operational control of an individual piece of electrical equipment.

1.2.2.9 Nuclear Fuel

1.2.2.9.1 Nuclear Fuel

The nuclear fuel assembly contains fissionable material which produces thermal power while

maintaining structural integrity. The configuration of the fuel assembly consists of fuel rods, spacers, water rods, upper and lower tie plates, channel and channel fastener, all fabricated into a transportable, interchangeable assembly. The outer envelope of the fuel assembly is square with distinguishing features which provide support, identification, orientation and handling capabilities. The fuel design interface is described in Subsection 4.2.2.1.

1.2.2.9.2 Fuel Channel

The fuel channel encloses the fuel bundle and provides:

(a) A barrier between two parallel coolant flow paths, one for flow inside the bundle and the other for flow in the bypass region between channels.

(b) A bearing surface for the control rod.

(c) Rigidity for the fuel bundle.

The channel fastener attaches the channel to the fuel bundle, and along with the channel spacer buttons provides channel-to-channel spacing with resilient engagement.

1.2.2.10 Radioactive Waste System

1.2.2.10.1 Radwaste System

1.2.2.10.1.1 Liquid Radwaste Management Systems

The liquid radwaste management system collects, monitors, and treats liquid radioactive wastes for return to the primary system whenever practicable. The radwaste processing equipment is located in the radwaste building. Processed waste volumes discharged to the environs are expected to be small. Any discharge is such that concentrations and quantities of radioactive material and other contaminants are in accord with applicable local, state, and federal regulations.

All potentially radioactive liquid wastes are collected in sumps or drain tanks at various locations in the plant. These wastes are transferred to collection tanks in the radwaste facility.

Waste processing is done on a batch basis. Each batch is sampled as necessary in the collection tanks to determine concentrations of radioactivity and other contamination. Equipment drains and other low-conductivity wastes are treated by filtration and demineralization and are transferred to the condensate storage tank for reuse. Laundry drain wastes and other detergent wastes of low activity are treated by filtration, sampled and released via the liquid discharge pathway and demineralization and may be released from the plant on a batch basis. Protection against inadvertent release of liquid radioactive waste is provided by design redundancy, instrumentation for the detection and alarm of abnormal conditions, automatic isolation, and administrative controls.

Equipment is selected, arranged, and shielded to permit operation, inspection, and maintenance with minimum radiation exposure to personnel.

1.2.2.10.1.2 Solid Radwaste Management System

The solid radwaste management system provides for the safe handling, packaging, and short-term storage of radioactive solid and concentrated liquid wastes that are produced. Wet waste processed by this system is transferred to the solidification system where it is solidified in containers. Dry active waste is surveyed and disposed of whenever possible via the provisions of 10CFR20.302 (a). The remaining combustible waste is compacted. Incinerator ash is compacted waste are shipped in containers for off-site disposal.

1.2.2.11 Power Cycle Systems

1.2.2.11.1 Turbine Main Steam System

The main steam system delivers steam from the reactor to the turbine generator, the reheaters, and the steam jet air ejectors from warmup to full-load operation. The main steam system also provides steam for the steam seal system and the auxiliary steam system when other steam sources are not available.

1.2.2.11.2 Condensate, Feedwater and Condensate Air Extraction System

The condensate and feedwater system provides a dependable supply of high-quality feedwater to the reactor at the required flow, pressure, and temperature. The condensate pumps take the

deaerated condensate from the condenser hotwell and deliver it through the steam jet air ejector condenser, the gland steam condenser, the off-gas condenser, the condensate demineralizer, and through three parallel strings of four low pressure feedwater heaters to the reactor feed pumps' suction. The reactor feed pumps discharge through two stages of two parallel high pressure feedwater heaters to the reactor. The drains from the high pressure heaters are pumped backward to the suction of the reactors feed pumps.

1.2.2.11.2.1 Main Condenser Evacuation System

The main condenser evacuation system removes the noncondensible gases from the main condenser and discharges them to the gaseous radwaste system. This system consists of two 100% capacity, multiple-element, multi-stage steam jet air ejectors (SJAE) with intercondensers, for normal station operation, and mechanical vacuum pumps for use during startup.

1.2.2.11.3 Heater, Drain and Vent System

The heater, drain and vent system permits efficient and dependable operation of the heat cycle balance of plant equipment and, particularly, the condensate and feedwater regenerative heaters. All process equipment drains and vents are collected and routed to the appropriate points in the cycle and flows are controlled for equipment protection.

1.2.2.11.4 Condensate Purification System

Each unit is served by a 100% capacity condensate cleanup system, consisting of three hollow fiber filters followed by six deep-bed demineralizer vessels designed for parallel operation. One demineralizer vessel is a spare. The condensate cleanup system with instrumentation and automatic controls is designed to ensure a constant supply of high-quality water to the reactor.

1.2.2.11.5 Condensate Filter Facility

The condensate filter facility continuously removes suspended solids by processing the full flow condensate through three, one-third capacity, hollow fiber filters. A fast acting full flow bypass valve opens at high pressure differential across the filter to protect against sudden loss of condensate flow.

1.2.2.11.6 Condensate Demineralizer

The condensate demineralizers continuously process dissolved solids to reactor feedwater quality through five, one-fifth capacity, demineralizers and a sixth one-fifth capacity unit in manual standby. An emergency bypass line protects the equipment and a demineralizer resin handling and cleaning system is included.

1.2.2.11.7 Main Turbine

The main turbine is a 1800-rpm, tandom compound six flow, reheat steam turbine with 52 inch last stage blades. The turbine generator is equipped with an electro hydraulic control system and supervisory instruments to monitor performance. The gross electrical output of the turbine generator is approximately 1400 MW.

1.2.2.11.8 Turbine Control System

The turbine control system is summarized in Subsection 10.2.2.3.

1.2.2.11.9 Turbine Gland Steam System

The turbine gland steam system provides steam to the turbine shafts glands and the turbine valve stems. The turbine gland steam system prevents leakage of air into or radioactive steam out of the turbine shaft and turbine valves. The gland steam condenser collects air and steam mixture, condenses the steam, and discharges the air leakage to the atmosphere via the main vent by a motor-driven blower.

1.2.2.11.10 Turbine Lubricating Oil System

The turbine lubricating oil system shall supply oil to turbine-generator bearing lubrication lines and mainly consists of lube oil tank, oil pumps, oil coolers, and oil purifier equipment.

1.2.2.11.11 Moisture Separator Heater

The moisture separator reheater is summarized in Subsection 10.2.2.2, Subtopic; Moisture Separator Reheater.

1.2.2.11.12 Extraction System

Extraction steam from the high pressure turbine supplies the last stage of feedwater heating and

extraction steam from the low pressure turbines supplies the first four stages. An additional low pressure extraction drained directly to the condenser protects the last stage bucket from erosion induced by water droplets.

1.2.2.11.13 Turbine Bypass System

The turbine bypass system is summarized in Subsection 10.4.4.

1.2.2.11.14 Reactor Feedwater Pump Driver

Each reactor feedwater pump is driven by an adjustable speed synchronous motor.

1.2.2.11.15 Turbine Auxiliary Steam System

The turbine auxiliary steam system supplies steam to the steam jet air injectors for condenser deaeration and to the turbine gland seal system which prevents radioactive steam leakage out of the turbine casings and atmospheric air leakage into the casing at specific operating conditions.

The house boiler steam is a backup to the reactor generated steam during operation and would be used only when reactor steam is unavailable or too radioactive.

1.2.2.11.16 Generator

Each generator is a direct-driven, three phase, 60 Hz, 2700 V, 1800 rpm, conductor cooled, synchronous generator rated at approximately 1600 MVA, at 0.90 power factor, 75 psig hydrogen pressure, and .60 short circuit ratio.

1.2.2.11.17 Hydrogen Gas Cooling System

The hydrogen gas cooling system is summarized in Subsection 1.2.2.2, Subtopic; Bulk Hydrogen System.

1.2.2.11.18 Generator Cooling System

The generator cooling system includes the hydrogen cooled rotor portion of the hydrogen gas cooling system and the water cooled stator portion of the turbine building cooling water system.

1.2.2.11.19 Generator Sealing Oil System

The generator sealing oil system prevents hydrogen gas from leaking from the generator. The sealing oil is vacuum-treated to maintain the hydrogen gas purity.

1.2.2.11.20 Exciter

The generator exciter is a 565 V, 0.50 response ratio completely static General Electric Generrex-PPs System.

Main turbine-generator excitation power is provided by the output of the ac alternator-exciter. This output is rectified by the stationary silicon-diode rectifiers. The dc output of the rectifier banks then is applied to the main generator field through the generator collectors.

1.2.2.11.21 Main Condenser

The main condenser is a multipressure three-shell deaerating type condenser. During plant operation, steam expanding through the low pressure turbines is directed downward into the main condenser and is condensed. The main condenser also serves as a heat sink for the turbine bypass system, emergency and high level feedwater heater and drain tank dumps, and various other startup drains and relief valve discharges.

1.2.2.11.22 Off-Gas System

The off-gas system is summarized in Subsection 11.3.1.

1.2.2.11.23 Circulating Water System

The circulating water system provides a continuous supply of cooling water to the condenser to remove the heat rejected by the steam cycle and transfers it to the ultimate heat sink.

1.2.2.11.24 Condenser Cleanup Facility

The condenser cleanup facility removes slime and sludge to prevent vacuum decline of the condenser and to suppress corrosion on the inner surface of the condenser tubes.

1.2.2.12 Station Auxiliary Systems

1.2.2.12.1 Makeup Water System (Purified)

The makeup water system (purified) is summarized in Subsection 9.2.10.2.

1.2.2.12.2 Makeup Water System (Condensate)

The makeup water system maintains the required capacity and flow of the condensate for the RCIC and HPCF systems and maintains the required level in the condenser hotwell. The system also stores and transfers water during refueling and cask storage pool water during fuel shipping cask loading, receives and stores the process effluent from the liquid radwaste system, provides makeup to other plant systems where required, and provides condensate to the control rod drive (CRD) hydraulic system.

The system consists of a condensate storage tank, three condensate transfer pumps, and the necessary controls and instrumentation.

1.2.2.12.2.1 Condensate Storage Facilities and Distribution System

The condensate storage tank receives demineralized water from the condensate water makeup system and may also receive low conductivity water from the condensate return of the primary loop, from the radwaste disposal system and the condensate system in the turbine building.

1.2.2.12.3 Reactor Building Cooling Water System

The reactor building cooling water (RCW) system provides cooling water to certain designated equipment located in the reactor building. Adequate capacity and redundancy is provided in heat exchangers and pumps to ensure performance of the cooling system under all of offsite power, emergency power for the system is available from the onsite emergency diesel generators. The closed loop design provides a barrier between radioactive systems and the service water discharged to the environment. Heat is removed from the closed loop by the cooling water system. Radiation monitors are provided to detect contaminated leakage into the closed systems.

1.2.2.12.4 Turbine Building Cooling Water System

The turbine building cooling water system is summarized in Subsection 9.2.11.2.2.

1.2.2.12.5 HVAC Normal Cooling Water System

The HVAC normal cooling water system provides chilled water to the air supply cooling coils of the reactor building, to the heating/cooling coils in the drywell, and the control building electrical equipment room.

1.2.2.12.6 HVAC Emergency Cooling Water System

The HVAC emergency cooling water system provides chilled water to the cooling coils in the control building essential electrical equipment room, the main control room and the diesel generator electrical equipment areas. The safety-related chilled-water system is designed to meet the requirements of Criterion 19 of 10CFR50.

1.2.2.12.7 Oxygen Injection System

The oxygen injection system is summarized in Subsection 9.3.10.2.

1.2.2.12.8 Ultimate Heat Sink

The ultimate heat sink system is summarized in Subsection 9.2.5.3.

1.2.2.12.9 Reactor Service Water System

The reactor service water system is summarized in Subsection 9.2.15.2.

1.2.2.12.10 Turbine Service Water System

The turbine service water system is summarized ion Subsection 9.2.16.2.1

1.2.2.12.11 Station Service Air System

The station service air system provides a continuous supply of compressed air of suitable quality and pressure for general plant use. The service air compressor discharges into the air receivers and the air is then distributed throughout the plant.

1.2.2.12.12 Instrument Air System

The instrument air system is summarized in Subsection 9.3.6.2.

1.2.2.12.13 High Pressure Nitrogen Gas Supply System

Nitrogen gas is normally supplied by the Atmospheric Control System to meet the requirement on the main steam system safety relief valve automatic depressurization and relief

function accumulators, the main steam isolation valves, instruments and pneumatic valves using nitrogen in the reactor building. When this supply of pressurized nitrogen is not available, the HPIN automatically maintains nitrogen pressure to this equipment. The HPIN system consists of high pressure nitrogen storage bottles with piping, valves, instruments, controls and control panel.

1.2.2.12.14 Heating Steam and Condensate Water Return System

The heating steam and condensate water return system supplies heating steam from the House Boiler for general plant use and recovers the condensate return to the boiler feedwater tanks. The system consists of piping, valves, condensate recovery set and associated controls and instrumentation.

1.2.2.12.15 House Boiler

The house boiler system consists of the house boilers, reboilers, feedwater components, boiler water treatment and control devices. The house boiler system supplies turbine gland steam and heating steam including the concentrating tanks and devices of the high conductivity waste equipment.

1.2.2.12.16 Hot Water Heating System

The hot water heating system is a closed loop hot water supply to the various heating coils of the HVAC systems. The system includes two heat exchangers, a backup heat exchanger, surge and chemical addition tanks and associated equipment, controls and instrumentation.

1.2.2.12.17 Hydrogen Water Chemistry System

The hydrogen water chemistry system is summarized in Subsection 9.3.9.2.

1.2.2.12.18 Zinc Injection System

The zinc injection system is summarized in Subsection 9.3.11.1.

1.2.2.12.19 Breathing Air System

The breathing air system includes air compressors, dryers, purifiers and a distribution network. This network makes breathing air available in all plant areas where operations or maintenance must be performed and high radioactivity could occur in the ambient air. Special connections are provided to assure that this air is used only for breathing apparatus.

1.2.2.12.20 Sampling System (Includes PASS)

The process sampling system is furnished to provide 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 plant operation for laboratory or on-line analyses.

1.2.2.12.21 Freeze Protection System

The freeze protection system provides insulation, steam and electrical heating for all external tanks and piping that may freeze during winter weather.

1.2.2.12.22 Iron Injection System

The iron injection system consists of an electrolytic iron ion solution generator and means to inject the iron solution into the feedwater system in controlled amounts.

1.2.2.13 Station Electrical Systems

1.2.2.13.1 Electrical Power Distribution System

The unit Class 1E a-c power system supplies power to the unit Class 1E loads. The offsite power sources converge at the system. The system includes diesel generators that serve as standby power sources, independent of any onsite or offsite source. Therefore, the system has three sources. Furthermore, the system is divided into three divisions, each with its own independent distribution network, diesel generator, and redundant load group. A fourth division for the safety logic and control system bus receives power from the division 1 source.

1.2.2.13.2 Unit Auxiliary Transformer

The unit auxiliary a-c power system supplies power to unit loads that are non-safety related and uses the main generator as the normal power source with the reserve auxiliary transformers as a backup source. The unit auxiliary transformer steps down the a-c power to the 6900 V and 4160 V station bus voltage.

1.2.2.13.3 Isolated Phase Bus

The isolated phase buses duct system provides electrical interconnection from the main generator output terminals to the low voltage generator breaker and from the low voltage generator breaker to the low voltage terminals of the main transformer, and the unit auxiliary transformers. During the time the main generator is off line, the low voltage generator breaker is open and power is fed to the unit auxiliary transformers by back feeding from the main transformer. During startup the generator breaker is closed at about 7% power to provide power to the main and the unit auxiliary transformers for normal operation of the plant.

A package cooling unit is supplied with the isolated bus duct system.

1.2.2.13.4 Non-Segregated Phase Bus

The non-segregated phase bus provides the electrical interconnection between the unit auxiliary transformers and their associated 6.9kv metal-clad switchgear.

1.2.2.13.5 Metalclad Switchgear

The metal-clad switchgear distributes the 6.9kv power. Circuit breakers are drawout type, stored energy vacuum breakers. The switchgear interrupting rating shall be determined in accordance with requirements of ANSI C37.10.

1.2.2.13.6 Power Center

The power center is summarized in Subsection 8.3.1.1.2.1.

1.2.2.13.7 Motor Control Center

The motor control center is summarized in Subsection 8.3.1.1.2.2.

1.2.2.13.8 Raceway System

The raceway system is a plant wide network comprised of metallic cable trays, metallic conduits and supports. Raceways are classified for carrying medium voltage power cables, low voltage power cables, control cables and low level signal/instrumentation cables. Divisional cables are routed in separate cable raceways for each division. Fiber optic dataways are not restricted to raceway classifications, but would generally be run with control cables due to their common destinations.

1.2.2.13.9 Grounding Wire

Grounding wire is summarized in Subsection 8.3.1.1.6.2.

1.2.2.13.10 Electrical Wiring Penetration

Electrical wiring penetrations are summarized in Subsection 8.3.14.1.2(7).

1.2.2.13.11 Combustion Turbine Generator

The primary function of the combustion turbine generator)CTG) is to act as a standby on-site non-safety power source to feed permanent non-safety loads during loss of offsite power (LOOP) events.

The unit also provides an alternate AC power source in case of a station blackout event, as defined by Appendix B of Regulatory Guide 1.155.

1.2.2.13.12 Direct Current Power Supply

The plant has four independent Class 1E 125-volt dc power systems.

1.2.2.13.12.1 Unit Auxiliary DC Power System

The unit auxiliary DC power system supplies power to unit DC loads that are nonsafety-related. The system consists of two battery chargers, two batteries, two motor control centers, and two distribution panels.

1.2.2.13.12.2 Unit Class 1E DC Power System

The unit Class 1E DC power system supplies 125 VDC power to the unit Class 1E loads. Battery chargers are the primary power sources. The system, which includes storage batteries that serve as standby power sources, is divided into four divisions, each with its own independent distribution network, battery, charger, and redundant load group.

1.2.2.13.13 Emergency Diesel Generator System

The emergency diesel generator system is supplied by three diesel generators. Each Class 1E

division is supplied by a separate diesel generator. There are no provisions for transferring Class 1E loads between standby ac power supplies or supplying more than one engineered safety feature (ESF) from one diesel generator. This one-to-one relationship ensures that a failure of one diesel generator can effect only one ESF division. The diesel generators are housed in the reactor building which is a Seismic Category I structure, to comply with applicable NRC and IEEE design guides and criteria.

1.2.2.13.14 Vital AC Power Supply

1.2.2.13.14.1 Safety System Protection System Power System

Four divisions of the safety system logic and control (SSLC) power system provide an uninterruptible Class 1E source of 120-VAC single phase control power. The primary power source for the SSLC power system is the Class 1E AC power system. On loss of AC power, the appropriate divisional battery immediately assumes load without interruption. When AC power is restored, it resumes the load without interruption.

1.2.2.13.14.2 Uninterruptible Power System

The uninterruptible power system (UPS) supplies regulated 120 VAC single phase power to non Class 1E instrument and control loads which require an uninterruptible source of power. The power sources for the UPS are similar to those for the SSLC, but are non-Class 1E.

1.2.2.13.14.3 Reactor Protection System Alternate Current Power Supply

The reactor protection system alternate current power supply is summarized in Subsection 8.3.1.1.4.2.2.

1.2.2.13.15 Instrument and Control Power Supply

The instrument and control power supply provides 120 VAC single phase power to instrument and control loads which do not require an uninterruptible power source.

1.2.2.13.16 Communication System

The communication system is summarized in Subsection 9.5.2.

1.2.2.13.17 Lighting and Servicing Power Supply

The design basis for the lighting facilities is the standard for the Illuminating Engineering Society. Special attention is given to areas where proper lighting is imperative during normal and emergency operations. The system design precludes the use of mercury vapor fixtures in the containment and the fuel handling areas. The normal lighting systems are fed from the unit auxiliary transformers. Emergency power is supplied by engineered safety buses backed up by diesel generators. Normal operation and regular simulated offsite power loss tests verify system integrity.

1.2.2.14 Power Transmission Systems

1.2.2.14.1 Reserve Transformer

The reserve auxiliary transformer provides the alternate preferred feed for the safety-related buses M/C, C, D, & E. It also provides an alternate feed to 6.9kv bus M/C B1 which supplies the "B" train for plant investment protection loads. The "A" train plant investment protection load alternate feed is from the combustion turbine via 6.9kv bus M/C A1.

1.2.2.15 Containment and Environmental Control Systems

1.2.2.15.1 Primary Containment System

The primary containment system design for this plant incorporates the drywell/pressure suppression feature of previous BWR containment designs into a dry containment type structure. In fulfilling its design basis as a fission product barrier, the primary containment is a low leakage structure even at the increased pressures that could follow a main steamline rupture or a fluid system line break.

The main features of the containment design include:

(1) the drywell, a cylindrical steel lined reinforced concrete structure surrounding the reactor pressure vessel (RPV);

(2) a suppression pool filled with water which serves as a heat sink during normal operation and accident conditions;

(3) the air space above the suppression pool; and

(4) the reactor building which is structurally integrated with the concrete primary containment structure.

A secondary containment which surrounds the primary containment permits monitoring and treating all potential radioactive leakage from the primary containment. Treatment consists of HEPA and activated charcoal filtration.

1.2.2.15.2 Containment Internal Structures

The containment internal structures are summarized in Subsection 6.2.1.1.2.

1.2.2.15.3 Reactor Pressure Vessel Pedestal

The reactor pressure vessel pedestal is a prefabricated cylindrical steel structure filled with concrete which supports the RPV and is maintained below design temperature by cooling. The pedestal provides drywell connecting vents which lead to the horizontal vent pipes to the suppression pool.

1.2.2.15.4 Standby Gas Treatment System

The standby gas treatment (SGTS) system minimizes exfiltration of contaminated air from the secondary containment to the environment following an accident or abnormal condition which could result in abnormally high airborne radiation in the reactor building. Because the fuel storage area is also in the secondary containment it also can be exhausted to the SGTS.

All safety-related components of the SGTS are operable during loss of offsite power.

1.2.2.15.5 PCV Pressure and Leak Testing Facility

The PCV pressure and leak testing facility is summarized in Subsection 9.1.5.2.8 Special Servicing Room/Areas.

1.2.2.15.6 Atmospheric Control System

The atmospheric control system is summarized in Subsection 6.2.5.2.1.

1.2.2.15.7 Drywell Cooling System

The drywell cooling system is summarized in Subsection 9.4.9.2.

1.2.2.15.8 Flammability Control System

An atmospheric control system is designed to establish and maintain an inert atmosphere within the primary containment during all plant operating modes except during plant shutdown for refueling or maintenance. A recombiner system is provided to control the concentration of oxygen produced by radiolysis in the primary containment.

1.2.2.15.9 Suppression Pool Temperature Monitoring System

The suppression pool temperature monitoring system is summarized in Subsection 7.6.1.7.1.

1.2.2.16 Structures and Servicing Systems

1.2.2.16.1 Foundation Work

The analytical design and evaluation methods for the containment and reactor building walls, slabs and foundation mat and foundation soil are summarized in Subsection 3.8.1.4.1.1.

1.2.2.16.2 Turbine Pedestal

The description for the turbine pedestal is the same as that for foundation work in Subsection 3.8.1.4.1.1.

1.2.2.16.3 Cranes and Hoists

The crane and hoist are summarized in Subsection 9.1.4.2.2.1.

1.2.2.16.4 Elevator

The controlled elevators service the reactor building radiation controlled zones from the basemat to the refueling floor. Two additional clean elevators service all elevations of the clean zone.

1.2.2.16.5 Heating, Ventilating and Air Conditioning

The plant environmental control systems control temperature, pressure, humidity, and airborne contamination to ensure the integrity of plant equipment, provide acceptable working conditions

for plant personnel, limit offsite releases of airborne contaminants.

The following environmental systems are provided:

(1) the control room air conditioning system consisting of supply, recirculation/exhaust and makeup air cleanup units to ensure the habitability of the control room under normal and abnormal conditions of plant operation;

(2) the reactor building secondary containment HVAC system maintains a negative pressure in the secondary containment under normal and abnormal operating conditions thereby isolating the environs from potential leak sources. This system removes heat generated during normal plant operation, shutdown, and refueling periods;

(3) the drywell cooling system to remove heat from the drywell generated during normal plant operations including startup, reactor scrams, hot standby, shutdown, and refueling periods;

(4) the power block pressure control supply and exhaust system to distribute air so that a negative pressure is maintained in the emergency core cooling equipment rooms, thereby isolating the potential airborne contamination in these rooms;

(5) the electrical equipment supply and exhaust system to pressurize the electrical rooms allowing exfiltration of air to the battery rooms for exhaust to the outside atmosphere;

(6) the power block exhaust system to maintain the refueling floor at a negative pressure with respect to the outside atmosphere to prevent the potential release of airborne contamination,

(7) the diesel generator area air exhaust system to provide cooling during operation of the diesel generators. A tempered air supply system controls the thermal environment when the diesel generators are not operating; and

(8) coolers in the steam tunnel and ECCS rooms to remove heat generated during operation of the equipment in these rooms.

1.2.2.16.6 Fire Protection System

The fire protection system is designed to provide an adequate supply of water or chemicals to points throughout the plant where fire protection is required. Diversified fire-alarm and fire-suppression types are selected to suit the particular areas or hazards being protected. Chemical fire fighting systems are also provided as additions to or in lieu of the water fire fighting systems. Appropriate instrumentation and controls are provided for the proper operation of the fire detection, annunciation and fire fighting systems.

1.2.2.16.7 Floor Leakage Detection System

The drainage system is also used to detect abnormal leakage in safety related equipment rooms and the fuel transfer area.

1.2.2.16.8 Vacuum Sweep System

A portable, submersible-type, underwater vacuum cleaner is provided to assist in removing crud and miscellaneous particulate matter from the pool floors or reactor vessel. The pump and the filter unit are completely submersible for extended periods. The filter "package" is capable of being remotely changed, and the filters will fit into a standard shipping container for offsite burial.

1.2.2.16.9 Decontamination System

The decontamination system provides areas, equipment and services to support low radiation level decontamination activities. The services may include electrical power, service air, demineralized water, condensate water, radioactive and nonradioactive drains, HVAC and portable shielding.

1.2.2.16.10 Reactor Building

The reactor building includes the containment, drywell, and major portions of the nuclear steam supply system, steam tunnel, refueling area, diesel generators, essential power, non-essential power, emergency core cooling systems, HVAC and supporting systems;

1.2.2.16.11 Turbine Building

The turbine building houses all equipment associated with the main turbine generator. Other auxiliary equipment is also located in this building.

1.2.2.16.12 Control Building

The control building includes the control room, the computer facility, the cable tunnels, some of the plant essential switchgear, some of the essential power, reactor building water system and the essential HVAC system.

1.2.2.16.13 Radwaste Building

The radwaste building houses all equipment associated with the collection and processing of solid and liquid radioactive waste generated by the plant.

1.2.2.16.14 Service Building

The service building houses the personnel facilities, and portions of the non-essential HVAC.

1.2.2.17 Yard Structures and Equipment

1.2.2.17.1 Stack

The plant stack is located on the reactor building and rises to an elevation of 76 meters above grade level. The stack is a steel shell construction supported by an external steel tubular frame work. The stack vents the reactor building, turbine building, radwaste building, control and service buildings.

1.2.2.17.2 Oil Storage and Transfer System

The major components of this system are the fuel-oil storage tanks, pumps, and day tanks. Each diesel generator has its own individual supply components. Each storage tank is designed to supply the diesel needs during the post-LOCA period and each day tank has capacity for two hours of diesel generator operation. Each fuel oil pump is controlled automatically by day-tank level and feeds its day tank from the storage tank. Additional fuel oil pumps supply fuel to each diesel fuel manifold from the day tank.

1.2.2.17.3 Site Security

Site Security is summarized in Subsection 13.6.3.1.

1.6 MATERIAL INCORPORATED BY REFERENCE

Table 1.6-1 is a list of all GE topical reports and any other report or document which is incorporated in whole or in part by reference in the ABWR SSAR.

1.7 DRAWINGS

1.7.1 Piping and Instrumentation and Process Flow Drawings

Table 1.7-1 contains a list of system Piping and Instrumentation diagrams (P&ID) and process flow diagrams (PFD) provided in the ABWR SSAR. Figure 1.7-1 defines the sysmbols used on these drawings.

1.7.2 Instrument, Control and Electrical Drawings

Interlocking block diagrams (IBD), instrument engineering diagrams (IED) and single line diagrams (SLD) are listed in Table 1.7-2. Figure 1.7-2 defines the graphic symbols used in the IBDs.

1.7.3 ASME Standard Units Metric Conversion Factors

The ASME Standard units are applied with the numerical values converted to the metric system as listed in Table 1.7-3.

1.7.4 Metric Conversion to ASME Standard Units

Selected flow, pressure, temperature and length metric units are converted to ASME standard units as tabulated in Table 1.7-4.

1.7.5 Drawing Standards

Guidelines for identifying systems, facilities, equipment types and numbers and for drawing P&ID's and FPD's are treated in Table 1.7-5.

1.7.6 Interfaces

Applicants references the ABWR shall complete P&ID pipe schedules indicated as: Interface.

Table 1.7-1

PIPING AND INSTRUMENTATION AND PROCESS FLOW DIAGRAMS

SSAR Fig. No.	Page No.	Title	Type
4.6-8	4.6-24	CRD System	P&ID
4.6-9	4.6-26	CRD System	PFD
5.1-3	5.1-5	Nuclear Boiler System	P&ID
5.4-4	5.4-47	Reactor Recirculation System	P&ID
5.4-5	5.4-48	Reactor Recirculation System	PFD
5.4-8	5.4-51	Reactor Core Isolation Cooling System	P&ID
5.4-9	5.4-53	Reactor Core Isolation Cooling System	PFD
5.4-10	5.4-55	Residual Heat Removal System	P&ID
5.4-11	5.4-59	Residual Heat Removal System	PFD
5.4-12	5.4-61	Reactor Water Clean-Up System	P&ID
5.4-13	5.4-63	Reactor Water Clean-Up System	PFD
6.2-39	6.2-90	Atmospheric Control System	P&ID
6.2-40	6.2-92	Flamibility Control System	P&ID
6.3-1	6.3-25	High Pressure Core Flooder System	PFD
6.3-7	6.3-33	High Pressure Core Flooder System	P&ID
6.5-1	6.5-13	Standby Gas Treatment System	P&ID
6.7-1	6.7-4	High Pressure Nitrogen Gas Supply System	P&ID
9.1-1	9.1-23	Fuel Pool Cooling and Cleanup System	P&ID
9.1-2	9.1-25	Fuel Pool Cooling and Cleanup System	PFD
9.2-1	9.2-26	Reactor Building Cooling Water System	P&ID
9.2-1A	9.2-34a	Reactor Building Cooling Water System	PFD

Table 1.7-1

PIPING AND INSTRUMENTATION AND PROCESS FLOW DIAGRAMS (Continued)

SSAR Fig. No.	Page No.	Title	Type
9.2-2	9.2-35	HVAC Normal Cooling Water System	P&ID
9.2-3	9.2-37	HVAC Emergency Cooling Water System	P&ID
9.2-4	9.2-39	Makeup Water System (Condensate)	P&ID
9.2-5	9.2-40	Makeup Water System (Purified)	P&ID
9.3-1	9.3-16	Standby Liquid Control System	P&ID
9.3-6	9.3-21	Instrument Air System	P&ID
9.3-7	9.3-22	Service Air System	P&ID
9.4-8	9.4-8	Drywell Cooling System	P&ID
9.5-1	9.5-11	Suppression Pool Cleanup System	P&ID

Table 1.7-3

ASME STANDARD UNITS METRIC CONVERSION FACTORS (Continued)

	1 Gallons (US)	3785.4	Cubic Centimeters
	1 Gallons (US)	3.785412	E-03 Cubic Centimeters
(6)	Volume Per Unit Time		
	1 Cubic Feet/Minute	472	Cubic Centimeters/Sec
	1 Cubic Feet/Minute	1.699	Cubic Meters/Hour
	1 Cubic Feet/Minute	0.472	Liters/Second
	1 Cubic Feet/Second	0.02832	Cubic Meters/Minute
	1 Cubic Feet/Second	1.699	Cubic Meters/Minute
	1 Gallons/Minute (US)	6.30902	E-05 Cubic Meters/Sec.
	1 Gallons/Minute (US)	0.22715	Cubic Meters/Hour
	1 STD CFM (14.696 Psia, 60°F)	0.4474	Liters/Sec. (760MMHG, 0°C)
	1 STD CFM (14.696 Psia, 60°F)	1.608	Cubic Meters/Hr (760MMHG, 0°C)
(7)	Velocity		
	1 Feet/Sec.	30.48	Centimeters/Sec.
	1 Feet/Min.	0.508	Centimeters/Sec.
	1 Feet/Min.	0.00508	Meters/Sec.
	1 Feet/Min.	0.3048	Meters/Min.
	1 Inches/Sec.	2.45	Centimeters/Sec.
(8)	Area		
	1 Square Inch	6.4516	Square Centimeters
	1 Square Inch	6.4516	E-04 Square Meter
	1 Square Feet	929.03	Square Centimeter
	1 Square Feet	9.2903	E-02 Square Meter
(9)	Torque		
	1 Foot Pound	0.13825	Kilogram-Meters
(10)	Mass Per Unit Time		
	1 Pounds/Sec.	0.453592	Kilogram/Sec.
	1 Pounds/Min.	0.453592	Kilogram/Min.
	1 Pounds/Min.	27.2155	Kilogram/Hr.
(11)	Mass Per Unit Volume		
	1 Pounds/Cubic Inch	27680	Kilogram/Cubic Meter
	1 Pounds/Cubic Foot	16.01846	Kilograms/Cubic Meter
	1 Pounds/Cubic Inch	0.02768	Kilograms/Cubic Cm.
	1 Gallons/Minute	0.06309	Liters/Sec.

Figure 1.7-3

ASME STANDARD UNITS METRIC CONVERSION FACTORS (Continued)

(12) Dynamic Viscosity

1 Pound-Sec/Sq Ft	0.4882	Cram-Sec/Sq Ft

(13) Specific Heat/Heat Transfer

1 Btu/Pound-Deg F	1.00002	Kcal/Kg-Deg C
1 Btu/Hr-Sq Ft-Deg F	0.0004882	Kcal/Hr-Sq Cm-Deg C
1 Btu/Sec-Sq Ft-Deg F	0.0004882	Kcal/Sec-Sq Cm-Deg C
1 Btu/Hr-Sq Ft	2.712	Kcal/Hr-Sq Meter

(14) Temperature

32 Degrees Fahrenheit	0	Degress Celsius
100 Degress Celsius	212	Degrees Fahrenheit
1 Degree F Increment	0.56	Degree C Increment
1 Degree C Increment	1.8	Degree F Increment

Note:

Rounding of Calculated values per ASTM Std. 380-86.

Table 1.7-4

CONVERSION TABLES-METRIC TO ASME STANDARD UNITS

Flow-Volume Per Unit Time

M^3/HR	GAL/MIN	M^3/HR	GAL/MIN	M^3/HR	GAL/MIN	M^3/HR	GAL/MIN
1	4.4	10	44	100	440	1000	4402
2	8.8	20	88	200	881	2000	8805
3	13.2	30	132	300	1321	3000	13207
4	17.6	40	176	400	1761	4000	17610
5	22.0	50	220	500	2201	5000	22012
6	26.4	60	264	600	2641	6000	26414
7	30.8	70	308	700	3082	7000	30817
8	35.2	80	352	800	3522	8000	35219
9	39.6	90	396	900	3962	9000	39621

Temperature

°C	°F	°C	°F	°C	°F	°C	°F
0.1	32.18	1	33.8	10	50	100	212
0.2	32.36	2	35.6	20	68	200	392
0.3	32.54	3	37.4	30	86	300	572
0.4	32.72	4	39.2	40	104	400	752
0.5	32.90	5	41.0	50	122	500	932
0.6	33.08	6	42.8	60	140	600	1112
0.7	33.26	7	44.6	70	158	700	1292
0.8	33.44	8	46.4	80	176	800	1472
0.9	33.62	9	48.2	90	194	900	1652

Pressure

Kg/Km^2	PSI	Kg/Km^2	PSI	Kg/Km^2	PSI	Kg/Km^2	PSI
0.01	0.142	0.1	1.422	1	14.22	10	142.2
0.02	0.285	0.2	2.845	2	28.45	20	284.5
0.03	0.427	0.3	4.267	3	42.67	30	426.7
0.04	0.569	0.4	5.689	4	56.89	40	568.9
0.05	0.711	0.5	7.11	5	71.11	50	711.1
0.06	0.853	0.6	8.534	6	85.34	80	853.4
0.07	0.996	0.7	9.956	7	99.56	70	995.6
0.08	1.138	0.8	11.378	8	113.78	80	1137.8
0.09	1.280	0.9	12.800	9	128.00	90	1280.0

Table 1.7-4

CONVERSION TABLES-METRIC TO ASME STANDARD UNITS (Continued)

Length

M	inch	M	Inch	M	Inch	M	Inch
0.001	0.004	0.1	0.039	1	3.28	10	32.81
0.002	0.008	0.2	0.079	2	6.56	20	65.62
0.003	0.012	0.3	0.118	3	9.84	30	98.43
0.004	0.016	0.4	0.157	4	13.12	40	131.2
0.005	0.020	0.5	0.197	5	16.40	50	164.0
0.006	0.024	0.6	0.236	6	19.69	60	196.9
0.007	0.028	0.7	0.276	7	22.97	70	229.7
0.008	0.032	0.8	0.315	8	26.25	80	262.5
0.009	0.035	0.9	0.354	9	29.53	90	295.3

1.8 CONFORMANCE WITH STANDARD REVIEW PLAN AND APPLICABILITY OF CODES AND STANDARDS

1.8.1 Conformance With Standard Review Plan

The subsection provides the information required by 10 CFR 50.34(g) showing conformance with the Standard Review Plan (SRP). The summary of differences from the SRP section is presented by SRP section in Tables 1.8-1 through 1.8-18. See Subsection 1.8.4 for interface requirements.

1.8.2 Applicability of Codes and Standards

Standard Review Plans, Branch Technical Positions, Regulatory Guides and Industrial Codes and Standards which are applicable to the ABWR design are provided in Tables 1.8-19, 1.8-20 and 1.8-21. Applicable revisions are also shown. See Subsection 1.8.4 for interface requirements.

1.8.3 Applicability of Experience Information

Experience information is routinely made available and distributed to design personnel in the design process. Nuclear field experience is maintained in hard copy form in functional component and library files and in the GE world wide computer retrieval system.

Generic Letters and IE Bulletins, Information Notices and Circulars covering the decade including 1980 through the current issues (late 1991) were reviewed for applicability to the ABWR design. The review was enhanced by associating related experiences and tracing referenced occurrences. This was accomplished starting with the current issues of the Generic Letters and proceeding back into the decade. The Circulars, Bulletins and Notices were reviewed in that order. Interfacing experience was included in the review. The selection of ABWR information was based on the significance to future design and operation guidance. Included is a list of NUREGs related to the closing of current safety issues. Experience that resulted in applicable rules, codes and standards was not repeated. Table 1.8-22 list the experience information that has been included in the ABWR design or has ABWR Standard Plant/remainder of plant interface impact. See Subsection 1.8.4 for interface requirements.

A systematic procedure encompassing available resources was used to identify the applicability of experience information resulting in Table 1.8-22. Engineering management surveyed the indices of annual experience information to identify those very likely to be applicable to the ABWR. The remaining potentially applicable experience were reviewed individually. Experience information not deemed applicable to the ABWR design (issues pertaining to other reactor types, scram discharge volume, etc.) were not included in Table 1.8-22. The experience information categories applicable to the ABWR design in Table 1.8-22 include experience information accommodated by a design change, covered by review of USIs/GSIs or a issue that impacts the ABWR design but must be addressed by the party referencing the ABWR design on a specific application. This later category is included as an interface requirement.

Experience related to identified regulatory or industry developed resolutions were eliminated to avoid repetition except for selected experiences that have a nuisance potential for reoccurring. Lead system engineers classified the more complex experiences.

Reference to the new or novel design features used in the ABWR are provided below:

Feature	SSAR Section
Fine Motion Control Rod Drive	4.6
Internal Reactor Pumps	5.4.1
Multiplexing	7A.2
Digital/Solid State Control	7A.7
Overpressure Protection System	6.2.5.2.6, 6.2.5.3, 6.2.5.4
AC-Independent Water Addition System	5.4.7.1.1.10
Lower Drywell Flooder	9.5.1.2

1.8.4 Interfaces

The SRP sections to be addressed and the applicable regulatory guides and experience

information for the remainder of the plant are those of Table 1.8-19, 1.8-20 and 1.8-22 identified in the comment column, as "interface".

Table 1.8-22

EXPERIENCE INFORMATION APPLICABLE TO ABWR (Continued)

TYPE: GENERIC LETTERS

No.	Issue Date	Title	Comment
88-14	8/8/88	Instrument Air Supply System Problems Affecting Safety-Related Equipment PAST RELATED CORRESPONDENCE IE Notice 87-28, Supp. 1 NUREG-1275, Volume 2	
88-15	9/12/88	Electric Power Systems - Inadequate Control Over Design Process PAST RELATED CORRESPONDENCE IE Notice 88-45	
88-16	10/4/88	Removal of Cycle-Specific Parameter Limits from Technical Specifications	
88-18	10/20/88	Plant Record Storage on Optical Disks PAST RELATED CORRESPONDENCE NUREG-0800 REG. Guide 1.28, Rev. 3	Interface
88-20	11/23/88	Individual Plant Examination for Severe Accident Vulnerabilities-10CFR Para. 50.54(f)	
88-20	8/29/89	Generic 88-20 Supplement No. 1	
89-01	1/31/89	Implementation of programmatic Controls for Radiological Effluent Technical Specifications in the Administrative Controls Section of the Technical Specifications and the Relocation of Procedural Details of RETS to the Off-site Dose Calculation Manual or to the Process Control Program	Interface
89-02	3/21/89	Actions to Improve the Detection of Counterfeit and Fraudulently Marketed Products PAST RELATED CORRESPONDENCE EPRI-NP-5652 "Guideline for the Utilization of Commercial-Grade Items in Nuclear Safety-Related Applications". Bulletins 87-02 and Supplements 1 and 2, 88-05 and Supplements 1 and 2, 88-10 IE Notioces 87-66, 88-19, 88-35, 88-46 and Supplements 1 and 2, 88-48 and Supplement 1, 88-97	Interface
89-04	4/3/89	Guidance on Developing Acceptable Inservice Testing Program	Subsection 19B.2.2
89-06	4/12/89	Task Action Plan Item I.D.2-Safety Parameter Display System CFR 50.54(f)	

Table 1.8-22

EXPERIENCE INFORMATION APPLICABLE TO ABWR (Continued)

TYPE: GENERIC LETTERS

No.	Issue Date	Title	Comment
89-07	4/28/89	Power Reactor Safeguards Contingency Planning for Surface Vehicle Bombs	
89-07 Supp 1	4/21/89	Power Reactor Safeguards Contingency Planning for Surface Vehicle Bombs	
89-08	5/2/89	Erosion/Corrision-Induced Pipe Wall Thinning	
89-10	6/28/89	Safety Related Motor-Operated Valve Testing and Surveillance	Interface
89-11	6/30/89	Resolution of Generic Issue 101 "Boiling Water Reactor Water Level Redundancy	Subsection 19B.2.16
89-13	7/18/89	Service Water System Problems Affecting Safety Related Equipment	Interface
89-14	8/21/89	Line Item Improvements in Technical Specifications-Removal of the 3.25 Limit on Extending Surveillance Intervals	
89-15	8/21/89	Emergency Response Data System	Interface
89-18	9/6/89	Resolution of USI A-17, Systems Interactions	Subsection 19B.2.3
89-19	9/20/89	Request for Action Related to Resolution of Unresolved Safety Issue A-47 "Safety Impucation of Control Systems in LWR Nuclear Power Plants" Pursuant to 10 CFR 50.54(f)	Subsection 19B.2.5
89-22	10/19/89	Potential for Increased Roof Loads and Plant Area Flood Runoff Depth at Licensed Nuclear Power Plants Due To Recent Change in Probable Maximum Precipitation Criteria Developed By The National Weather Service	
90-02	01/01/90	Alternative Requirements for Fuel Assemblies in the Design Features Section of Technical Specifications.	
90-05	06/15/90	Guidance for Performing Temporary Non-Code Repair of ASME Code Class 1, 2 and 3 Piping	

Table 1.8-22

EXPERIENCE INFORMATION APPLICABLE TO ABWR (Continued)

TYPE: GENERIC LETTERS

No.	Issue Date	Title	Comment
90-09	12/11/90	Alternative Requirements for Snubber Visual Inspection Intervals and Corrective Actions	
91-03	03/06/91	Reporting of Safeguards Events	Interface
91-04	04/02/91	Changes in Technical Specification Surveillance Intervals to Accommodate a 24-Month Fuel Cycle	
91-05	04/04/91	Licensee Commercial Grade Procurement and Dedication Programs	
91-06	04/29/91	Resolution of Generic Issue A-30, "Adequacy of Safety-Related DC Power Supplies." Pursuant to 10CFR50.54(f)	Subsection 19B.2.24
91-10	07/08/91	Explosive Searches at Protected Area Portals	Interface
91-11	07/19/91	Resolution of Generic Issues 48, "LCOs for Class 1E Tie Breakers" Pursuant to 10CFR50.54(f)	Subsection 1B.2.24
91-14	09/23/91	Emergency Telecommunications	
91-16	10/03/91	Licensed Operators' and Other Nuclear Facility Personnel Fitness for Duty	Interface
91-17	10/17/91	Generic Safety Issue 29, "Bolting Degradation or Failure in Nuclear Power Plants"	

Table 1.8-22

EXPERIENCE INFORMATION APPLICABLE TO ABWR (Continued)

TYPE: IE BULLETINS

No.	Issue Date	Title	Comment
88-07, Supp 1	12/30/88	Power Oscillations in Boiling Water Reactors (BWRs)	
90-01	03/09/90	Loss of Fill-Oil in Transmitters Manufactured by Rosemount	
90-02	03/20/90	Loss of Thermal Margin Caused by Channel Box Bow	
91-01	10/18/91	Reporting Loss of Criticality Safety Controls	

Table 1.8-22

EXPERIENCE INFORMATION APPLICABLE TO ABWR (Continued)

TYPE: IE INFORMATION NOTICES

No.	Issue Date	Title	Comment
80-12	3/31/80	Instrumentation Failure Causes PORV Opening	
80-21	5/16/80	Anchorage & Support of Safety-Related Electrical Equipment	
80-22	5/28/80	Breakdowns in Contamination Control Programs	Interface
80-40	11/7/80	Excessive N_2 Supply Pressure	
80-42	11/24/80	Effect of Radiation on Hydraulic Snubber Fluid	
81-05	3/13/81	Degraded DC Systems at Palisades	Interface
81-07	3/16/81	Potential Problem with Water Soluble Purge Dam Materials used during Inert Gas Welding	Interface
81-10	3/25/81	Inadvertent Containment Spray	Interface
81-20	7/13/81	Test Failures of Electrical Penetrations	
81-21	7/21/81	Potential Loss of Direct Access to Ultimate Heat Sink	Interface
81-31	10/8/81	Failure of Safety Injection Valves	Interface
81-38	12/17/81	Potential Significant Equipment Failures Resulting from Contamination of Air-Operated Systems	Interface
82-03	3/22/82	Environmental Tests of Electrical Terminal Block	
82-10	3/3/82	Following Up Symptomatic Repairs	Interface
82-12	4/21/82	Surveillance of Hydraulic Snubbers	
82-22	7/9/82	Failures in Turbine Exhaust Lines	
82-23	7/16/82	Main Steam Isolation Valve Leakage	
82-25	7/20/82	Failures of Hiller Actuators Upon Gradual Loss of Air Pressure	
82-32	8/19/82	Contamination of Reactor Coolant System by Organics	Interface

Table 1.8-22

EXPERIENCE INFORMATION APPLICABLE TO ABWR (Continued)

TYPE: IE INFORMATION NOTICES

No.	Issue Date	Title	Comment
89-04	1/17/89	Potential Problems from the Use of Space Heaters	Interface
89-07	1/25/89	Failures of Small-Diameter Tubing in Control Air, Fuel Oil, and Lube Oil Systems Which Render Emergency Diesel Generators Inoperable	
89-08	1/26/89	Pump Damage Caused by Low-Flow Operation	
89-10	1/27/89	Undetected Installation Errors in Main Steam Line Pipe Tunnel Differential Temperature-Sensing Elements at Boiling Water Reactors	
89-11	2/2/89	Failure of DC Motor-Operated Valves to Develop Rated Torque Because of Improper Cabling Sizing	
89-14	2/16/89	Inadequate Dedication Process for Commercial Grade Components Which Could Lead to Common Mode Failure of a Safety System	
89-16	2/16/89	ExcessiveVoltage Drop in DC Systems PAST RELATED CORRESPONDENCE: Generic Letter 88-15	
89-17	2/22/89	Contamination and Degradation of Safety-Related Battery Cells	
89-20	2/24/89	Weld Failures in a Pump of Byron-Jackson Design	
89-21	2/27/89	Changes in Performance Characteristics of Molded-Case Circuit Breakers	
89-26	3/7/89	Instrument Air Supply to Safety-Related Equipment PAST RELATED CORRESPONDENCE: Generic Letter 88-14	
89-30	3/15/89	High Temperature Environments at Nuclear Power Plants	
89-36	4/4/89	Excessive Temperatures in Emergency Core Cooling System Piping Located Outside Containment	
89-37	4/4/89	Proposed Amendments to 40CFR Part 61, Air Emission Standards for Radionuclides	
89-39	4/5/89	List of Parties Excluded from Federal Procurement of Non-procurement Programs	Interface

Table 1.8-22

EXPERIENCE INFORMATION APPLICABLE TO ABWR (Continued)

TYPE: IE INFORMATION NOTICES

No.	Issue Date	Title	Comment
89-52	6/8/89	Potential Fire Damper Operational Problems	
89-61	8/30/89	Failure of Borg-Warner Gate Valves to Close Against Differential Pressure	
89-63	9/5/89	Possible Submergence of Electrical Circuits Located Above the Flood Level Because of Water Intrusion and Lack of Drainage	
89-64	9/7/89	Electrical Bus Bar Failures	Interface
89-66	9/11/89	Qualification Life of Solenoid Valves	
89-68	9/25/89	Evaluation of Instrument Setpoints During Modifications	Interface
89-69	9/29/89	Loss of Thermal Margin Caused by Channel Box Bow	Interface
89-70	10/11/89	Possible Indications of Misrepresented Vendor Products	Interface
89-71	10/19/89	Diversion of the Residual Heat Removal Pump Seal Cooling Water Flow During Recirculation Operation Following a Loss-of-Coolant Accident	
89-72	10/24/89	Failure of Licensed Senior Operators to Classify Emergency Events Properly	Interface
89-73	11/1/89	Potentail Overpressurization of Low Pressure Systems	Interface
89-76	11/21/89	Biofouling Agent: Zebra Mussel	Interface
89-77	11/21/89	Debris in Containment Emergency Sumps and Incorrect Screen Configurations	
89-79	12/1/89	Degraded Coatings and Corrosion of Steel Containment Vessels	
89-80	12/1/89	Potential for Water Hammer, Thermal Stratification, and Steam Binding in High-Pressure Coolant Injection Piping	
89-81	12/6/89	Inadequate Control of Temporary Modifications to Safety-Related Systems	Interface

Table 1.8-22

EXPERIENCE INFORMATION APPLICABLE TO ABWR (Continued)

TYPE: IE INFORMATION NOTICES

No.	Issue Date	Title	Comment
89-83	12/11/89	Sustained Degraded Voltage on the Off-Site Electrical Grid and Loss of Other Generating Stations as a Result of a Plant Trip	Interface
89-87	12/19/89	Disabling of Emergency Diesel Generators by Their Neutral Ground-Fault Protection Circuitry	
89-88	12/16/89	Recent NRC-Sponsored Testing of Motor-Operated Valves	
90-02	01/22/90	Potential Degradation of Secondary Containment	
90-05	01/29/90	Inter-System Discharge of Reactor Coolant	
90-07	01/30/90	New Information Regarding Insulation Material Performance and Debris Blockage of PWR Containment Sumps	
90-8	02/01/90	KR-85 Hazards From Decayed Fuel	
90-13	03/05/90	Importance of Review and Analysis of Safeguards Event Logs	Interface
90-20	03/22/90	Personnel Injuries Resulting Form Improper Operation of Radwaste Incinerators	Interface
90-21	03/22/90	Potential Failure of Motor-Operated Butterfly Valves to Operate Because Valve Seat Friction was Underestimated	Interface
90-22	03/23/90	Unanticipated Equipment Actuation Following Restoration of Power to Rosemount Transmitter Trip Units	Interface
90-25	04/16/90	Loss of Vital AC Power With Subsequent Reactor Coolant System Heatup	Interface
90-25 Supp. 1	03/11/90	Loss of Vital AC Power With Subsequent Reactor Coolant System Heatup	Interface
90-26	04/24/90	Inadequate Flow of Essential Service Water to Room Coolers and Heat Exchangers for Engineered Safety-Feature Systems	Interface

Table 1.8-22

EXPERIENCE INFORMATION APPLICABLE TO ABWR (Continued)

TYPE: IE INFORMATION NOTICES

No.	Issue Date	Title	Comment
90-30	05/01/90	Ultrasonic Inspection Techniques for Dissimilar Metal Welds	
90-33	05/09/90	Sources of Unexpected Occupational Radiation Exposure at Spent Fuel Storage Pools	Interface
90-39	06/01/90	Recent Problems with Service Water Systems	Interface
90-40	06/05/90	Results of NRC-Sponsored Testing of Motor-Operated Valves	Interface
90-42	06/19/90	Failure of Electrical Power Equipment Due to Solar Magnetic Disturbances	
90-47	[illegible]/27/90	Unplanned Radiation Exposures to Personnel Extremities Due to Improper Handling of Potentially Highly Radioactive Sources	Interface
90-56	08/08/90	Minimization of Methane Gas in Plant Systems and Radwaste Shipping Containers	Interface
90-53	08/16/90	Potential Failures of Auxiliary Steam Piping and the Possible Effects on the Operability of Vital Equipment	
90-54	08/28/90	Summary of Requalification Program Deficiencies	Interface
90-61	09/20/90	Potential for Residual Heat Removal Pump Damage Caused by Parallel Pump Interaction	
90-63	10/03/90	Management Attention To The Establishment And Maintenance of a Nuclear Criticality Safety Program	Interface
90-67	10/29/90	Potential Security Equipment Weaknesses	
90-68	10/30/90	Stress Corrision Cracking of Reactor Coolant Pump Bolts	
90-69	10/31/90	Adequacy of Emergency and Essential Lighting	
90-70	11/06/90	Pump Explosions Involving Ammonium Nitrate	
90-72	11/28/90	Testing of Parallel Disc Gate Valves in Europe	

Table 1.8-22

EXPERIENCE INFORMATION APPLICABLE TO ABWR (Continued)

TYPE: IE INFORMATION NOTICES

No.	Issue Date	Title	Comment
90-74	12/04/90	Information on Precursors to Severe Accidents	
90-78	12/18/90	Previously Unidentified Release Path From Boiling Water Reactor Control Rod Hydraulic Units	
90-81	12/24/90	Fitness For Duty	Interface
90-82	12/31/90	Requirements For Use of Nuclear Regulatory Commission-(NRC)-Approved Transport Packages For Shipment of Type A Quantities of Radioactive Material	Interface
91-04	01/28/91	Reactor Scram Following Control Rod Withdrawal Associated With Low Power Turbine Testing	
91-06	01/31/91	Lock-up of Emergency Diesel Generator And Load Sequencer Control Circuits Preventing Restart of Tripped Emergency Diesel Generator	
91-12	02/15/91	Potential Loss of Net Positive Suction Head (NPSH) of Standby Liquid Control System Pumps	
91-13	03/04/91	Inadequate Testing of Emergency Diesel Generators (EDGs)	
91-14	03/05/91	Recent Safety-Related Incidents at Large Irradiators	
91-17	03/11/91	Fire Safety of Temporary Installation of Services	Interface
91-19	03/12/91	High-Energy Piping Failures Caused by Wall Thinning	
91-22	03/19/91	Four Plant Outage Events Involving Loss of AC Power or Coolant Spills	
91-23	03/26/91	Accident Radiation Overexposures to Personnel Due to Industrial Radiography Accessory Equipment Malfunctions	Interface
91-29	04/15/91	Deficiencies Identified During Electrical Distribution System Functional Inspections	
91-33	05/31/91	Reactor Safety Information for States During Exercises and Emergencies	Interface

Table 1.8-22

EXPERIENCE INFORMATION APPLICABLE TO ABWR (Continued)

TYPE: IE INFORMATION NOTICES

No.	Issue Date	Title	Comment
91-34	06/03/91	Potential Problems in Identifying Causes of Emergency Diesel Generator Malfunctions	
91-37	06/10/91	Compressed Gas Cylinder Missile Hazards	Interface
91-38	06/13/91	Thermal Stratification in Feedwater System Piping	
91-40	06/19/91	Contamination of Nonradioactive System and Resulting Possibility for Unmonitored, Uncontrolled Release to the Environment	Interface
91-41	06/27/91	Potential Problems with the Use of Freeze Seals	Interface
91-42	07/27/91	Plant Outage Events Involving Poor Coordination Between Operations and Maintenance Personnel During Valve Testing and Manipulations	Interface
91-46	07/18/91	Degradation of Emergency Diesel Generator Fuel Oil Delivery Systems	Interface
91-47	08/06/91	Failure of Thermo-Lag Fire Barrier Material to Pass Fire Endurance Test	
91-49	08/15/91	Enforcement of Safety Requirements for Radiographers	Interface
91-50	08/20/91	A Review of Water Hammer Events After 1985	
91-51	08/20/91	Inadequate Fuse Control Programs	Interface
91-57	09/19/91	Operational Experience on Bus Transfers	
91-58	09/20/91	Dependency of Offset Disc Butterfly Valve's Operation of Orientation With Respect to Flow	
91-59	09/23/91	Problems With Access Authorization Programs	Interface
91-60	11/01/91	Reissuance of Information Notice 91-60: False Alarms of Alarm Ratemeters Because of Radio Frequency Interference	Interface
91-61	09/30/91	Preliminary Results of Validation Testing of Motor-Operated Valve Diagnostic Equipment	

Table 1.8-22

EXPERIENCE INFORMATION APPLICABLE TO ABWR (Continued)

TYPE: IE INFORMATION NOTICES

No.	Issue Date	Title	Comment
91-63	10/03/91	Natural Gas Hazards at Fort St. Vrain Nuclear Generating Station	Interface
91-64	10/03/91	Site Area Emergency Resulting From a Loss of Non-Class 1E Uninterruptable Power Supplies	
91-65	10/17/91	Emergency Access to Low-Level Radioactive Waste Disposal Facilities	Interface
91-66	10/18/91	(1) Erroneous Date in "Nuclear Safety Guide, TID-7016, Revision 2," (NUREG/CR-0095, ORNL/NUREG/CSD-6 (1978) And (2) Thermal Scattering Data Limitation in the Cross-Section Sets Provided With the Keno and Scale Codes	
91-68	10/28/91	Careful Planning Significantly Reduces the Potential Adverse Impacts of Loss of Offsite Power Events During Shutdown	Interface
91-72	11/19/91	Issuance of a Revision to the EPA Manual of Protective Action Guides and Protective Actions for Nuclear Incidents	

Table 1.8-22

EXPERIENCE INFORMATION APPLICABLE TO ABWR (Continued)

TYPE: IE CIRCULARS

No.	Issue Date	Title	Comment
80-03	3/6/80	Protection from Toxic Gas Hazards	Interface
80-05	4/1/80	Emergency D/G Lube Oil	Interface
80-08	4/18/80	RPS Response Time	
80-09	4/28/80	Problems with Plant Internal Communications Systems	Interface
80-10	4/29/80	Failure to Maintain Environmental Qualification of Equipment	Interface
80-11	5/13/80	Emergency Diesel Generator Lube Oil Cooler Failures	Interface
80-14	6/24/80	Radioactive Contamination of Demin Water System	Interface
80-18	8/22/80	10 CFR 50.59 Safety Evaluation for Changes to Radioactive Waste Treatment Systems	Interface
81-03	3/2/81	Inoperable Seismic Monitoring Instrument	Interface
81-05	3/31/81	Self-Aligning Rod End Bushing for Pipe Supports	Interface
81-07	5/14/81	Control of Radioactivity Contaminated Material	Interface
81-08	5/29/81	Foundation Materials	Interface
81-09	7/10/81	Containment Effluent Water	
81-11	7/24/81	Inadequate Decay Heat Removal	Interface
81-13	9/25/81	Torque Switch Electrical Bypass Circuit	Interface
81-14	11/5/81	Main Steam Isolation Valve Failures to Close	Interface

Table 1.8-22

EXPERIENCE INFORMATION APPLICABLE TO ABWR (Continued)

TYPE: NUREG

No.	Issue Date	Title	Comment
0313 Rev. 2	6/88	Technical Report on Material Selection and Processing Guidelines for BWR Coolant Pressure Boundary Piping	
0371	10/78	Task Action Plans for Generic Activities Category A	
0471	6/78	Generic Task Problem Description: Category B, C & D Tasks	
0578	9/80	Performance Testing of BWR and PWR Relief and Safety Valves.	
0588	12/79	Interim Staff Position On Environmental Qualification of Safety-Related Electrical Equipment	
0619	4/80	BWR Feedwater Nozzle and Control Rod Drive Return Line Nozzle Cracking	
0626	1/80	Generic Evaluation of Feedwater Transients and Small Break LOCA in GE-Designed Operating Plants and Near-Term Operating License Applications	
0660	5/80	NRC Action Plan Developed as a Result of the TMI-2 Accident	
0661 Supp. 1	8/82	Safety Evaluation Report-Mark I Containment Long-Term Program-Resolution of Generic Technical Activity A-7	
0710 Rev. 1	6/81	Licensing Requirements for Pending Applications for Construction Permits and Manufacturing License.	
0737 Supp.1	12/82	Clarification of TMI Action PLan Requirements	
0744 Rev. 1	10/82	Resolution of the Task A-11 Reactor Vessel Materials Toughness Safety Issue	
0808	8/81	Mark II Containment Program Load Evaluation and Acceptance Criteria	
0813	9/81	Draft Environmental Statement Related to the Operation of Calloway Plant, Unit No.1	
0977	3/83	NRC Fact-Finding Task Force Report on the ATWS Events at the Salem Nuclear Generating Station, Unit 1, on February 22 and 25, 1983	

Table 1.8-22

EXPERIENCE INFORMATION APPLICABLE TO ABWR (Continued)

TYPE: NUREG

No.	Issue Date	Title	Comment
1150	6/89	Severe Accident Risks: An assessment for Five U.S. Nuclear Power Plants, Vol. 1 & 2.	
1161	5/80	Recommended Revisions to USNRC-Seismic Design Criteria	Subsection 19B.2.27
1174	5/89	Evaluation of Systems Interactions in Nuclear Power Plants	Subsection 19B.2.3
1212	6/86	Status of Maintenance in the US Nuclear Power Industry 1985 Vol. 1, 2	
1216	8/86	Safety Evaluation PP2 Related to Operability and Reliability of Emergency Diesel Generators	
1217	4/88	Evaluation of Safety Implications of Control Systems in LWR Nuclear Power Plants-Technical Findings Related to USI A-47	Subsection 19B.2.5
1218	4/88	Regulatory Analysis for Resolution of USI A-47	Subsection 19B.2.5
1229	8/89	Regulatory Analysis for Resolution of USI A-17	Subsection 19B 2.3 & 19B2.27
1233	9/89	Regulatory Analysis for USI A-40	Subsection 19B.2.27
1273	4/88	Containment Integrity Check-Technical Finds Regulatory Analysis	
1289	11/88	Regulatory and Backfit Analysis: Unresolved Safety Issue A-45, Shutdown Decay Heat Removal Requirements	Subsection 19B.2.29
1296	2/88	Peer Review of High Level Nuclear Waste	
1341	5/89	Regulatory Analysis for Resolution of Generic Issue 115, Enhancement	
1353	4/89	Regulatory Analysis for the Resolution of Generic Issue 82, "Beyond Design Basis Accidents in Spent Fuel Pools"	Subsection 19B.2.14
1370	9/89	Resolution of USI A-48	Subsection 19B.2.6

Table 1.8-22

EXPERIENCE INFORMATION APPLICABLE TO ABWR (Continued)

TYPE: NUREG

No.	Issue Date	Title	Comment
1275	2/91	Volume 6, Operating Experience Feedback Report - Solenoid Operated Valve Problems	
1339	6/90	Resolution of Generic Safety Issue 29: Bolting Degradetion or Failure in Nuclear Power Plants	Subsection 19B.2.12
CR-3922	1/85	Survey and Evaluation of System Interaction Events and Sources Vol. 1, 2	Subsection 19B.2.3
CR-4261	3/86	Assessment of Systems Interactions in Nuclear Power Plants	Subsection 19B.2.3
CR-4262	5/85	Effects of Control System Failures on Transients, Accidents at a GE BWR Vol. 1 and 2	
CR-4387	12/85	Effects of Control System Failures on Transient and Accidents and Core-Melt Frequencies at a GE BWR	
CR-4470	5/86	Survey and Evaluation of Vital Instrumentation and Control Power Supply Events	
CR-5055	5/88	Atmospheric Diffusion for Control Room Habitability Assessment	Subsection 19B.2.31
CR-5088	1/89	Fire Risk Scoping Study: Investigation of Nuclear Power Plant Fire Risk, Including Previously Unaddressed Issues.	
CR-5112	3/89	Evolution of Boiling Water Reactor Water - Level Sensing Line Break and Single Failure	Subsection 19B.2.16
CR-5230	4/89	Shutdown Decay Heat Removal Analysis: Plant Case Studies and Special Issues	Subsection 19B.2.29
CR-5347	6/89	Recommendations for Resolution of Public Comments on USI A-40	Subsection 19B.2.27
CR-5458	12/89	Value-Impact Assess for Candidate Operating Procedure Upgrade Program	
CR-4674	84/89	Precusors to Potential Severe Core Damage Accidents: Series	

Table 1.9-1

SUMMARY OF ABWR STANDARD PLANT INTERFACES

WITH REMAINDER OF PLANT (Continued)

ITEM NO.	SUBJECT	INTERFACE TYPE	SUBSECTION
5.2	Conversion of Indicators	Procedural	5.2.6.2
5.3	Fracture Toughness Data	Confirmatory	5.3.4.1
5.4	Materials and Surveillance Capsule	Confirmatory	5.3.4.2
6.1	Protection Coatings and Organic Materials	Confirmatory	6.1.3.1
6.2	External Temperature	Confirmatory	6.4.7.1
6.3	Meterology (X/Qs)	Confirmatory	6.4.7.2
6.4	Toxic Gases	Confirmatory	6.4.7.3
7.1	Effects of Sation Blackout on HVAC	Confirmatory	7.8.1
7.2	Deleted		
7.3	Localized High Heat Spots in Semiconductor Material for Computing Devices	Confirmatory	7.8.3
8.1	Stability of offsite power system	Confirmatory	8.1.4.1
8.2	Diesel Generator Reliability	Procedural	8.1.4.2
8.3	Class IE Feeder Circuits	Design	8.2.3.1
8.4	Non-class IE Feeders	Design	8.2.3.2
8.5	Specific ABWR Standard Plant/remainder of plant power sysytem interfaces	Design	8.2.3.3
8.6	Interupting Capability of Electrical Distribution Equipment	Confirmatory	8.3.4.1
8.7	Diesel Generator Design Details	Confirmatory	8.3.4.2
8.8	Certified Proof Tests on Cable Samples	Confirmatory	8.3.4.3
8.9	Electrical Penetration Assemblies	Confirmatory	8.3.4.4
8.10	Analysis Testing for Spatial Seperation per IEEE 304	Confirmatory	8.3.4.5

Table 1.9-1

SUMMARY OF ABWR STANDARD PLANT INTERFACES

WITH REMAINDER OF PLANT (Continued)

ITEM NO.	SUBJECT	INTERFACE TYPE	SUBSECTION
8.11	DC Voltage Analysis	Confirmatory	8.3.4.6
8.12	Seismic Qualification of Eyewash Equipment	Confirmatory	8.3.4.7
8.13	Diesel Generator Load Table Changes	Confirmatory	8.3.4.8
8.14	Offsite Power Supply Arrangements	Procedural	8.3.4.9
8.15	Diesel Generator Qualification Tests	Confirmatory	8.3.4.10
8.16	Defective Refurbished Circuit Breakers	Confirmatory	8.3.4.11
8.17	Minimum Starting Voltages for Class 1E Motors	Confirmatory	8.3.4.12
9.1	New Fuel Storage Racks Criticality Analysis	Confirmatory	9.1.6.1
9.2	New Fuel Storage Racks Dynamic and Impact Analysis	Confirmatory	9.1.6.2
9.3	Spent Fuel Storage Racks Criticality Analysis	Confirmatory	9.1.6.3
9.4	Spent Fuel Storage Rack Load Drop Analysis	Confirmatory	9.1.6.4
9.5	Ultimate heat sink capability	Design	9.2.17.1
9.6	Makeup water system capability	Design	9.2.17.2
9.7	Potable and Sanitary Water System	Design	9.2.17.3
9.8	Radioactive Drain Transfer System Collection Piping	Design	9.3.12.1
9.9	Contamination of DG Combustion Air Intake	Confirmatory	9.5.13.1
9.10	Use of Communication System in Emergencies	Procedural	9.5.13.2
9.11	Maintenance and Testing Procedures for Communication Equipment	Procedural	9.5.13.3
9.12	Fire Hazard Analysis Database	Confirmatory	9A.6.3
10.1	Low Pressure Turbine Disk Fracture Toughness	Confirmatory	10.2.5.1

SECTION 1A.3

CONTENTS

ATTACHMENT A

CONTENTS

ATTACHMENT A

CONTENTS (Continued)

TABLES

TABLE 2.1-1

LIMITS IMPOSED ON SRP SECTION II ACCEPTANCE CRITERIA BY ABWR DESIGN (Continued)

SRP SECTION	SUBJECT	LIMITS
GEOLOGY, SEISMOLOGY AND GEOTECHNICAL ENGINEERING		
2.5.1	Basic Geology and Seismic Information	None.
2.5.2	Vibratory Ground Motion	Per Table 2.0-1.*
2.5.3	Surface Faulting	None.
2.5.4	Stability of Subsurface Materials and Foundations	Per Table 2.0-1.
2.5.5	Stability of Slopes	None.

* Acceptance Criteria II 2.5.2.7 of the SRP Section 2.5.2 specifies the minimum value of the OBE to one-half of the SSE. The ABWR Standard Plant has adopted an OBE equal to one-third of the SSE.

The relationship between the magnitude of the OBE and the SSE established in paragraph V of 10 CFR 100 Appendix A is inconsistent with their definitions. The OBE is defined in 10 CFR 100 as the earthquake which could reasonable be expected to affect the plant site during the operating life of the plant, for which those features necessary for continued safe operation of the plant are designed to remain functional. The SSE is based upon the maximum earthquake potential which produces the maximum vibratory ground motion for which certain structures, systems, and components are designed to remain functional. In coupling the events, as implied by the current regulatory requirement, the intent of the OBE as a reasonably likely event is lost. The use of a 100 year recurrence level of the OBE is appropriate compared to the plant life and is also appropriately conservative relative to the Uniform Building Code requirements for non-safety related structures.

Decoupling the OBE from the SSE has been an issue in the technical community for quite some time. Both industry and regulatory have recognized the inconsistency in the definitions and some of the undesirable results such as greatly stiffened structures and systems to meet the more restrictive OBE stress levels.

Generic Issue 119.3, *Decoupling of the OBE from the SSE*, was introduced into the regulatory process by recommendation A-3 of the Piping Review Committee. In the historical background of this generic issue, it is noted that in developing the current regulations, it was assumed that the OBE would serve as a separate check of those systems where continued operation was desired at a lower level of ground motion. However, in practice, the assumed load factors, damping, stress levels, and service limits have caused the OBE, rather than the SSE, to control the design for many systems including concrete and steel structures and nuclear piping. In addition, seismic design for OBE accounts for certain safety-related factors such as fatigue and seismic anchor

TABLE 2.1-1

LIMITS IMPOSED ON SRP SECTION II ACCEPTANCE CRITERIA
BY ABWR DESIGN (Continued)

SRP SECTION	SUBJECT	LIMITS
GEOLOGY, SEISMOLOGY AND GEOTECHNICAL ENGINEERING		
movement that are not considered in the design for the SSE. As a further consequence, structures and systems have been greatly stiffened to meet restrictive OBE allowable stress levels. This stiffening is detrimental to actual plant conditions.		
Since the ABWR is a plant of the future and there is sufficient evidence that more flexible designs can exhibit reliability and safety levels equal to or greater than original stiffer designs (such as piping designs studied by (NUREG/CR-4263), this lower magnitude OBE is warranted.		

Table 2A-1

SPATIAL SUBGROUP
(NUREG/CR-2326, pg 2-11[1])

SPATIAL		20	NO				
.25	0.75	1.25	1.75	2.25	2.75	3.75	4.25
5.75	6.25	7.75	8.25	9.75	10.25	13.75	16.25
18.75	21.25	23.75	26.25				

[1]This reference specifies the location for the following CRAC input parameters, their definitions and formatting instructions.

Table 2A-2

SITE SUBGROUP
(NUREG/CR-2326, pg 2-13)

SITE	1	
GENERIC SITE		50001
29 06		

Table 2A-3

ECONOMIC SUBGROUP
(NUREG/CR-2326, pg 2-22)

ECONOMIC		54		NO			
499.0	3349.0		0.2	31527.0	4344.0	135.0	685.0
MAINE	5	9	0.077	0.182	250.0	485.0	
N.H.	5	9	0.097	0.444	150.0	802.0	
VT	5	9	0.283	0.791	177.0	657.0	
MASS	5	9	0.123	0.283	372.0	1366.0	
R.I.	5	9	0.081	0.220	476.0	2133.0	
CONN	5	9	0.140	0.313	500.0	2158.0	
N.Y.	5	9	0.315	0.579	188.0	642.0	
N.J.	5	9	0.197	0.162	376.0	2222.0	
PA	5	9	0.307	0.413	239.0	669.0	
OHIO	5	9	0.618	0.153	183.0	1516.0	
IND	5	9	0.728	0.067	206.0	1498.0	
ILL	5	9	0.795	0.041	213.0	1786.0	
MICH	5	9	0.285	0.238	197.0	955.0	
WIS	5	9	0.520	0.598	194.0	807.0	
MINN	5	9	0.563	0.185	160.0	854.0	
IOWA	5	9	0.944	0.050	242.0	1458.0	
MO	5	9	0.724	0.079	111.0	674.0	
N.D.	5	9	0.922	0.047	45.0	306.0	
S.D.	5	9	0.922	0.074	46.0	257.0	
NEBR	5	9	0.967	0.027	99.0	470.0	
KANS	5	9	0.915	0.034	92.0	437.0	
DEL	4	10	0.471	0.046	508.0	1725.0	

Table 2A-3 (Cont'd)

ECONOMIC SUBGROUP
(NUREG/CR-2326, pg 2-22)

MD	4	10	0.414	0.227	273.0	1799.0
VA	4	10	0.371	0.171	126.0	864.0
W.VA	4	10	0.270	0.203	44.0	472.0
N.C.	4	10	0.368	0.056	261.0	819.0
S.C.	4	10	0.327	0.063	148.0	635.0
GA	4	10	0.417	0.058	164.0	609.0
FLA	4	10	0.368	0.077	233.0	930.0
KY	4	10	0.557	0.117	141.0	792.0
TENN	4	10	0.507	0.140	118.0	669.0
ALA	4	10	0.400	0.041	144.0	515.0
MISS	4	10	0.475	0.047	135.0	520.0
ARK	4	10	0.494	0.030	158.0	691.0
LA	4	10	0.332	0.087	137.0	763.0
OKLA	4	10	0.782	0.051	68.0	442.0
TEXAS	4	10	0.811	0.053	54.0	354.0
MONTANA	5	9	0.658	0.026	20.0	186.0
IDAHO	5	9	0.894	0.114	93.0	485.0
WYOMING	5	9	0.560	0.024	15.0	119.0
COLORADO	4	10	0.570	0.039	69.0	332.0
N.MEXICO	4	10	0.600	0.056	21.0	100.0
ARIZONA	4	10	0.556	0.069	36.0	134.0
UTAH	4	10	0.236	0.215	36.0	265.0
NEVADA	4	10	0.127	0.117	19.0	104.0
WASH	5	9	0.369	0.138	132.0	586.0
OREGON	5	9	0.300	0.093	68.0	330.0
CALIF	4	10	0.318	0.119	316.0	936.0
NOVA SCO	5	9	0.0	0.0	0.0	0.0
QUEBEC	5	9	0.0	0.0	0.0	0.0
ONTARIO	5	9	0.0	0.0	0.0	0.0
BAJA CAL	5	9	0.0	0.0	0.0	0.0
SONORA	5	9	0.0	0.0	0.0	0.0
CHIHUAHU	5	9	0.0	0.0	0.0	0.0

Table 2A-4

POPULATION SUBGROUP
(NUREG/CR-2326, pg 2-26)

To be supplied by utility for specified spatial mesh above.

Table 2A-5

TOPOGRAPHY SUBGROUP
(NUREG/CR-2326, pg 2-33)

To be supplied by utility for specified spatial mesh above.

Table 2A-6

ISOTOPIC SUBGROUP
(NUREG/CR-2326, pg 2-37)

ISOTOPE		54	NO	EVALUATED FOR 3926MWT CINDR SOURCE		
CO-58	7	3.730E+05	7.130E+01	1.000E-02	1.000E-04	0.5 OF GESSAR
CO-60	7	2.247E+03	1.921E+03	1.000E-02	1.000E-04	0.5 OF GESSAR
KR-85	1	1.184E+06	3.919E+03	0.	0.	
KR-85M	1	2.644E+07	1.867E-01	0.	0.	
KR-87	1	5.070E+07	5.278E-02	0.	0.	
KR-88	1	7.185E+07	1.167E-01	0.	0.	
RB-86	4	1.844E+05	1.865E+01	1.000E-02	1.000E-04	
SR-89	6	9.700E+07	5.200E+01	1.000E-02	1.000E-04	
SR-90	6	1.014E+07	1.026E+04	1.000E-02	1.000E-04	
SR-91	6	1.242E+08	3.950E-01	1.000E-02	1.000E-04	
Y-90	8 SR-90	1.094E+07	2.670E+00	1.000E-02	1.000E-04	
Y-91	8 SR-91	1.263E+08	5.880E+01	1.000E-02	1.000E-04	
ZR-95	8	1.735E+08	6.550E+01	1.000E-02	1.000E-04	
ZR-97	8	1.781E+08	7.000E-01	1.000E-02	1.000E-04	
NB-95	8 ZR-95	1.734E+08	3.510E+01	1.000E-02	1.000E-04	
MO-99	7	1.966E+08	2.751E+00	1.000E-02	1.000E-04	
TC-99M	7 MO-99	1.696E+08	2.508E-01	1.000E-02	1.000E-04	
RU-103	7	1.664E+08	3.959E+01	1.000E-02	1.000E-04	
RU-105	7	1.174E+08	1.850E-01	1.000E-02	1.000E-04	
RU-106	7	5.909E+07	3.690E+02	1.000E-02	1.000E-04	
RH-105	7 RU-105	9.907E+07	1.479E+00	1.000E-02	1.000E-04	
SB-127	5	8.969E+06	3.800E+00	1.000E-02	1.000E-04	
SB-129	5	3.172E+07	1.808E-01	1.000E-02	1.000E-04	
TE-127	5 SB-127	8.853E+06	3.896E-01	1.000E-02	1.000E-04	
TE-127M	5	1.339E+06	1.090E+02	1.000E-02	1.000E-04	
TE-129	5 SB-129	2.983E+07	4.861E-02	1.000E-02	1.000E-04	
TE-129M	5	8.090E+06	3.340E+01	1.000E-02	1.000E-04	
TE-131M	5	1.464E+07	1.250E+00	1.000E-02	1.000E-04	
TE-132	5	1.488E+08	3.250E+00	1.000E-02	1.000E-04	
I-131	3 TE-131M	1.033E+08	8.040E+00	1.000E-02	1.000E-04	
I-132	3 TE-132	1.510E+08	9.521E-02	1.000E-02	1.000E-04	
I-133	3	2.160E+08	8.667E-01	1.000E-02	1.000E-04	
I-134	3	2.378E+08	3.653E-02	1.000E-02	1.000E-04	
I-135	3	2.039E+08	2.744E-01	1.000E-02	1.000E-04	
XE-133	1 I-133	2.170E+08	5.290E+00	0.	0.	
XE-135	1 I-135	3.806E+07	3.821E-01	0.	0.	
CS-134	4	2.103E+07	7.524E+02	1.000E-02	1.000E-04	
CS-136	4	4.630E+06	1.300E+01	1.000E-02	1.000E-04	
CS-137	4	1.305E+07	1.099E+04	1.000E-02	1.000E-04	
BA-140	6	1.863E+08	1.279E+01	1.000E-02	1.000E-04	
LA-140	8 BA-140	1.974E+08	1.676E+00	1.000E-02	1.000E-04	
CE-141	8	1.727E+08	3.253E+01	1.000E-02	1.000E-04	
CE-143	8	1.629E+08	1.375E+00	1.000E-02	1.000E-04	
CE-144	8	1.387E+08	2.844E+02	1.000E-02	1.000E-04	
PR-143	8 CE-143	1.612E+08	1.358E+01	1.000E-02	1.000E-04	

Table 2A-6 (Cont'd)

ISOTOPIC SUBGROUP
(NUREG/CR-2326, pg 2-37)

ND-147	8	7.103E+07	1.099E+01	1.000E-02	1.000E-04
NP-239	8	2.401E+09	2.350E+00	1.000E-02	1.000E-04
PU-2[illegible]	8 CM-242	6.224E+05	3.251E+04	1.000E-02	1.000E-04
PU-239	8 NP-239	5.364E+04	8.912E+06	1.000E-02	1.000E-04
PU-240	8 CM-244	8.826E+04	2.469E+06	1.000E-02	1.000E-04
PU-241	8	2.121E+07	5.333E+03	1.000E-02	1.000E-04
AM-241	8 PU-241	1.726E+04	1.581E+05	1.000E-02	1.000E-04
CM-242	8	1.260E+07	1.630E+02	1.000E-02	1.000E-04
CM-244	8	2.885E+05	6.611E+03	1.000E-02	1.000E-04

Table 2A-7

LEAKAGE SUBGROUP
(NUREG/CR-2326, pg 2-41)

(This group input as a dummy in reference deck and overwritten by individual cases.)
(See INDIVIDUAL CASES)

LEAKAGE		1 NO	DUMMY INPUT - OVERLAYED IN ACTUAL RUN SEE TABLE 2A-15				
ABWR CS1	1.0	31.9	2.78	1.5	4.0E+07	10.	
1.0E+00		0.8	0.8	1.0E-03	1.1E-03	2.6E-04	1.5E-07

Table 2A-8

DISPERSION SUBGROUP
(NUREG/CR-2326, pg 2-45)

DISPERSION				
54.0	37.7	4	0	0

Table 2A-9

EVACUATION SUBGROUP
(NUREG/CR-2326, pg 2-47)

EVACUATE	1 NO	YES					
1.0	0.	0.	0.	24135.	0.	2.	1.0
1.0	1.	1.0	1.0	1.0	1.0	1.0	1.0
2.66E-4	2.66E-4	1.33E-4	2.66E-4				
8045.	90.	95.	3.	0			

Table 2A-10

ACUTE SUBGROUP
(NUREG/CR-2326, pg 2-53)

ACUTE		7					
T MARROW	320.	400.	510.	615.	.03	.5	1.
LLI WALL	2000.	5000.	5000.	5000.	1.	1.	1.
LUNG	5000.	14800.	22400.	24000.	.24	.73	1.
W BODY	55.	150.	280.	370.	.30	.8	0.
LUNG	3000.	3000.1	6000.	6000.	.05	1.0	0.
LLI WALL	1000.	1000.1	2500.	2500.	.05	1.0	0.
THYROID	1.E10	1.E10	1.E10	1.E10	1.01	1.0	0.0

Table 2A-11

LATENT SUBGROUP
(NUREG/CR-2326, pg 2-57)

LATENT	8						
10CENT EST	30.	5.	300.	2.5			
T MARROW	LEUKEMIA	2.836E-05	2.720E-05	1.872E 05	1.382E-05	9.720E-06	6.770E-06
4.040E-06	1.700E-06	4.900E-07	0.0	1.0			
LUNG	LUNG	2.749E-05	2.749E-05	2.749E-05	1.587E-05	8.130E-06	3.990E-06
1.500E-06	2.200E-07	0.0	0.0	0.5			
OTHER	BREAST	3.172E-05	3.172E-05	3.172E-05	1.831E-05	9.380E-06	4.600E-06
1.730E-06	2.500E-07	0.0	0.0	1.000E+09			
SKELETON	BONE	1.107E-05	1.064E-05	6.990E-06	3.020E-06	1.670E-06	9.100E-07
4.200E-07	1.200E-07	1.000E-08	0.0	1.0			
LLI WALL	GI TRK	1.688E-05	1.688E-05	1.688E-05	9.740E-06	4.990E-06	2.450E-06
9.200E-07	1.300E-07	0.0	0.0	1.0			
OTHER	OTHER	4.235E-05	3.557E-05	2.539E-05	1.466E-05	7.520E-06	3.690E-06
1.390E-06	2.000E-07	0.0	0.0	1.0			
W BODY	W BODY	1.579E-04	1.533E-04	1.274E-04	7.542E-05	4.141E-05	2.241E-05
1.000E-05	2.620E-06	5.000E-07	0.0	1.0			
THYROID	THYROID	3.34E-04					
				1.00E 09			

Table 2A-12

CHRONIC SUBGROUP
(NUREG/CR-2326, pg 2-62)

CHRONIC EXPOSURE			6			
10	1	1.000	365.	25550.	3.0	15.0
SR-90		0.0525	0.0718			
RU-106		0.0397	0.0533			
CS-137		0.0525	0.105			
PU-238		0.0529	0.107			

Table 2A-12 (Cont'd)

CHRONIC SUBGROUP
(NUREG/CR-2326, pg 2-62)

PU-239		0.0530	0.108				
PU-240		0.0530	0.108				
PU-241		0.0520	0.101				
AM-241		0.0530	0.108				
CM-242		0.0292	0.0327				
CM-244		0.0522	0.102				
3	11	1.0	365.	365.	14.0	2.0	3.3
CS-134		8.44	4.22				
LUNG		6.47E+4	7.31E+4				
T MARROW		6.50E+4	7.34E+4				
SKELETON		6.41E+4	7.24E+4				
T E C L		6.41E+4	7.24E+4				
ST WALL		7.40E+4	8.34E+4				
SI+CONT		8.05E+4	9.09E+4				
ULI WALL		7.95E+4	8.96E+4				
LLI WALL		8.28E+4	9.33E+4				
THYROID		6.49E+4	7.33E+4				
OTHER		6.27E+4	7.08E+4				
W BODY		6.32E+4	7.14E+4				
TESTES		7.57E+4	8.55E+4				
OVARIES		6.68E+4	7.55E+4				
CS-136		2.84	1.42				
LUNG		8.82E+3	8.82E+3				
T MARROW		9.29E+3	9.29E+3				
SKELETON		9.10E+3	9.10E+3				
T E C L		9.10E+3	9.10E+3				
ST WALL		1.15E+4	1.15E+4				
SI+CONT		1.19E+4	1.19E+4				
ULI WALL			1.20E+4		1.20E+4		
LLI WALL		1.35E+4	1.35E+4				
THYROID			9.23E+3		9.23E+3		
OTHER		8.88E+3	8.88E+3				
W BODY		8.96E+3	8.96E+3				
TESTES		1.03E+4	1.03E+4				
OVARIES		9.48E+3	9.48E+3				
CS-137		8.44	4.22				
LUNG		4.71E+4	5.59E+4				
T MARROW		4.73E+4	5.61E+4				
SKELETON		4.68E+4	5.56E+4				
T E C L		4.68E+4	5.56E+4				
ST WALL		5.18E+4	6.13E+4				
SI+CONT		5.39E+4	6.39E+4				
ULI WALL		5.40E+4	6.39E+4				
LLI WALL		5.64E+4	6.64E+4				
THYROID		4.68E+4	5.55E+4				
OTHER		4.60E+4	5.45E+4				

Table 2A-12 (Cont'd)

CHRONIC SUBGROUP
(NUREG/CR-2326, pg 2-62)

W BODY	4.62E+4	5.49E+4				
TESTES	5.18E+4	6.15E+4				
OVARIES	4.81E+4	5.70E+4				
2 2	1.0	365.	365.	14.0	2.0	3.3
SR-89	.397	0.402				
LUNG	2.91E+3	5.81E+2				
T MARROW	2.63E+4	5.26E+3				
SKELETON	5.95E+4	1.19E+4				
T E C L	6.00E+4	1.20E+4				
ST WALL	1.56E+4	3.12E+3				
SI+CONT	2.73E+4	5.45E+3				
ULI WALL		1.46E+5	2.91E+4			
LLI WALL		4.27E+5	8.53E+4			
THYROID		2.91E+3	5.81E+2			
OTHER	2.91E+3	5.81E+2				
W BODY	9.55E+3	1.91E+3				
TESTES	2.91E+3	5.81E+2				
OVARIES	2.91E+3	5.81E+2				
SR-90	.505	.588				
LUNG	1.59E+4	3.18E+3	5.50E+2	1.80E+1		
T MARROW	1.04E+6	2.08E+5	5.25E+4	1.29E+4	1.00E+4	3.10E+3
SKELETON	3.08E+6	6.15E+5	2.57E+5	9.81E+4	1.09E+5	4.30E+4
T E C L	2.64E+6	5.27E+5	1.93E+5	6.77E+4	7.20E+4	2.76E+4
ST WALL	2.03E+4	4.05E+3	5.50E+1	1.80E+1		
SI+CONT	2.64E+4	5.28E+3	5.50E+1	1.80E+1		
ULI WALL		1.06E+5	2.11E+4	5.00E+1	2.00E+1	
LLI WALL		4.06E+5	8.12E+4	5.00E+1		
THYROID		1.59E+4	3.18E+3	5.50E+1	1.80E+1	
OTHER	1.59E+4	3.18E+3	5.50E+1	1.30E+1		
W BODY	2.76E+5	5.52E+4	2.03E+4	7.44E+3	8.08E+3	3.13E+3
TESTES	1.59E+4	3.18E+3	5.50E+1	1.80E+1		
OVARIES	1.59E+4	3.18E+3	5.50E+1	1.80E+1		
2 9	1.0	0.0	365.	14.0	0.0	10.0
I-133	1.00E-8	0.00486				
LUNG	8.53E+2	1.58E+2				
T MARROW	7.99E+2	1.48E+2				
SKELETON	7.88E+2	1.46E+2				
T E C L	7.88E+2	1.46E+2				
ST WALL	1.11E+4	2.06E+3				
SI+CONT	2.30E+3	4.25E+2				
ULI WALL		6.16E+3	1.14E+3			
LLI WALL	9.83E+3	1.82E+3				
THYROID		1.73E+6	3.21E+5			
OTHER	9.07E+2	1.68E+2				
W BODY	1.46E+3	2.70E+2				
TESTES	7.13E+2	1.32E+2				

Table 2A-12 (Cont'd)

CHRONIC SUBGROUP
(NUREG/CR-2326, pg 2-62)

OVARIES	9.99E+2	1.85E+2				
I-131	1.00E-8	0.692				
LUNG	1.92E+3	3.56E+2				
T MARROW	1.55E+3	2.87E+2				
SKELETON	1.67E+3	3.10E+2				
T E C L	1.67E+3	3.10E+2				
ST WALL	6.16E+3	1.14E+3				
SI+CONT	1.79E+3	3.32E+2				
ULI WALL		4.49E+3	8.32E+2			
LLI WALL	1.03E+4	1.91E+3				
THYROID		9.07E+6	1.68E+6			
OTHER	2.20E+3	4.07E+2				
W BODY	4.75E+3	8.79E+2				
TESTES	7.51E+3	1.39E+2				
OVARIES	1.15E+3	2.21E+2				
4 2	1.000	3650.	3650.	2400.	5.0	5.0
CS-134	.164	.0547				
LUNG	7.31E+4					
T MARROW	7.34E+4					
SKELETON	7.24E+4					
T E C L	7.24E+4					
ST WALL	8.34E+4					
SI+CONT	9.09E+4					
ULI WALL		8.96E+4				
LLI WALL	9.33E+4					
THYROID		7.33E+4				
OTHER	7.08E+4					
W BODY	7.14E+4					
TESTES	8.55E+4					
OVARIES	7.55E+4					
CS-137	.250	.0835				
LUNG	5.59E+4					
T MARROW	5.61E+4					
SKELETON	5.56E+4					
T E C L	5.56E+4					
ST WALL	6.13E+4					
SI+CONT	6.39E+4					
ULI WALL		6.39E+4				
LLI WALL	6.64E+4					
THYROID		5.55E+4				
OTHER	5.45E+4					
W BODY	5.49E+4					
TESTES	6.15E+4					
OVARIES	5.70E+4					
SR-89	.0136	.0068				
LUNG	5.81E+2					

Table 2A-12 (Cont'd)

CHRONIC SUBGROUP
(NUREG/CR-2326, pg 2-62)

T MARROW	5.26E+3					
SKELETON	1.19E+4					
T E C L	1.20E+4					
ST WALL	3.12E+3					
SI+CONT	5.45E+3					
ULI WALL		2.91E+4				
LLL WALL		8.53E+4				
THYROID		5.81E+2				
OTHER	5.81E+2					
W BODY	1.91E+3					
TESTES	5.81E+2					
OVARIES	5.81E+2					
SR-90	1.340	0.669				
LUNG	3.18E+3	5.50E+2	1.80E+1			
T MARROW	2.08E+5	5.25E+4	1.29E+4	1.00E+4	3.10E+3	3.103+3
SKELETON	6.15E+5	2.57E+5	9.81E+4	1.09E+5	4.30E+4	4.30E+4
T E C L	5.27E+5	1.93E+5	6.77E+4	7.20E+4	2.76E+4	2.76E+4
ST WALL	4.05E+3	5.50E+1	1.80E+1			
SI+CONT	5.28E+3	5.50E+1	1.80E+1			
ULI WALL		2.11E+4	5.00E+1	2.00E+1		
LLL WALL		8.12E+4	5.00E+1			
THYROID		3.18E+3	5.50E+1	1.80E+1		
OTHER	3.18E+3	5.50E+1	1.30E+1			
W BODY	5.52E+4	2.03E+4	7.44E+3	8.08E+3	3.13E+3	3.13E+3
TESTES	3.18E+3	5.50E+1	1.80E+1			
OVARIES	3.18E+3	5.50E+1	1.80E+1			
10 11	0.333	365.	10950.	3285.	5.0	25.0
CO-58						
CO-60						
NB-95						
ZR-95						
RU-103						
RU-106						
I-131						
CS-134						
CS-136						
CS-137						

Table 2A-13

SCALE SUBGROUP
(NUREG/CR-2326, pg 2-72)

SCALE		36	NO					
1.	E02.	E03.	E05.	E07.	E01.	E12.	E13.	E1
5.	E17.	E11.	E22.	E23.	E25.	E27.	E21.	E3
2.	E33.	E35.	E37.	E31.	E42.	E43.	E45.	E4
7.	E41.	E52.	E53.	E55.	E57.	E51.	E62.	E6
3.	E65.	E67.	E61.	E7				

Table 2A-14

RESULTS SUBGROUP
(NUREG/CR-2326, pg 2-74)

RESULT	28	NO
ACUTE FATALITIES		
ACUTE INJURIES		
POP W/BMR DS>200		
RSK OF FAT-INT 2		1.0E-06
RSK OF FAT-INT 4		1.0E-06
RSK OF FAT-INT10		1.0E-06
RSK OF FAT-INT14		1.0E-06
FATAL RADIUS(MI)		
RSK OF INJ-INT 2		1.0E-06
RSK OF INJ-INT14		1.0E-06
RSK OF INJ-INT18		1.0E-06
INJUR RADIUS(MI)		
ACU BMR DS-INT 2		1.0E-02
ACU BMR DS-INT10		1.0E-02
ACU BMR DS-INT14		1.0E-02
ACU BMR DS-INT18		1.0E-02
ACU THY DS-INT 2		1.0E-02
ACU THY DS-INT10		1.0E-02
ACU THY DS-INT14		1.0E-02
ACU THY DS-INT18		1.0E-02
TOT LAT/INITIAL		
TOT LAT/TOTAL		
CANCER RSK-INT 2		1.0E-06
CANCER RSK-INT14		1.0E-06
CANCER RSK-INT18		1.0E-06
DECON AREA		
DECON DIST		
TOT WBODY MANREM		1.0E-02

Table 2A-15

INDIVIDUAL ACCIDENT EVENT GROUPS
LEAKAGE SUBGROUP
(NUREG/CR-2326, pg 2-41)

GE PROPRIETARY - PROVIDED UNDER SEPARATE COVER

Table 2A-15 (Cont'd)

INDIVIDUAL ACCIDENT EVENT GROUPS
LEAKAGE SUBGROUP
(NUREG/CR-2326, pg 2-41)

GE PROPRIETARY - PROVIDED UNDER SEPARATE COVER

Table 2A-15 (Cont'd)

INDIVIDUAL ACCIDENT EVENT GROUPS
LEAKAGE SUBGROUP
(NUREG/CR-2326, pg 2-41)

GE PROPRIETARY - PROVIDED UNDER SEPARATE COVER

Table 2A-15 (Cont'd)

INDIVIDUAL ACCIDENT EVENT GROUPS
LEAKAGE SUBGROUP
(NUREG/CR-2326, pg 2-41)

GE PROPRIETARY - PROVIDED UNDER SEPARATE COVER

Table 2A-15 (Cont'd)

INDIVIDUAL ACCIDENT EVENT GROUPS
LEAKAGE SUBGROUP
(NUREG/CR-2326, pg 2-41)

GE PROPRIETARY - PROVIDED UNDER SEPARATE COVER

Table 2A-16

FILE 20 DOSE CONVERSION FILE
(NUREG/CR-2326, pg 4-2)

```
co-58    co-60    kr-85    kr-85m   kr-87    kr-88    rb-86    sr-89    sr-90    sr-91
y-90     y-91     zr-95    zr-97    nb-95    mo-99    tc-99m   ru-103   ru-105   ru-106
rh-105   te-127   te-127m  te-129   te-129m  te-131m  te-132   sb-127   sb-129   i-131
i-132    i-133    i-134    i-135    xe-133   xe-135   cs-134   cs-136   cs-137   ba-140
la-140   ce-141   ce-143   ce-144   pr-143   nd-147   np-239   pu-238   pu-239   pu-240
pu-241   am-241   cm-242   cm-244
lung
 .5900e+05  .5900e+05  .2000e+040.  0.         0.         0.
 .1030e+03  .2100e+04  .1130e+06  .2010e+00
 .4600e+06  .4600e+06  .7400e+060 .          .1000e+060.  0.
 .2660e+03  .5580e+04  .2920e+06  .5670e+00
 .1800e+00  .1800e+000 . 0.       0.         0.         0.
 .2300e+00  .4820e+01  .2510e+03  .4470e-03
 .2100e+00  .2100e+000 . 0.       0.         0.         0.
 .9220e+01  .1300e+02  .1760e+05  .3220e-01
 .9600e+00  .9600e+000 . 0.       0.         0.         0.
 .1730e+02  .1750e+02  .8400e+05  .1720e+00
 .2000e+01  .2000e+010 . 0.       0.         0.         0.
 .1110e+03  .1300e+03  .2090e+06  .4470e+00
 .1400e+05  .1400e+050 . 0.       0.         0.         0.
 .9270e+01  .1730e+03  .1020e+05  .1940e-01
 .7800e+04  .7800e+040 . 0.       0.         0.         0.
0.          0.         0.         0.
 .1600e+05  .1600e+05  .2000e+040 . 0.       0.         0.
0.          0.         0.         0.
 .4300e+04  .4300e+040  0.         0.         0.         0.
 .8250e+02  .1930e+03  .8580e+05  .1600e+00
 .3300e+05  .3300e+050  0.         0.         0.         0.
0.          0.         0.         0.
 .2000e+06  .2000e+060  0.         0.         0.         0.
 .2810e+00  .5660e+01  .3080e+03  .5940e-03
 .1300e+06  .1300e+060  0.         0.         0.         0.
 .7720e+02  .1670e+04  .8440e+05  .1520e+00
 .1500e+05  .1500e+050  0.         0.         0.         0.
 .1310e+03  .5100e+03  .2120e+05  .4000e-01
 .3100e+05  .3100e+050  0.         0.         0.         0.
 .7820e+02  .1540e+04  .8590e+05  .1560e+00
 .1600e+05  .1600e+050  0.         0.         0.         0.
 .2180e+02  .2910e+03  .2080e+05  .3420e-01
 .8900e+02  .8900e+020  0.         0.         0.         0.
 .8120e+01  .1350e+02  .1360e+05  .2540e-01
 .5400e+05  .5400e+050  0.         0.         0.         0.
 .5520e+02  .1090e+04  .6050e+05  .1050e+00
 .2200e+04  .2200e+040  0.         0.         0.         0.
 .4850e+02  .7370e+02  .9200e+05  .1670e+00
 .2500e+07  .2500e+07  .1400e+070  0.         0.         0.
 .2060e+02  .4300e+03  .2250e+05  .4060e-01
```

Table 2A-16 (Cont'd)

FILE 20 DOSE CONVERSION FILE
(NUREG/CR-2326, pg 4-2)

.3600e+04	.3600e+040	0.	0.	0.	0.
.7530e+01	.5010e+02	.8940e+04	.1610e-01		
.1600e+04	.1600e+040	0.	0.	0.	0.
.3430e+00	.7670e+00	.4980e+03	.8780e-03		
.1200e+06	.1200e+060	0.	0.	0.	0.
.1340e+01	.3370e+02	.1350e+04	.5610e-03		
.5600e+03	.5600e+030	0.	0.	0.	0.
.1810e+01	.1830e+01	.9550e+04	.1350e-01		
.1500e+06	.1500e+060	0.	0.	0.	0.
.1020e+02	.2220e+03	.6480e+04	.6970e-02		
.1100e+05	.1100e+050	0.	0.	0.	0.
.1390e+03	.8900e+03	.1590e+06	.2940e+00		
.3000e+05	.3000e+050	0.	0.	0.	0.
.1690e+03	.2880e+04	.2230e+05	.4190e-01		
.2500e+05	.2500e+050	0.	0.	0.	0.
.7050e+02	.8650e+03	.7950e+05	.1430e+00		
.3200e+04	.3200e+040	0.	0.	0.	0.
.6990e+02	.9780e+02	.1360e+06	.2530e+00		
.2400e+04	.2400e+040	0.	0.	0.	0.
.4150e+02	.6630e+03	.4610e+05	.8220e-01		
.1000e+04	.1000e+040	0.	0.	0.	0.
.9230e+02	.1010e+03	.2700e+06	.4830e+00		
.3100e+04	.3100e+040	0.	0.	0.	0.
.6500e+02	.2910e+03	.8090e+05	.1460e+00		
.5600e+03	.5600e+030	0.	0.	0.	0.
.3870e+02	.3880e+02	.2690e+06	.5000e+00		
.2500e+04	.2500e+040	0.	0.	0.	0.
.1400e+03	.2690e+03	.2040e+06	.4000e+00		
.4100e+00	.4100e+000	0.	0.	0.	0.
.5750e+01	.3080e+02	.6440e+04	.6970e-02		
.9400e+00	.9400e+000	0.	0.	0.	0.
.1890e+02	.4160e+02	.2750e+05	.5060e-01		
.4500e+05	.4500e+05	.6000e+040	0.	0.	0.
.1660e+03	.3470e+04	.1810e+06	.3280e+00		
.8200e+04	.8200e+040	0.	0.	0.	0.
.2160e+03	.3820e+04	.2380e+06	.4440e+00		
.3400e+05	.3400e+05	.6000e+040	0.	0.	0.
.5840e+02	.1240e+04	.6450e+05	.1150e+00		
.6300e+04	.6300e+040	0.	0.	0.	0.
.3970e+02	.3460e+04	.2580e+05	.4140e-01		
.1600e+05	.1600e+050	0.	0.	0.	0.
.2330e+03	.1710e+04	.2730e+06	.5390e+00		
.6200e+05	.6200e+050	0.	0.	0.	0.
.7640e+01	.1500e+03	.8400e+04	.1500e-01		
.1300e+05	.1300e+050	0.	0.	0.	0.
.3140e+02	.2000e+03	.3720e+05	.6080e-01		
.2100e+07	.2100e+07	.8000e+060	0.	0.	0.

Table 2A-16 (Cont'd)

FILE 20 DOSE CONVERSION FILE
(NUREG/CR-2326. pg 4-2)

.4980e+01	.1070e+03	.2250e+04	.3440e-02			
.4900e+05	.4900e+050	0.	0.	0.	0.	
0.	0.	0.	0.			
.3700e+05	.3700e+05	.1000e+040	0.	0.	0.	
.1570e+02	.2700e+03	.1740e+05	.2780e-01			
.9200e+04	.9200e+04	.1000e+030	0.	0.	0.	
.1870e+02	.1740e+03	.2150e+05	.2650e-01			
.1200e+09	.1200e+09	.1900e+090	0.	0.	0.	
.5410e-01	.1140e+01	.5920e+02	.9580e-05			
.1200e+09	.1200e+09	.1700e+090	0.	0.	0.	
.2960e-01	.6220e+00	.3240e+02	.5420e-05			
.1200e+09	.1200e+09	.1700e+09	.1000e+080	0.	0.	
.5150e-01	.1080e+01	.5640e+02	.9170e-05			
.6400e+05	.6400e+05	.4660e+06	.3000e+05	.1000e+05	.1000e+05	.1000e+05
.3530e-05	.1560e-02	.7740e-02	.2940e-09			
.1300e+09	.1300e+09	.1800e+090	0.	0.	.1000e+08	
.4840e+01	.1020e+03	.5300e+04	.3220e-02			
.7600e+08	.7600e+08	.1100e+080	0.	0.	0.	
.4370e-01	.9050e+00	.4790e+02	.8310e-05			
.1300e+09	.1300e+09	.1800e+090	0.	0.	0.	
.1240e+01	.2610e+02	.1360e+04	.1070e-02			
t marrow						
.7950e+03	.3000e+04	.1000e+030	0.	0.	0.	
.1230e+03	.2500e+04	.1350e+06	.2400e+00			
.2000e+04	.2100e+05	.3500e+05	.1000e+04	.1000e+040	0.	
.2960e+03	.6220e+04	.3250e+06	.6310e+00			
.6100e+00	.6100e+600	0.	0.	0.	0.	
.2970e+00	.6220e+01	.3250e+03	.5780e-03			
.3900e+00	.3900e+000	0.	0.	0.	0.	
.1570e+02	.2210e+02	.3000e+05	.5500e-01			
.1300e+01	.1300e+010	0.	0.	0.	0.	
.1930e+02	.1960e+02	.9370e+05	.1920e+00			
.3100e+01	.3100e+010	0.	0.	0.	0.	
.1190e+03	.1390e+03	.2250e+06	.4830e+00			
.3250e+04	.6500e+040	0.	0.	0.	0.	
.1090e+02	.2020e+03	.5200e+05	.2270e-01			
.3350e+04	.1300e+050	0.	0.	0.	0.	
0.	0.	0.	0.			
.6100e+04	.1100e+06	.4200e+06	.1300e+06	.3000e+05	.3000e+05	.1000e+05
0.	0.	0.	0.			
.2150e+03	.3100e+03	.1000e+020	0.	0.	0.	
.1020e+03	.2390e+03	.1030e+06	.1930e+00			
.4700e+03	.5100e+030	0.	0.	0.	0.	
0.	0.	0.	0.			
.1430e+04	.9200e+04	.1000e+030	0.	0.	0.	
.3010e+00	.6060e+01	.3300e+03	.6390e-03			
.6700e+03	.3500e+04	.1000e+030	0.	0.	0.	

Table 2A-16 (Cont'd)

FILE 20 DOSE CONVERSION FILE
(NUREG/CR-2326. pg 4-2)

.9460e+02	.2040e+04	.1030e+06	.1670e+00		
.1900e+03	.1900e+030	0.	0.	0.	0.
.1670e+03	.6520e+03	.2510e+05	.4720e-01		
.5750e+03	.1400e+040	0.	0.	0.	0.
.9170e+02	.1800e+04	.1010e+06	.1830e+00		
.1250e+03	.1300e+030	0.	0.	0.	0.
.3130e+02	.4650e+03	.2700e+05	.4440e-01		
.1100e+02	.1100e+020	0.	0.	0.	0.
.1740e+02	.2880e+02	.2910e+05	.5420e-01		
.4050e+03	.1100e+040	0.	0.	0.	0.
.7160e+02	.1420e+04	.7830e+05	.1360e+00		
.2400e+02	.2400e+020	0.	0.	0.	0.
.6440e+02	.9990e+02	.1220e+06	.2210e+00		
.4400e+03	.3600e+04	.2600e+040	0.	0.	0.
.2650e+02	.5540e+03	.2910e+05	.5220e-01		
.2300e+02	.2300e+020	0.	0.	0.	0.
.1280e+02	.8550e+02	.1520e+05	.2740e-01		
.3900e+01	.3900e+010	0.	0.	0.	0.
.4540e+00	.1010e+01	.6590e+03	.1160e-02		
.1820e+03	.7500e+03	.5000e+020	0.	0.	0.
.4080e+01	.9200e+02	.4310e+04	.1790e-02		
.1100e+01	.1100e+010	0.	0.	0.	0.
.2420e+01	.2440e+01	.1270e+05	.1810e-01		
.3750e+03	.8300e+03	.1000e+020	0.	0.	0.
.1420e+02	.3070e+03	.9200e+04	.9920e-02		
.3000e+03	.3100e+030	0.	0.	0.	0.
.1710e+03	.1100e+04	.1920e+06	.3560e+00		
.9400e+03	.1000e+040	0.	0.	0.	0.
.2170e+03	.3630e+04	.3880e+05	.7310e-01		
.3100e+03	.3300e+030	0.	0.	0.	0.
.9070e+02	.1110e+04	.1020e+06	.1840e+00		
.4600e+02	.4600e+020	0.	0.	0.	0.
.8250e+02	.1160e+03	.1600e+06	.2970e+00		
.1500e+03	.1900e+030	0.	0.	0.	0.
.5450e+02	.8730e+03	.6060e+05	.1080e+00		
.5000e+02	.5000e+020	0.	0.	0.	0.
.1130e+03	.1230e+03	.3290e+06	.5890e+00		
.9350e+02	.9400e+020	0.	0.	0.	0.
.8130e+02	.3750e+03	.1010e+06	.1830e+00		
.2000e+02	.2000e+020	0.	0.	0.	0.
.4350e+02	.4560e+02	.3160e+06	.5890e+00		
.9100e+02	.9100e+020	0.	0.	0.	0.
.1600e+03	.3180e+03	.2260e+06	.4420e+00		
.1600e+01	.1600e+010	0.	0.	0.	0.
.1310e+02	.1840e+03	.1470e+05	.1590e-01		
.2100e+01	.2100e+010	0.	0.	0.	0.
.3160e+02	.6980e+02	.4620e+05	.8470e-01		

Table 2A-16 (Cont'd)

FILE 20 DOSE CONVERSION FILE
(NUREG/CR-2326, pg 4-2)

.4950e+04	.4300e+05	.5000e+040	0.	0.	0.	
.2030e+03	.4260e+04	.2230e+06	.4030e+00			
.3550e+04	.6000e+040	0.	0.	0.	0.	
.2640e+03	.4680e+04	.2930e+06	.5420e+00			
.3250e+04	.3100e+05	.6500e+040	0.	0.	0.	
.7560e+02	.1600e+04	.8340e+05	.1490e+00			
.2100e+04	.3400e+040	0.	0.	0.	0.	
.5000e+02	.3980e+04	.3500e+05	.5610e-01			
.6700e+03	.6800e+030	0.	0.	0.	0.	
.2610e+03	.1920e+04	.3070e+06	.6060e+00			
.1130e+03	.2700e+030	0.	0.	0.	0.	
.1650e+02	.3240e+03	.1810e+05	.3220e-01			
.9550e+02	.1100e+030	0.	0.	0.	0.	
.4830e+02	.3080e+03	.5740e+05	.9360e-01			
.2350e+03	.3600e+04	.5600e+040	0.	0.	0.	
.7840e+01	.1670e+03	.4980e+04	.7610e-02			
.1780e+02	.3400e+020	0.	0.	0.	0.	
0.	0.	0.	0.			
.1400e+03	.1900e+03	.1000e+020	0.	0.	0.	
.2480e+02	.4260e+03	.2740e+05	.4390e-01			
.6200e+02	.6400e+020	0.	0.	0.	0.	
.3490e+02	.3260e+03	.4020e+05	.4970e-01			
.1710e+03	.6000e+04	.2240e+06	.3400e+06	.3000e+06	.2300e+06	.2000e+06
.2390e+00	.5020e+01	.2620e+03	.4250e-04			
.1590e+03	.5600e+04	.2240e+06	.3500e+06	.3400e+06	.2800e+06	.3000e+06
.1190e+00	.2490e+01	.1300e+03	.2170e-04			
.1640e+03	.5600e+04	.2240e+06	.3500e+06	.3500e+06	.2700e+06	.3000e+06
.2180e+00	.4570e+01	.2380e+03	.3890e-04			
.4200e-01	.6100e+01	.1790e+04	.5700e+04	.7500e+04	.8000e+04	.9000e+04
.1030e-04	.4520e-02	.2240e-01	.8530e-09			
.2650e+03	.7200e+04	.2430e+06	.3800e+06	.3600e+06	.3100e+06	.3000e+06
.1400e+02	.2950e+03	.1530e+05	.9330e-02			
.2030e+03	.3200e+04	.2900e+04	.1200e+04	.1200e+04	.1000e+04	.5000e+03
.2050e+00	.4250e+01	.2240e+03	.3890e-04			
.2010e+03	.6600e+04	.2030e+06	.2200e+06	.1500e+06	.1000e+06	.6000e+05
.3240e+01	.6810e+02	.3560e+04	.2810e-02			
skeleton						
.3300e+03	.2500e+04	.1000e+030	0.	0.	0.	
.1230e+03	.2500e+04	.1350e+06	.2410e+00			
.8000e+03	.1800e+05	.3000e+05	.2000e+040	0.	0.	
.3010e+03	.6270e+04	.3290e+06	.6390e+00			
.1500e+00	.1500e+000	0.	0.	0.	0.	
.2960e+00	.6240e+01	.3250e+03	.5780e-03			
.1900e+00	.1900e+000	0.	0.	0.	0.	
.1580e+02	.2220e+02	.3000e+05	.5500e-01			
.8300e+00	.8300e+000	0.	0.	0.	0.	
.1980e+02	.2000e+02	.9640e+05	.1970e+00			

Table 2A-16 (Cont'd)

FILE 20 DOSE CONVERSION FILE
(NUREG/CR-2326. pg 4-2)

.1800e+01	.1800e+010	0.	0.	0.	0.	
.1230e+03	.1440e+03	.2310e+06	.4940e+00			
.1700e+04	.6500e+040	0.	0.	0.	0.	
.1080e+02	.2020e+03	.1190e+05	.2270e-01			
.3000e+04	.3000e+050	0.	0.	0.	0.	
0.	0.	0.	0.			
.3800e+04	.2600e+06	.1340e+07	.6000e+06	.2000e+06	.3000e+06	.1000e+06
0.	0.	0.	0.			
.2000e+03	.3400e+030	0.	0.	0.	0.	
.1020e+03	.2390e+03	.1040e+06	.1940e+00			
.8600e+03	.1000e+040	0.	0.	0.	0.	
0.	0.	0.	0.			
.1200e+04	.1900e+050	0.	0.	0.	0.	
.3080e+00	.6240e+01	.3380e+03	.6530e-03			
.2800e+03	.3300e+04	.1000e+030	0.	0.	0.	
.9440e+02	.2030e+04	.1040e+06	.1880e+00			
.1300e+03	.1300e+030	0.	0.	0.	0.	
.1680e+03	.6510e+03	.2530e+05	.4750e-01			
.2600e+03	.1200e+040	0.	0.	0.	0.	
.9150e+02	.1800e+04	.1010e+06	.1830e+00			
.1000e+03	.1100e+030	0.	0.	0.	0.	
.3130e+02	.4650e+03	.2690e+05	.4440e-01			
.1000e+02	.1000e+020	0.	0.	0.	0.	
.1740e+02	.2880e+02	.2910e+05	.5420e-01			
.1800e+03	.8800e+030	0.	0.	0.	0.	
.7160e+02	.1420e+04	.7810e+05	.1360e+00			
.1700e+02	.1700e+020	0.	0.	0.	0.	
.6450e+02	.9960e+02	.1220e+06	.2220e+00			
.1900e+03	.3400e+04	.2500e+040	0.	0.	0.	
.2660e+02	.5550e+03	.2900e+05	.5220e-01			
.1600e+02	.1600e+020	0.	0.	0.	0.	
.1280e+02	.8500e+02	.1520e+05	.2740e-01			
.5200e+01	.5200e+010	0.	0.	0.	0.	
.4530e+00	.1010e+01	.6600e+03	.1160e-02			
.1300e+03	.1800e+04	.2000e+030	0.	0.	0.	
.3810e+01	.8680e+02	.4020e+04	.1670e-02			
.1200e+01	.1200e+010	0.	0.	0.	0.	
.2410e+01	.2440e+01	.1270e+05	.1810e-01			
.2600e+03	.1400e+040	0.	0.	0.	0.	
.1410e+02	.3050e+03	.9150e+04	.9830e-02			
.2300e+03	.2500e+030	0.	0.	0.	0.	
.1720e+03	.1100e+04	.1940e+06	.3580e+00			
.8000e+03	.9100e+030	0.	0.	0.	0.	
.2170e+03	.3640e+04	.3870e+05	.7280e-01			
.2100e+03	.2600e+030	0.	0.	0.	0.	
.9100e+02	.1110e+04	.1020e+06	.1840e+00			
.3800e+02	.3900e+020	0.	0.	0.	0.	

Table 2A-16 (Cont'd)

FILE 20 DOSE CONVERSION FILE
(NUREG/CR-2326, pg 4-2)

.8300e+02	.1160e+03	.1600e+06	.2990e+00			
.1200e+03	.2100e+030	0.	0.	0.	0.	
.5450e+02	.8750e+03	.6030e+05	.1080e+00			
.4700e+02	.4700e+020	0.	0.	0.	0.	
.1130e+03	.1230e+03	.3290e+06	.5890e+00			
.9200e+02	.9200e+020	0.	0.	0.	0.	
.8130e+02	.3750e+03	.1010e+06	.1830e+00			
.1900e+02	.1900e+020	0.	0.	0.	0.	
.4570e+02	.4570e+02	.3160e+06	.5890e+00			
.8700e+02	.8700e+020	0.	0.	0.	0.	
.1620e+03	.3220e+03	.2300e+06	.4500e+00			
.3600e+00	.3600e+000	0.	0.	0.	0.	
.1300e+02	.1820e+03	.1450e+05	.1570e-01			
.7200e+00	.7200e+000	0.	0.	0.	0.	
.3170e+02	.6990e+02	.4620e+05	.8470e-01			
.2000e+04	.4200e+05	.5000e+040	0.	0.	0.	
.2030e+03	.4260e+04	.2230e+06	.4030e+00			
.2000e+04	.5900e+040	0.	0.	0.	0.	
.2650e+03	.4690e+04	.2930e+06	.5440e+00			
.1300e+04	.3100e+05	.5000e+040	0.	0.	0.	
.7590e+02	.1600e+04	.8370e+05	.1490e+00			
.2000e+04	.5200e+040	0.	0.	0.	0.	
.5030e+02	.4040e+04	.3500e+05	.5610e-01			
.6700e+03	.7000e+030	0.	0.	0.	0.	
.2660e+03	.1950e+04	.3120e+06	.6140e+00			
.6100e+02	.3200e+030	0.	0.	0.	0.	
.1650e+02	.3220e+03	.1810e+05	.3220e-01			
.7100e+02	.1100e+030	0.	0.	0.	0.	
.4790e+02	.3060e+03	.5700e+05	.9310e-01			
.2000e+03	.7200e+04	.1180e+050	0.	0.	0.	
.7820e+01	.1670e+03	.4950e+04	.7550e-02			
.2400e+02	.8600e+020	0.	0.	0.	0.	
0.	0.	0.	0.			
.8500e+02	.1900e+03	.4000e+02	.1000e+020	0.	0.	
.2460e+02	.4230e+03	.2720e+05	.4340e-01			
.4900e+02	.5400e+02	.2600e+02	.4000e+02	.4000e+02	.3000e+02	.4000e+02
.3490e+02	.3250e+03	.4010e+05	.4950e-01			
.3500e+05	.3400e+07	.1270e+09	.1900e+09	.1700e+09	.1400e+09	.1300e+09
.2260e+00	.4740e+01	.2480e+03	.4020e-04			
.3300e+05	.3100e+07	.1270e+09	.2000e+09	.1900e+09	.1800e+09	.1700e+09
.1110e+00	.2340e+01	.1220e+03	.2030e-04			
.3300e+05	.3200e+07	.1270e+09	.2000e+09	.1900e+09	.1800e+09	.1700e+09
.2050e+00	.4320e+01	.2250e+03	.3660e-04			
.1200e+01	.2700e+04	.9770e+06	.3120e+07	.4200e+07	.4700e+07	.4000e+07
.9460e-05	.4170e-02	.2070e-01	.7860e-09			
.3900e+05	.3600e+07	.1360e+09	.2100e+09	.2000e+09	.1800e+09	.1700e+09
.1300e+02	.2720e+03	.1420e+05	.8620e-02			

Table 2A-16 (Cont'd)

FILE 20 DOSE CONVERSION FILE
(NUREG/CR-2326, pg 4-2)

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.4300e+05 .1800e+07 .1600e+07 .7000e+06 .7000e+06 .6000e+06 .4000e+06
.1930e+00 .4000e+01 .2120e+03 .3680e-04
.4100e+05 .3700e+07 .1160e+09 .1300e+09 .8000e+08 .5000e+08 .4000e+08
.3070e+01 .6440e+02 .3360e+04 .2670e-02
t c l
.3300e+03 .2500e+04 .1000e+030 0.        0.        0.
.1300e+03 .2640e+04 .1430e+06 .2530e+00
.8000e+03 .1800e+05 .3000e+05 .2000e+040 0.        0.
.3120e+03 .6540e+04 .3420e+06 .6640e+00
.3700e+00 .3700e+000 0.       0.        0.        0.
.3110e+00 .6540e+02 .3410e+03 .6080e-03
.2800e+00 .2800e+000 0.       0.        0.        0.
.1670e+02 .2360e+02 .3190e+05 .5860e-01
.1000e+01 .1000e+010 0.       0.        0.        0.
.2050e+02 .2080e+02 .9990e+05 .2040e+00
.2400e+01 .2400e+010 0.       0.        0.        0.
.1270e+03 .1480e+03 .2400e+06 .5140e+00
.1700e+04 .6500e+040 0.       0.        0.        0.
.1150e+02 .2140e+03 .1270e+05 .2400e-01
.3200e+04 .3000e+050 0.       0.        0.        0.
0.        0.        0.        0.
.3600e+04 .2400e+06 .1060e+07 .5000e+06 .2000e+06 .2000e+060.
0.        0.        0.        0.
.2200e+03 .3700e+030 0.       0.        0.        0.
.1070e+03 .2510e+03 .1100e+06 .2030e+00
.8000e+03 .9700e+030 0.       0.        0.        0.
0.        0.        0.        0.
.1200e+04 .2060e+050 0.       0.        0.        0.
.3140e+00 .6330e+01 .3440e+03 .6670e-03
.2800e+03 .3300e+040 0.       0.        0.        0.
.9980e+02 .2150e+04 .1100e+06 .1970e+00
.1300e+03 .1300e+030 0.       0.        0.        0.
.1750e+03 .6850e+03 .2640e+05 .4970e-01
.2600e+03 .1200e+040 0.       0.        0.        0.
.9710e+02 .1910e+04 .1070e+06 .1940e+00
.1000e+03 .1100e+030 0.       0.        0.        0.
.3320e+02 .4960e+03 .2850e+05 .4690e-01
.1000e+02 .1000e+020 0.       0.        0.        0.
.1870e+02 .3110e+02 .3140e+05 .5830e-01
.1800e+03 .8800e+030 0.       0.        0.        0.
.7530e+02 .1490e+04 .8230e+05 .1430e+00
.1700e+02 .1700e+020 0.       0.        0.        0.
.6780e+02 .1050e+03 .1290e+06 .2330e+00
.1900e+03 .3400e+04 .2500e+040 0.       0.        0.
.2790e+02 .5820e+03 .3060e+05 .5470e-01
.1600e+02 .1600e+020 0.       0.        0.        0.
.1370e+02 .9090e+02 .1610e+05 .2920e-01
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Table 2A-16 (Cont'd)

FILE 20 DOSE CONVERSION FILE
(NUREG/CR-2326, pg 4-2)

.5400e+01	.5400e+010	0.	0.	0.	0.
.4770e+00	.1070e+01	.6930e+03	.1220e-02		
.1300e+03	.1900e+04	.1000e+030	0.	0.	0.
.4670e+01	.1050e+03	.4950e+04	.2060e-02		
.1200e+01	.1200e+010	0.	0.	0.	0.
.2560e+01	.2580e+01	.1350e+05	.1910e-01		
.2600e+03	.1400e+040	0.	0.	0.	0.
.1510e+02	.3260e+03	.9810e+04	.1060e-01		
.2300e+03	.2500e+030	0.	0.	0.	0.
.1810e+03	.1160e+04	.2030e+06	.3780e+00		
.8000e+03	.9100e+030	0.	0.	0.	0.
.2290e+03	.3830e+04	.4130e+05	.7780e-01		
.2100e+03	.2600e+030	0.	0.	0.	0.
.9540e+02	.1170e+04	.1080e+06	.1930e+00		
.3800e+02	.3900e+020	0.	0.	0.	0.
.8710e+02	.1220e+03	.1690e+06	.3140e+00		
.1200e+03	.2100e+030	0.	0.	0.	0.
.5730e+02	.9170e+03	.6370e+05	.1140e+00		
.4700e+02	.4700e+020	0.	0.	0.	0.
.1190e+03	.1300e+03	.3460e+06	.6190e+00		
.9200e+02	.9200e+020	0.	0.	0.	0.
.8540e+02	.3950e+03	.1060e+06	.1920e+00		
.1900e+02	.1900e+020	0.	0.	0.	0.
.4810e+02	.4820e+02	.3340e+06	.6190e+00		
.8700e+02	.8700e+020	0.	0.	0.	0.
.1680e+03	.3340e+03	.2370e+06	.4640e+00		
.9200e+00	.9200e+000	0.	0.	0.	0.
.1430e+02	.2010e+03	.1600e+05	.1740e-01		
.1300e+01	.1300e+010	0.	0.	0.	0.
.3370e+02	.7420e+02	.4910e+05	.9030e-01		
.2000e+04	.4200e+05	.5000e+040	0.	0.	0.
.2140e+03	.4490e+04	.2350e+06	.4250e+00		
.2000e+04	.5900e+040	0.	0.	0.	[illegible]
.2800e+03	.4950e+04	.3090e+06	.5750e+00		
.1300e+04	.3100e+05	.5000e+040	0.	0.	0.
.7940e+02	.1680e+04	.8760e+05	.1560e+00		
.2100e+04	.5500e+040	0.	0.	0.	0.
.5260e+02	.4180e+04	.3690e+05	.5920e-01		
.6800e+03	.7100e+030	0.	0.	0.	0.
.2740e+03	.2010e+04	.3210e+06	.6330e+00		
.6100e+02	.3200e+030	0.	0.	0.	0.
.1790e+02	.3500e+03	.1960e+05	.3500e-01		
.7200e+02	.1100e+030	0.	0.	0.	0.
.5140e+02	.3280e+03	.6110e+05	.9940e-01		
.1900e+03	.6700e+04	.1130e+050	0.	0.	0.
.8410e+01	.1790e+03	.5400e+04	.8250e-02		
.2600e+02	.9300e+020	0.	0.	0.	0.

Table 2A-16 (Cont'd)

FILE 20 DOSE CONVERSION FILE
(NUREG/CR-2326, pg 4-2)

0.	0.	0.	0.			
.8600e+02	.1900e+03	.4000e+020	0.	0.	0.	
.2640e+02	.4540e+03	.2930e+05	.4670e-01			
.4800e+02	.5200e+020	0.	0.	0.	0.	
.3740e+02	.3490e+03	.4300e+05	.5310e-01			
.6300e+02	.6000e+04	.2240e+06	.3400e+06	.3000e+06	.2300e+06	.3000e+06
.2790e+00	.5870e+01	.3060e+03	.4940e-04			
.5900e+02	.5600e+04	.2240e+06	.3500e+06	.3400e+06	.2800e+06	.3000e+06
.1370e+00	.2880e+01	.1500e+03	.2500e-04			
.5900e+02	.5700e+04	.2240e+06	.3600e+06	.3400e+06	.3700e+06	.3000e+06
.2530e+00	.5320e+01	.2730e+03	.4500e-04			
.5900e-01	.1100e+02	.1990e+04	.6000e+04	.8000e+04	.8000e+04	.9000e+04
.1110e-04	.4880e-02	.2430e-01	.9220e-02			
.9900e+02	.7300e+04	.2530e+06	.3900e+06	.3500e+06	.4000e+06	.3000e+06
.1520e+02	.3190e+03	.1660e+05	.1010e-01			
.7600e+02	.3200e+04	.2900e+04	.1300e+04	.1100e+04	.1000e+04	.5000e+03
.2410e+00	.4990e+01	.2640e+03	.4580e-04			
.7400e+02	.6600e+04	.2030e+06	.2300e+06	.1500e+06	.9000e+05	.6000e+05
.3500e+01	.7350e+02	.3830e+04	.3030e-02			
st wall						
.9600e+03	.5100e+04	.1000e+030	0.	0.	0.	
.9530e+02	.1940e+04	.1040e+06	.1860e+00			
.2300e+04	.3700e+05	.6000e+05	.3000e+040	0.	0.	
.2510e+03	.5260e+04	.2750e+06	.5330e+00			
.1800e+00	.1800e+000	0.	0.	0.	0.	
.2430e+00	.5100e+01	.2660e+03	.4750e-03			
.2200e+00	.2200e+000	0.	0.	0.	0.	
.6800e+01	.9580e+01	.1300e+05	.2380e-01			
.1000e+01	.1000e+010	0.	0.	0.	0.	
.1690e+02	.1710e+02	.8200e+05	.1680e+00			
.2100e+01	.2100e+010	0.	0.	0.	0.	
.1060e+03	.1230e+03	.1990e+06	.4250e+00			
.2200e+04	.7000e+040	0.	0.	0.	0.	
.8310e+01	.1550e+03	.9200e+04	.1740e-01			
.8700e+03	.1800e+040	0.	0.	0.	0.	
0.	0.	0.	0.			
.8900e+03	.6100e+04	.2100e+04	.2000e+030	0.	0.	
0.	0.	0.	0.			
.6200e+03	.6800e+030	0.	0.	0.	0.	
.8240e+02	.1940e+03	.8380e+05	.1560e+00			
.1400e+04	.1400e+040	0.	0.	0.	0.	
0.	0.	0.	0.			
.1100e+04	.1600e+040	0.	0.	0.	0.	
.2740e+00	.5520e+01	.3000e+03	.5810e-03			
.9300e+03	.5700e+04	.1000e+030	0.	0.	0.	
.7480e+02	.1610e+04	.8170e+05	.1480e+00			
.1800e+04	.1800e+040	0.	0.	0.	0.	

Table 2A-16 (Cont'd)

FILE 20 DOSE CONVERSION FILE
(NUREG/CR-2326, pg 4-2)

.1370e+03	.5340e+03	.2060e+05	.3890e-01		
.7600e+03	.2300e+040	0.	0.	0.	0.
.7010e+02	.1380e+04	.7700e+05	.1400e+00		
.8500e+03	.8600e+030	0.	0.	0.	0.
.2090e+02	.2660e+03	.2070e+05	.3420e-01		
.2700e+02	.2700e+020	0.	0.	0.	0.
.6250e+01	.1040e+02	.1050e+05	.1960e-01		
.6600e+03	.1900e+040	0.	0.	0.	0.
.5830e+02	.1160e+04	.6400e+05	.1110e+00		
.3200e+03	.3200e+030	0.	0.	0.	0.
.4920e+02	.7320e+02	.9370e+05	.1700e+00		
.3200e+04	.7600e+04	.3400e+040	0.	0.	0.
.2150e+02	.4500e+03	.2360e+05	.4250e-01		
.2700e+03	.2700e+030	0.	0.	0.	0.
.5560e+01	.3700e+02	.6580e+04	.1190e-01		
.1600e+03	.1600e+030	0.	0.	0.	0.
.3570e+00	.7980e+00	.5180e+03	.9140e-03		
.5200e+03	.1500e+040	0.	0.	0.	0.
.1070e+01	.2850e+02	.1050e+04	.4390e-03		
.5700e+02	.5700e+020	0.	0.	0.	0.
.1820e+01	.1840e+01	.9550e+04	.1360e-01		
.1200e+04	.1800e+040	0.	0.	0.	0.
.1040e+02	.2260e+03	.6680e+04	.7190e-02		
.1000e+04	.1000e+040	0.	0.	0.	0.
.1260e+03	.8170e+03	.1440e+06	.2660e+00		
.2200e+04	.2400e+040	0.	0.	0.	0.
.1610e+03	.2780e+04	.1660e+05	.3110e-01		
.1100e+04	.1200e+040	0.	0.	0.	0.
.7150e+02	.8780e+03	.8060e+05	.1450e+00		
.4300e+03	.4300e+030	0.	0.	0.	0.
.6530e+02	.9150e+02	.1270e+06	.2360e+00		
.2400e+03	.2800e+030	0.	0.	0.	0.
.4330e+02	.6920e+03	.4800e+05	.8580e-01		
.3700e+03	.3700e+030	0.	0.	0.	0.
.9060e+02	.99[illegible]+02	.2650e+06	.4750e+00		
.3900e+03	.3900e+030	0.	0.	0.	0.
.6680e+02	.2960e+03	.8310e+05	.1510e+00		
.2700e+03	.2700e+030	0.	0.	0.	0.
.3580e+02	.3580e+02	.2480e+06	.4610e+00		
.4200e+03	.4200e+030	0.	0.	0.	0.
.1340e+03	.2520e+03	.1940e+06	.3810e+00		
.4000e+00	.4000e+000	0.	0.	0.	0.
.4440e+01	.6240e+02	.4970e+04	.5390e-02		
.1000e+01	.1000e+010	0.	0.	0.	0.
.1440e+02	.3170e+02	.2090e+05	.3860e-01		
.2500e+04	.4800e+05	.6000e+040	0.	0.	0.
.1620e+03	.3390e+04	.1770e+06	.3220e+00		

Table 2A-16 (Cont'd)

FILE 20 DOSE CONVERSION FILE
(NUREG/CR-2326, pg 4-2)

.2500e+04	.6900e+040	0.	0.	0.	0.	
.1920e+03	.3400e+04	.2120e+06	.3940e+00			
.1600e+04	.3300e+05	.7000e+040	0.	0.	0.	
.6180e+02	.1310e+04	.6820e+05	.1220e+00			
.9600e+03	.1500e+040	0.	0.	0.	0.	
.3950e+02	.3360e+04	.2630e+05	.4190e-01			
.1700e+04	.1700e+040	0.	0.	0.	0.	
.2250e+03	.1650e+04	.2640e+06	.5190e+00			
.4200e+03	.5900e+030	0.	0.	0.	0.	
.5890e+01	.1150e+03	.6470e+04	.1150e-01			
.7800e+03	.7900e+030	0.	0.	0.	0.	
.2700e+02	.1720e+03	.5210e+05	.5220e-01			
.2600e+04	.4000e+04	.8000e+030	0	0.	0.	
.4500e+01	.9690e+02	.1740e+04	.2[illegible]50e-02			
.6100e+03	.6200e+030	0.	0.	0.	0.	
0.	0.	0.	0.			
.6500e+03	.7300e+030	0.	0.	0.	0.	
.1520e+02	.2620e+03	.1690e+05	.2690e-01			
.5200e+03	.5300e+030	0.	0.	0.	0.	
.1400e+02	.1310e+03	.1610e+05	.1990e-01			
.1200e+04	.7200e+04	.2230e+06	.3400e+06	.3000e+06	.2300e+06	.2000e+06
.3520e-01	.7380e+00	.3850e+02	.6220e-05			
.1100e+04	.6700e+04	.2230e+06	.3500e+06	.3400e+06	.2800e+06	.3000e+06
.2110e-01	.4420e+00	.2300e+02	.3860e-05			
.1100e+04	.6800e+04	.2230e+06	.3500e+06	.3500e+06	.2700e+06	.3000e+06
.3440e-01	.7230e+00	.3780e+02	.6140e-05			
.1100e+02	.1700e+02	.1780e+04	.5600e+04	.7600e+04	.8000e+04	.8000e+04
.3050e-05	.1350e-02	.6680e-02	.2540e-09			
.1300e+04	.8260e+04	.2420e+06	.3700e+06	.3600e+06	.3200e+06	.3000e+06
.4180e+01	.8780e+02	.4570e+04	.2780e-02			
.1300e+04	.4500e+04	.2900e+04	.1300e+04	.1100e+04	.1200e+04	.1000e+04
.2760e-01	.5710e+00	.3020e+02	.5250e-05			
.1300e+04	.7900e+04	.2020e+06	.2300e+06	.1400e+06	.1000e+06	.6000e+05
.1030e+01	.2160e+02	.1130e+04	.8920e-03			
si+cont						
.1900e+04	.2800e+040	0.	0.	0.	0.	
.8430e+02	.1710e+04	.9290e+05	.1640e+00			
.4700e+04	.1200e+05	.1300e+05	.1000e+040	0.	0.	
.2170e+03	.4560e+04	.2380e+06	.4640e+00			
.1800e+00	.1800e+000	0.	0.	0.	0.	
.1760e+00	.3680e+01	.1920e+03	.3420e-03			
.2300e+00	.2300e+000	0.	0.	0.	0.	
.7330e+01	.1030e+02	.1400e+05	.2570e-01			
.1000e+01	.1000e+010	0.	0.	0.	0.	
.1410e+02	.1420e+02	.6820e+05	.1400e+00			
.2200e+01	.2200e+010	0.	0.	0.	0.	
.9180e+02	.1070e+03	.1720e+06	.3670e+00			

Table 2A-16 (Cont'd)

FILE 20 DOSE CONVERSION FILE
(NUREG/CR-2326. pg 4-2)

.1800e+04	.6600e+040	0.	0.	0.	0.
.7670e+01	.1430e+03	.8450e+04	.1600e-01		
.1200e+04	.2200e+040	0.	0.	0.	0.
0.	0.	0.	0.		
.1200e+04	.6400e+04	.2100e+04	.1000e+03	.1000e+030	0.
0.	0.	0.	0.		
.9600e+03	.1000e+040	0.	0.	0.	0.
.6510e+02	.1520e+03	.6860e+05	.1270e+00		
.3200e+04	.3300e+040	0.	0.	0.	0.
0.	0.	0.	0.		
.2700e+04	.3300e+04	.1000e+030	0.	0.	0.
.2270e+00	.4580e+01	.2490e+03	.4810e-03		
.2000e+04	.3200e+040	0.	0.	0.	0.
.6170e+02	.1330e+04	.6750e+05	.1220e+00		
.3900e+04	.3900e+040	0.	0.	0.	0.
.1010e+03	.3930e+03	.1700e+05	.3190e-01		
.1600e+04	.1900e+040	0.	0.	0.	0.
.6470e+02	.1270e+04	.7110e+05	.1290e+00		
.1900e+04	.1900e+040	0.	0.	0.	0.
.1700e+02	.2260e+03	.1620e+05	.2670e-01		
.1100e+02	.1100e+020	0.	0.	0.	0.
.6250e+01	.1040e+02	.1050e+05	.1950e-01		
.1400e+04	.1700e+040	0.	0.	0.	0.
.4220e+02	.8360e+03	.4630e+05	.8030e-01		
.4900e+03	.4900e+030	0.	0.	0.	0.
.3750e+02	.5710e+02	.7120e+05	.1290e+00		
.7400e+04	.1000e+05	.3000e+040	0.	0.	0.
.1580e+02	.3300e+03	.1730e+05	.3110e-01		
.5900e+03	.5900e+030	0.	0.	0.	0.
.5990e+01	.3990e+02	.7090e+04	.1280e-01		
.2400e+03	.2400e+030	0.	0.	0.	0.
.2630e+00	.5870e+00	.3810e+03	.6720e-03		
.9500e+03	.2400e+040	0.	0.	0.	0.
.5160e+00	.1530e+02	.4740e+03	.1970e-03		
.3800e+02	.3800e+020	0.	0.	0.	0.
.1390e+01	.1410e+01	.7320e+04	.1040e-01		
.2200e+04	.2700e+040	0.	0.	0.	0.
.7700e+01	.1670e+03	.4780e+04	.5140e-02		
.1700e+04	.1700e+040	0.	0.	0.	0.
.1140e+03	.7230e+03	.1300e+06	.2400e+00		
.1500e+04	.1500e+040	0.	0.	0.	0.
.1340e+03	.2290e+04	.1760e+05	.3310e-01		
.2400e+04	.2500e+040	0.	0.	0.	0.
.5480e+02	.6730e+03	.6180e+05	.1110e+00		
.6500e+03	.6500e+030	0.	0.	0.	0.
.5690e+02	.7960e+02	.1100e+06	.2060e+00		
.9600e+02	.1300e+030	0.	0.	0.	0.

Table 2A-16 (Cont'd)

FILE 20 DOSE CONVERSION FILE
(NUREG/CR-2326. pg 4-2)

.3170e+02	.5080e+03	.3530e+05	.6280e-01			
.8700e+02	.8700e+020	0.	0.	0.	0.	
.7340e+02	.8040e+02	.2140e+06	.3830e+00			
.1200e+03	.1200e+030	0.	0.	0.	0.	
.5060e+02	.2250e+03	.6300e+05	.1140e+00			
.4600e+02	.4600e+020	0.	0.	0.	0.	
.3170e+02	.3170e+02	.2200e+06	.4080e+00			
.1300e+03	.1300e+030	0.	0.	0.	0.	
.1140e+03	.2170e+03	.1660e+06	.3250e+00			
.4200e+00	.4200e+000	0.	0.	0.	0.	
.4090e+01	.5750e+02	.4580e+04	.4970e-02			
.1000e+01	.1000e+010	0.	0.	0.	0.	
.1500e+02	.3300e+02	.2180e+05	.4000e-01			
.2500e+04	.5200e+05	.7600e+040	0.	0.	0.	
.1320e+03	.2770e+04	.1450e+06	.2620e+00			
.2600e+04	.7500e+040	0.	0.	0.	0.	
.1770e+03	.3140e+04	.1960e+06	.3640e+00			
.1500e+04	.3500e+05	.7000e+040	0.	0.	0.	
.4470e+02	.9440e+03	.4920e+05	.8780e-01			
.1600e+04	.2300e+040	0.	0.	0.	0.	
.3100e+02	.2770e+04	.1980e+05	.3170e-01			
.3600e+04	.3600e+040	0.	0.	0.	0.	
.1880e+03	.1380e+04	.2210e+06	.4360e+00			
.1000e+04	.1100e+040	0.	0.	0.	0.	
.5800e+01	.1140e+03	.6380e+04	.1140e-01			
.1800e+04	.1800e+040	0.	0.	0.	0.	
.2450e+02	.1560e+03	.2910e+05	.4750e-01			
.6700e+04	.7400e+04	.4000e+030	0.	0.	0.	
.3870e+01	.8310e+02	.1670e+04	.2550e-02			
.1500e+04	.1500e+040	0.	0.	0.	0.	
0.	0.	0.	0.			
.1600e+04	.1600e+040	0.	0.	0.	0.	
.1180e+02	.2030e+03	.1310e+05	.2090e-01			
.1200e+04	.1200e+040	0.	0.	0.	0.	
.1460e+02	.1360e+03	.1680e+05	.2080e-01			
.2900e+04	.9000e+04	.2310e+06	.3300e+06	.3000e+06	.2300e+06	.2000e+06
.1540e-01	.3230e+00	.1680e+02	.2730e-05			
.2700e+04	.8400e+04	.2220e+06	.3500e+06	.3400e+06	.2800e+06	.3000e+06
.9990e-02	.2100e+00	.1090e+02	.1830e-05			
.2700e+04	.8500e+04	.2210e+06	.3600e+06	.3400e+06	.2700e+06	.3000e+06
.1680e-01	.3520e+00	.1840e+02	.3000e-05			
.2700e+02	.3500e+02	.1760e+04	.5600e+04	.7600e+04	.8000e+04	.8000e+04
.2110e-05	.9280e-03	.4620e-02	.1750e-09			
.3100e+04	.9700e+04	.2400e+06	.3700e+06	.3600e+06	.3200e+06	.3000e+06
.2880e+01	.6060e+02	.3150e+04	.1920e-02			
.3200e+04	.6500e+04	.2800e+04	.1700e+04	.1000e+04	.1000e+04	.1000e+04
.9920e-02	.2050e+00	.1090e+02	.1880e-05			

Table 2A-16 (Cont'd)

FILE 20 DOSE CONVERSION FILE
(NUREG/CR-2326, pg 4-2)

.3100e+04	.9800e+04	.2000e+06	.2300e+06	.1500e+06	.9000e+05	.6000e+05
.8240e+00	.1730e+02	.9020e+03	.7140e-03			
uli wall						
.3400e+04	.4500e+040	0.	0.	0.	0.	
.9730e+02	.1980e+04	.1070e+06	.1900e+00			
.8500e+04	.1800e+05	.1700e+05	.1000e+040	0.	0.	
.2730e+03	.5730e+04	.3000e+06	.5830e+00			
.1800e+00	.1800e+000	0.	0.	0.	0.	
.1890e+00	.3960e+01	.2070e+03	.3690e-03			
.2300e+00	.2300e+000	0.	0.	0.	0.	
.7540e+01	.1060e+02	.1440e+05	.2640e-01			
.1000e+01	.1000e+010	0.	0.	0.	0.	
.1900e+02	.1930e+02	.9200e+05	.1890e+00			
.2200e+01	.2200e+010	0.	0.	0.	0.	
.1330e+03	.1550e+03	.2510e+06	.5360e+00			
.2100e+04	.6900e+040	0.	0.	0.	0.	
.8950e+01	.1670e+03	.9900e+04	.1870e-01			
.4900e+04	.5800e+040	0.	0.	0.	0.	
0.	0.	0.	0.			
.4200e+04	.9400e+04	.2600e+040	0.	0.	0.	
0.	0.	0.	0.			
.2400e+04	.2400e+040	0.	0.	0.	0.	
.7380e+02	.1730e+03	.7920e+05	.1470e+00			
.1700e+05	.1700e+050	0.	0.	0.	0.	
0.	0.	0.	0.			
.1600e+05	.1800e+050	0.	0.	0.	0.	
.3010e+00	.6060e+01	.3300e+03	.6390e-03			
.5500e+04	.7400e+040	0.	0.	0.	0.	
.6960e+02	.1510e+04	.7600e+05	.1370e+00			
.1300e+05	.1300e+050	0.	0.	0.	0.	
.1100e+03	.4290e+03	.2050e+05	.3860e-01			
.3200e+04	.3600e+040	0.	0.	0.	0.	
.7550e+02	.1490e+04	.8300e+05	.1510e+00			
.8600e+04	.8600e+040	0.	0.	0.	0.	
.1830e+02	.2380e+03	.1780e+05	.2920e-01			
.1300e+02	.1300e+020	0.	0.	0.	0.	
.6120e+01	.1020e+02	.1020e+05	.1910e-01			
.4100e+04	.4500e+040	0.	0.	0.	0.	
.4540e+02	.9000e+03	.4980e+05	.8640e-01			
.1100e+04	.1100e+040	0.	0.	0.	0.	
.4050e+02	.5140e+02	.7690e+05	.1390e+00			
.4200e+05	.4700e+05	.3000e+040	0.	0.	0.	
.1710e+02	.3580e+03	.1880e+05	.3390e-01			
.2500e+04	.2500e+040	0.	0.	0.	0.	
.6160e+01	.4100e+02	.7290e+04	.1320e-01			
.7300e+03	.7300e+030	0.	0.	0.	0.	
.2820e+00	.6300e+00	.4090e+03	.7220e-03			

Table 2A-16 (Cont'd)

FILE 20 DOSE CONVERSION FILE
(NUREG/CR-2326. pg 4-2)

.5500e+04	.9900e+040	0.	0.	0.	0.
.6880e+00	.1920e+02	.6560e+03	.2730e-03		
.2600e+02	.2600e+020	0.	0.	0.	0.
.1530e+01	.1540e+01	.8020e+04	.1140e-01		
.1200e+05	.1300e+050	0.	0.	0.	0.
.8410e+01	.1830e+03	.5220e+04	.5610e-02		
.4900e+04	.5000e+040	0.	0.	0.	0.
.1350e+03	.8530e+03	.1550e+06	.2860e+00		
.3100e+04	.3200e+040	0.	0.	0.	0.
.1530e+03	.2620e+04	.1810e+05	.3420e-01		
.1000e+05	.1000e+050	0.	0.	0.	0.
.6000e+02	.7360e+03	.6760e+05	.1210e+00		
.1300e+04	.1300e+040	0.	0.	0.	0.
.6700e+02	.9370e+02	.1310e+06	.2420e+00		
.1700e+03	.2000e+030	0.	0.	0.	0.
.3410e+02	.5450e+03	.3780e+05	.6750e-01		
.9800e+02	.9800e+020	0.	0.	0.	0.
.8450e+02	.9260e+02	.2470e+06	.4420e+00		
.2300e+03	.2300e+030	0.	0.	0.	0.
.5670e+02	.2520e+03	.7060e+05	.1280e+00		
.3700e+02	.3700e+020	0.	0.	0.	0.
.3720e+02	.3730e+02	.2580e+06	.4810e+00		
.2100e+03	.2100e+030	0.	0.	0.	0.
.1420e+03	.2680e+03	.2110e+06	.4140e+00		
.4200e+00	.4200e+000	0.	0.	0.	0.
.4120e+01	.5790e+02	.4620e+04	.5000e-02		
.1000e+01	.1000e+010	0.	0.	0.	0.
.1540e+02	.3410e+02	.2250e+05	.4140e-01		
.2500e+04	.5200e+05	.6000e+040	0.	0.	0.
.1500e+03	.3130e+04	.1640e+06	.2970e+00		
.2500e+04	.7300e+040	0.	0.	0.	0.
.2080e+03	.3670e+04	.2300e+06	.4250e+00		
.1600e+04	.3500e+05	.6000e+040	0.	0.	0.
.4810e+02	.1020e+04	.5300e+05	.9440e-01		
.4900e+04	.5500e+040	0.	0.	0.	0.
.3600e+02	.3490e+04	.2120e+05	.3390e-01		
.1100e+05	.1100e+050	0.	0.	0.	0.
.2410e+03	.1770e+04	.2820e+06	.5580e+00		
.5200e+04	.5300e+040	0.	0.	0.	0.
.5710e+01	.1120e+03	.6270e+04	.1120e-01		
.7600e+04	.7600e+040	0.	0.	0.	0.
.2650e+02	.1690e+03	.3140e+05	.5110e-01		
.3900e+05	.4200e+05	.1000e+040	0.	0.	0.
.4680e+01	.1010e+03	.1660e+04	.2530e-02		
.8700e+04	.8800e+040	0.	0.	0.	0.
0.	0.	0.	0.		
.7800e+04	.7800e+040	0.	0.	0.	0.

Table 2A-16 (Cont'd)

FILE 20 DOSE CONVERSION FILE
(NUREG/CR-2326. pg 4-2)

.1250e+02	.2150e+03	.1380e+05	.2210e-01			
.5600e+04	.5600e+040	0.	0.	0.	0.	
.1480e+02	.1380e+03	.1700e+05	.2100e-01			
.1700e+05	.2400e+05	.2260e+06	.3400e+06	.2900e+06	.2200e+06	.3000e+06
.1630e-01	.3430e+00	.1790e+02	.2890e-05			
.1600e+05	.2200e+05	.2280e+06	.3500e+06	.3300e+06	.3700e+06	.2000e+06
.1150e-01	.2410e+00	.1250e+02	.2100e-05			
.1600e+05	.2300e+05	.2270e+06	.3500e+06	.3400e+06	.3600e+06	.3000e+06
.1770e-01	.3730e+00	.1940e+02	.3170e-05			
.1600e+03	.1800e+03	.1820e+04	.5600e+04	.7400e+04	.8000e+04	.8000e+04
.2340e-05	.1030e-02	.5120e-02	.1940e-09			
.1800e+05	.2600e+05	.2440e+06	.3700e+06	.3600e+06	.3000e+06	.3000e+06
.3200e+01	.6730e+02	.3500e+04	.2130e-02			
.1900e+05	.2300e+05	.2000e+04	.2000e+04	.1000e+04	.1000e+04	.1000e+04
.1080e-01	.2230e+00	.1180e+02	.2040e-05			
.1800e+05	.2600e+05	.2040e+06	.2300e+06	.1400e+06	.1000e+06	.6000e+05
.8670e+00	.1820e+02	.9460e+03	.7500e-03			
lli wall						
.4280e+04	.4650e+040	0.	0.	0.	0.	
.7270e+02	.1480e+04	.7970e+05	.1420e+00			
.1010e+05	.1230e+05	.3920e+040	0.	0.	0.	
.2170e+03	.4540e+04	.2370e+06	.4610e+00			
.6120e-01	.6120e-010	0.	0.	0.	0.	
.1760e+00	.3680e+01	.1920e+03	.3420e-03			
.9460e-01	.9460e-010	0.	0.	0.	0.	
.5760e+01	.8110e+01	.1100e+05	.2020e-01			
.8100e+00	.8100e+000	0.	0.	0.	0.	
.1440e+02	.1460e+02	.6990e+05	.1430e+00			
.2070e+01	.2070e+010	0.	0.	0.	0.	
.9500e+02	.1110e+03	.1790e+06	.3830e+00			
.1400e+04	.3900e+040	0.	0.	0.	0.	
.6390e+01	.1190e+03	.7040e+04	.1330e-01			
.6860e+04	.6860e+040	0.	0.	0.	0.	
0.	0.	0.	0.			
.9100e+04	.1300e+05	.1300e+040	0.	0.	0.	
0.	0.	0.	0.			
.1430e+04	.1480e+040	0.	0.	0.	0.	
.6240e+02	.1470e+03	.6470e+05	.1210e+00			
.2600e+05	.2600e+050	0.	0.	0.	0.	
0.	0.	0.	0.			
.2300e+05	.2600e+050	0.	0.	0.	0.	
.2540e+00	.5120e+01	.2790e+03	.5390e-03			
.3920e+04	.4480e+040	0.	0.	0.	0.	
.5580e+02	.1200e+04	.6110e+05	.1100e+00			
.1100e+05	.1100e+050	0.	0.	0.	0.	
.1010e+03	.3920e+03	.1680e+05	.3170e-01			
.3110e+04	.3240e+040	0.	0.	0.	0.	

Table 2A-16 (Cont'd)

FILE 20 DOSE CONVERSION FILE
(NUREG/CR-2326. pg 4-2)

.5390e+02	.1060e+04	.5920e+05	.1070e+00		
.7400e+04	.7770e+040	0.	0.	0.	0.
.1560e+02	.2020e+03	.1520e+05	.2520e-01		
.6930e+01	.6930e+010	0.	0.	0.	0.
.5060e+01	.8400e+01	.8470e+04	.1580e-01		
.2000e+04	.2200e+040	0.	0.	0.	0.
.4220e+02	.8360e+03	.4630e+05	.8030e-01		
.4920e+03	.4920e+030	0.	0.	0.	0.
.3620e+02	.5430e+02	.6890e+05	.1250e+00		
.8880e+05	.9620e+05	.7400e+040	0.	0.	0.
.1570e+02	.3280e+03	.1720e+05	.3080e-01		
.4080e+03	.4080e+030	0.	0.	0.	0.
.4710e+01	.3130e+02	.5570e+04	.1010e-01		
.1790e+03	.1790e+030	0.	0.	0.	0.
.2590e+00	.5800e+00	.3770e+03	.6640e-03		
.3570e+04	.5040e+040	0.	0.	0.	0.
.3860e+00	.1260e+02	.3330e+03	.1380e-03		
.2560e+01	.2560e+010	0.	0.	0.	0.
.1330e+01	.1340e+01	.6980e+04	.9920e-02		
.1500e+05	.1670e+050	0.	0.	0.	0.
.7460e+01	.1620e+03	.4700e+04	.5060e-02		
.5350e+04	.3350e+040	0.	0.	0.	0.
.1010e+03	.6490e+03	.1160e+06	.2140e+00		
.3300e+04	.3350e+040	0.	0.	0.	0.
.1250e+03	.2140e+04	.1380e+05	.2610e-01		
.9100e+04	.9100e+040	0.	0.	0.	0.
.5260e+02	.6460e+03	.5930e+05	.1060e+00		
.4860e+03	.4860e+030	0.	0.	0.	0.
.5120e+02	.7170e+02	.9990e+05	.1850e+00		
.7260e+02	.7920e+020	0.	0.	0.	0.
.3140e+02	.5030e+03	.3490e+05	.6220e-01		
.4440e+02	.4440e+020	0.	0.	0.	0.
.6930e+02	.7600e+02	.2020e+06	.3640e+00		
.1520e+03	.1520e+030	0.	0.	0.	0.
.5030e+02	.2220e+03	.6250e+05	.1130e+00		
.1720e+02	.1720e+020	0.	0.	0.	0.
.2800e+02	.2800e+02	.1940e+06	.3610e+00		
.1080e+03	.1080e+030	0.	0.	0.	0.
.1160e+03	.2170e+03	.1710e+06	.3330e+00		
.1300e+00	.1300e+000	0.	0.	0.	0.
.3280e+01	.4610e+02	.3670e+04	.3970e-02		
.4850e+00	.4850e+000	0.	0.	0.	0.
.1200e+02	.2640e+02	.1740e+05	.3190e-01		
.2100e+04	.3970e+05	.5250e+040	0.	0.	0.
.1220e+03	.2550e+04	.1330e+06	.2410e+00		
.2320e+04	.6390e+040	0.	0.	0.	0.
.1530e+03	.2710e+04	.1690e+06	.3140e+00		

Table 2A-16 (Cont'd)

FILE 20 DOSE CONVERSION FILE
(NUREG/CR-2326, pg 4-2)

.9120e+03	.1730e+05	.2880e+040	0.	0.	0.	
.4460e+02	.9440e+03	.4920e+05	.8780e-01			
.7840e+04	.7840e+040	0.	0.	0.	0.	
.3130e+02	.2900e+04	.1920e+05	.3080e-01			
.1070e+05	.1070e+050	0.	0.	0.	0.	
.1980e+03	.1460e+04	.2320e+06	.4580e+00			
.1200e+04	.1200e+040	0.	0.	0.	0.	
.4680e+01	.9170e+02	.5140e+04	.9170e-02			
.5760e+04	.5760e+040	0.	0.	0.	0.	
.2090e+02	.1330e+03	.2480e+05	.4060e-01			
.8520e+05	.8520e+05	.7100e+040	0.	0.	0.	
.3690e+01	.7960e+02	.1350e+04	.2060e-02			
.6750e+04	.6750e+040	0.	0.	0.	0.	
0.	0.	0.	0.			
.3990e+04	.4180e+040	0.	0.	0.	0.	
.1110e+02	.1910e+03	.1240e+05	.1970e-01			
.1040e+04	.1040e+040	0.	0.	0.	0.	
.1170e+02	.1090e+03	.1340e+05	.1660e-01			
.5000e+04	.6100e+04	.2290e+05	.3400e+05	.3000e+05	.2700e+05	.2000e+05
.7960e-01	.1670e+01	.8710e+02	.1410e-04			
.4600e+04	.5600e+04	.2240e+05	.3500e+05	.3400e+05	.3300e+05	.3000e+05
.3290e-01	.6910e+00	.3600e+02	.6030e-05			
.4700e+04	.5700e+04	.2330e+05	.3500e+05	.3400e+05	.3200e+05	.3000e+05
.7110e-01	.1490e+01	.7790e+02	.1270e-04			
0.	0.	0.	0.	0.	0.	0.
.1980e-05	.8720e-03	.4330e-02	.1640e-09			
.5200e+04	.6400e+04	.2460e+05	.3700e+05	.3200e+05	.4000e+05	.3000e+05
.2710e+01	.5690e+02	.2960e+04	.1800e-02			
.5500e+04	.6100e+04	.3000e+03	.2000e+03	.1000e+03	.1000e+03	.1000e+03
.6920e-01	.1430e+01	.7590e+02	.1310e-04			
.5200e+04	.6400e+04	.2060e+05	.2300e+05	.1500e+05	.9000e+04	.6000e+04
.7270e+00	.1530e+02	.7950e+03	.6280e-03			
thyroid						
.1100e+03	.3200e+04	.1000e+030	0.	0.	0.	
.8700e+02	.1780e+04	.9640e+05	.1710e+00			
.2400e+03	.2100e+05	.3700e+05	.1000e+04	.1000e+040	0.	
.2510e+03	.5280e+04	.2730e+06	.5330e+00			
.1800e+00	.1800e+000	0.	0.	0.	0.	
.2550e+00	.5380e+01	.2800e+03	.5000e-03			
.2000e+00	.2000e+000	0.	0.	0.	0.	
.1000e+02	.1400e+02	.1890e+05	.3500e-01			
.9700e+00	.9700e+000	0.	0.	0.	0.	
.1770e+02	.1790e+02	.8580e+05	.1740e+00			
.2000e+01	.2000e+010	0.	0.	0.	0.	
.1080e+03	.1260e+03	.2020e+06	.4310e+00			
.5000e+03	.6500e+040	0.	0.	0.	0.	
.7300e+01	.1370e+03	.8060e+04	.1540e-01			

Table 2A-16 (Cont'd)

FILE 20 DOSE CONVERSION FILE
(NUREG/CR-2326. pg 4-2)

.2600e+03	.1500e+040	0.	0.	0	0.
0.	0.	0.	0.		
.1800e+03	.5900e+04	.2100e+04	.2000e+030	0.	0.
0.	0.	0.	0.		
.1300e+03	.2000e+030	0.	0.	0.	0.
.8300e+02	.1950e+03	.8320e+05	.1540e+00		
.8200e+01	.2100e+020	0.	0.	0.	0.
0.	0.	0.	0.		
.7100e+01	.3900e+030	0.	0.	0.	0.
.2930e+00	.5940e+01	.3220e+03	.6220e-03		
.7900e+02	.3500e+04	.1000e+030	0.	0.	0.
.7240e+02	.1540e+04	.7880e+05	.1430e+00		
.7700e+02	.8600e+020	0.	0.	0.	0.
.1440e+03	.5580e+03	.2110e+05	.3940e-01		
.8100e+02	.1300e+040	0.	0.	0.	0.
.6230e+02	.1220e+04	.6830e+05	.1220e+00		
.9400e+02	.1500e+030	0.	0.	0.	0.
.2420e+02	.3460e+03	.2150e+05	.3560e-01		
.4600e+02	.4600e+020	0.	0.	0.	0.
.1180e+02	.1970e+02	.1990e+05	.3720e-01		
.5200e+02	.9600e+030	0.	0.	0.	0.
.6140e+02	.1220e+04	.6710e+05	.1170e+00		
.1400e+02	.1500e+020	0.	0.	0.	0.
.5260e+02	.8010e+02	.1010e+06	.1820e+00		
.4800e+02	.3600e+04	.2700e+040	0	0.	0.
.2270e+02	.4760e+03	.2480e+05	.4470e-01		
.6400e+01	.9900e+010	0.	0.	0.	0.
.8200e+01	.5450e+02	.9640e+04	.1730e-01		
.2900e+01	.3000e+010	0.	0.	0.	0.
.3830e+00	.8600e+00	.5590e+03	.9810e-03		
.1600e+02	.2100e+030	0.	0.	0.	0.
.2660e+01	.6200e+02	.2790e+04	.1150e-02		
.8100e+00	.8100e+000	0.	0.	0.	0.
.1960e+01	.1980e+01	.1020e+05	.1470e-01		
.4300e+02	.3900e+030	0.	0.	0.	0.
.1160e+02	.2500e+03	.7620e+04	.8170e-02		
.4500e+04	.9500e+050	0.	0.	0.	0.
.1240e+03	.8100e+03	.1410e+06	.2610e+00		
.4800e+05	.9700e+050	0.	0.	0.	0.
.1680e+03	.2850e+04	.2440e+05	.4580e-01		
.1000e+03	.2200e+030	0.	0.	0.	0.
.7430e+02	.9200e+03	.8320e+05	.1500e+00		
.3700e+02	.3800e+020	0.	0.	0.	0.
.6150e+02	.8700e+02	.1200e+06	.2250e+00		
.1300e+06	.1100e+070	0.	0.	0.	0.
.4650e+02	.7370e+03	.5150e+05	.9170e-01		
.6600e+04	.6600e+040	0.	0.	0.	0.

Table 2A-16 (Cont'd)

FILE 20 DOSE CONVERSION FILE
(NUREG/CR-2326, pg 4-2)

.9000e+02	.9900e+022	.2640e+06	.4720e+00			
.1200e+06	.1800e+060	0.	0.	0.	0.	
.6970e+02	.3150e+03	.8670e+05	.1560e+00			
.1100e+04	.1100e+040	0.	0.	0.	0.	
.3330e+02	.3340e+02	.2300e+06	.4280e+00			
.4300e+05	.4400e+050	0.	0.	0.	0.	
.1380e+03	.2660e+03	.1980e+06	.3890e+00			
.3900e+00	.4000e+000	0.	0.	0.	0.	
.8900e+01	.1250e+03	.9900e+04	.1070e-01			
.9100e+00	.9100e+000	0.	0.	0.	0.	
.2050e+02	.4520e+02	.2980e+05	.5470e-01			
.5800e+03	.4300e+05	.5000e+040	0.	0.	0.	
.1590e+03	.3310e+04	.1730e+06	.3140e+00			
.6900e+03	.6000e+040	0.	0.	0.	0.	
.1860e+03	.3280e+04	.2050e+06	.3810e+00			
.3600e+03	.3100e+05	.5000e+040	0.	0.	0.	
.6540e+02	.1380e+04	.7170e+05	.1280e+00			
.2200e+03	.1200e+040	0.	0.	0.	0.	
.4270e+02	.3550e+04	.2880e+05	.4610e-01			
.1500e+03	.2300e+030	0.	0.	0.	0.	
.2370e+03	.1740e+04	.2770e+06	.5470e+00			
.6000e+01	.9200e+020	0.	0.	0.	0.	
.1130e+02	.2200e+03	.1240e+05	.2220e-01			
.1800e+02	.2500e+020	0.	0.	0.	0.	
.3290e+02	.2100e+03	.3900e+05	.6360e-01			
.5100e+01	.3500e+03	.2600e+030	0.	0.	0.	
.6060e+01	.1300e+03	.3370e+04	.5140e-02			
.9300e-01	.1000e+010	0.	0.	0.	0.	
0.	0.	0.	0.			
.1200e+02	.6900e+02	.1000e+010	0.	0.	0.	
.1910e+02	.3280e+03	.2110e+05	.3360e-01			
.8200e+01	.1500e+020	0.	0.	0.	0.	
.2270e+02	.2110e+03	.2600e+05	.3220e-01			
.1800e+02	.5900e+04	.2240e+06	.3400e+06	.2900e+06	.2400e+06	.2000e+06
.4400e-01	.9200e+00	.4820e+02	.7780e-05			
.1600e+02	.5600e+04	.2240e+06	.3500e+06	.3400e+06	.2800e+06	.3000e+06
.3200e-01	.6900e+00	.3590e+02	.6000e-05			
.1600e+02	.5600e+04	.2240e+06	.3500e+06	.3400e+06	.2800e+06	.3000e+06
.4500e-01	.9500e+00	.4910e+02	.7780e-05			
.7700e-02	.5300e+01	.1690e+04	.5700e+04	.7600e+04	.8000e+04	.8000e+04
.9900e-02	.2120e-02	.1060e-01	.4000e-09			
.3800e+02	.6500e+04	.2430e+06	.3700e+06	.3500e+06	.3300e+06	.3000e+06
.3600e+01	.1380e+03	.7270e+04	.4390e-02			
.2100e+02	.3200e+04	.2800e+04	.1300e+04	.1200e+04	.1000e+04	.5000e+03
.3200e-01	.6600e+00	.3590e+02	.6110e-05			
.2000e+02	.6600e+04	.2030e+06	.2200e+06	.1500e+06	.1000e+06	.6000e+05
.1740e+01	.3590e+02	.1890e+04	.1500e-02			

Table 2A-16 (Cont'd)

FILE 20 DOSE CONVERSION FILE
(NUREG/CR-2326. pg 4-2)

other

.2400e+03	.2900e+040	0.	0.	0.	0.
.1400e+03	.2[illegible]e+04	.1540e+06	.2750e+00		
.5800e+03	.2400e+05	.4000e+05	.2000e+04	.1000e+040	0.
.3310e+03	.6960e+04	.3630e+06	.7060e+00		
.3100e+00	.3100e+000	0.	0.	0.	0.
.2960e+00	.6230e+01	.3250e+03	.5810e-03		
.2600e+00	.2600e+000	0.	0.	0.	0.
.1260e+02	.1780e+02	.2410e+05	.4430e-01		
.1000e+01	.1000e+010	0.	0.	0.	0.
.2200e+02	.2220e+02	.1070e+06	.2180e+00		
.2300e+01	.2300e+010	0.	0.	0.	0.
.1410e+03	.1640e+03	.2640e+06	.5640e+00		
.5000e+03	.6500e+040	0.	0.	0.	0.
.1280e+02	.2380e+03	.1410e+05	.2670e-01		
.2600e+03	.1500e+040	0.	0.	0.	0.
0.	0.	0.	0.		
.1800e+03	.5900e+04	.2100e+04	.2000e+030	0.	0.
0.	0.	0.	0.		
.1500e+03	.2200e+030	0.	0.	0.	0.
.1080e+03	.2520e+03	.1120e+06	.2100e+00		
.2000e+02	.5200e+020	0.	0.	0.	0.
0.	0.	0.	0.		
.1800e+02	.9700e+03	.1000e+020	0.	0.	0.
.3210e+00	.6480e+01	.3520e+03	.6810e-03		
.1900e+03	.3700e+04	.1000e+030	0.	0.	0.
.1030e+03	.2240e+04	.1130e+06	.2040e+00		
.1700e+03	.1800e+030	0.	0.	0.	0.
.1680e+03	.6580e+03	.2690e+05	.5060e-01		
.1900e+03	.1500e+040	0.	0.	0.	0.
.1080e+03	.2120e+04	.1190e+06	.2150e+00		
.6200e+02	.1100e+030	0.	0.	0.	0.
.2930e+02	.3950e+03	.2780e+05	.4570e-01		
.8200e+01	.8200e+010	0.	0.	0.	0.
.1130e+02	.1880e+02	.1890e+05	.3530e-01		
.1200e+03	.1100e+040	0.	0.	0.	0.
.7280e+02	.1450e+04	.7810e+05	.1360e+00		
.2400e+02	.2500e+020	0.	0.	0.	0.
.6330e+02	.9670e+02	.1200e+06	.2170e+00		
.7700e+02	.3700e+04	.2800e+040	0.	0.	0.
.2660e+02	.5560e+03	.2920e+05	.5250e-01		
.1500e+02	.2100e+020	0.	0.	0.	0.
.1030e+02	.6850e+02	.1210e+05	.2190e-01		
.4900e+01	.5100e+010	0.	0.	0.	0.
.4480e+00	.1000e+01	.6480e+03	.1140e-02		
.3100e+02	.4700e+030	0.	0.	0.	0.
.5750e+01	.1260e+03	.6160e+04	.2560e-02		

Table 2A-16 (Cont'd)

FILE 20 DOSE CONVERSION FILE
(NUREG/CR-2326. pg 4-2)

.1300e+01	.1300e+010	0.	0.	0.	0.
.2520e+01	.2540e+01	.1320e+05	.1870e-01		
.7900e+02	.7400e+030	0.	0.	0.	0.
.1500e+02	.3240e+03	.9850e+04	.1060e-01		
.2200e+03	.3300e+030	0.	0.	0.	0.
.1870e+03	.1190e+04	.2120e+06	.3950e+00		
.5000e+03	.1000e+040	0.	0.	0.	0.
.2240e+03	.3810e+04	.3070e+05	.5770e-01		
.1700e+03	.3300e+030	0.	0.	0.	0.
.9240e+02	.1130e+04	.1050e+06	.1880e+00		
.4900e+02	.5000e+020	0.	0.	0.	0.
.9280e+02	.1300e+03	.1800e+06	.3370e+00		
.8500e+02	.2700e+030	0.	0.	0.	0.
.5390e+02	.8570e+03	.5970e+05	.1070e+00		
.5400e+02	.5400e+020	0.	0.	0.	0.
.1220e+03	.1330e+03	.3560e+06	.6360e+00		
.9700e+02	.1000e+030	0.	0.	0.	0.
.8370e+02	.3800e+03	.1040e+06	.1870e+00		
.2200e+02	.2200e+020	0.	0.	0.	0.
.5200e+02	.5210e+02	.3600e+06	.6720e+00		
.1000e+03	.1000e+030	0.	0.	0.	0.
.1720e+03	.3340e+03	.2490e+06	.4890e+00		
.6600e+00	.6900e+000	0.	0.	0.	0.
.1040e+02	.1460e+03	.1170e+05	.1260e-01		
.1200e+01	.1200e+010	0.	0.	0.	0.
.2570e+02	.5680e+02	.3770e+05	.6920e-01		
.5700e+03	.4100e+05	.5000e+040	0.	0.	0.
.2200e+03	.4610e+04	.2420e+06	.4370e+00		
.6700e+03	.5800e+040	0.	0.	0.	0.
.2920e+03	.5160e+04	.3220e+06	.6000e+00		
.3600e+03	.3000e+05	.6000e+040	0.	0.	0.
.7540e+02	.1610e+04	.8390e+05	.1490e+00		
.3200e+03	.1400e+040	0.	0.	0.	0.
.5060e+02	.4210e+04	.3430e+05	.5500e-01		
.3900e+03	.5800e+030	0.	0.	0.	0.
.2800e+03	.2050e+04	.3290e+06	.6470e+00		
.2200e+02	.1800e+030	0.	0.	0.	0.
.1110e+02	.2180e+03	.1230e+05	.2190e-01		
.5400e+02	.7800e+020	0.	0.	0.	0.
.4470e+02	.2850e+03	.5310e+05	.8640e-01		
.2700e+02	.1500e+04	.1900e+040	0.	0.	0.
.6910e+01	.1480e+03	.3570e+04	.5440e-02		
.1200e+01	.1300e+020	0.	0.	0.	0.
0.	0.	0.	0.		
.3900e+02	.1400e+03	.1000e+020	0.	0.	0.
.2250e+02	.3870e+03	.2490e+05	.3980e-01		
.2900e+02	.4700e+02	.1000e+01	.1000e+010.		.1000e+010.

Table 2A-16 (Cont'd)

FILE 20 DOSE CONVERSION FILE
(NUREG/CR-2326, pg 4-2)

.2710e+02	.2530e+03	.3120e+05	.3840e-01			
.1900e+03	.1900e+06	.4610e+07	.4100e+07	.3100e+07	.3000e+07	.2000e+07
.2380e+01	.5000e+02	.2610e+04	.4220e-03			
.1800e+03	.1800e+06	.4520e+07	.4300e+07	.4000e+07	.3000e+07	.3000e+07
.9040e+00	.1900e+02	.9890e+03	.1650e-03			
.1800e+03	.1800e+06	.4520e+07	.4400e+07	.3900e+07	.3000e+07	.3000e+07
.2040e+01	.4290e+02	.2230e+04	.3620e-03			
.1800e-01	.1600e+03	.3080e+05	.6300e+05	.7600e+05	.7000e+05	.6000e+05
.8010e-05	.3530e-02	.1760e-01	.6660e-09			
.2200e+03	.2000e+06	.4800e+07	.4600e+07	.3400e+07	.4000e+07	.2000e+07
.1100e+02	.2300e+03	.1200e+05	.7280e-02			
.2300e+03	.9200e+05	.7800e+05	.2000e+05	.1000e+05	.1000e+05	.1000e+05
.2170e+01	.4490e+02	.2380e+04	.4120e-03			
.2200e+03	.2100e+06	.4190e+07	.2800e+07	.1600e+07	.9000e+06	.3000e+06
.2330e+01	.4890e+02	.2550e+04	.2020e-0			
w body						
.2800e+03	.4100e+04	.1000e+030	0.	0.	0.	
.1100e+03	.2240e+04	.1200e+06	.2160e+00			
.6800e+03	.3000e+05	.4900e+05	.2000e+04	.1000e+040	0.	
.2820e+03	.5880e+04	.3070e+06	.6000e+00			
.3100e+00	.3100e+000	0.	0.	0.	0.	
.2420e+00	.5050e+01	.2660e+03	.4750e-03			
.2600e+00	.2600e+000	0.	0.	0.	0.	
.1040e+02	.1470e+02	.1980e+05	.3640e-01			
.1000e+01	.1000e+010	0.	0.	0.	0.	
.1810e+02	.1830e+02	.8760e+05	.1810e+00			
.2300e+01	.2300e+010	0.	0.	0.	0.	
.1170e+03	.1350e+03	.2180e+06	.4670e+00			
.6000e+03	.6600e+040	0.	0.	0.	0.	
.9890e+01	.1850e+03	.1090e+05	.2070e-01			
.4200e+03	.4100e+040	0.	0.	0.	0.	
0.	0.	0.	0.			
.3100e+03	.2800e+05	.1120e+06	.5000e+05	.2000e+05	.2000e+05	.1000e+05
0.	0.	0.	0.			
.2200e+03	.3100e+030	0.	0.	0.	0.	
.8730e+02	.2050e+03	.9110e+05	.1690e+00			
.4100e+03	.7800e+030	0.	0.	0.	0.	
0.	0.	0.	0.			
.3400e+03	.5600e+040	0.	0.	0.	0.	
.2940e+00	.5910e+01	.3220e+03	.6250e-03			
.2700e+03	.5500e+04	.1000e+030	0.	0.	0.	
.8180e+02	.1770e+04	.9020e+05	.1620e+00			
.4600e+03	.5200e+030	0.	0.	0.	0.	
.1390e+03	.5380e+03	.2240e+05	.[illegible]220e-01			
.2300e+03	.1900e+040	0.	0.	0.	0.	
.8340e+02	.1640e+04	.9200e+05	.1660e+00			
.2400e+03	.4200e+030	0.	0.	0.	0.	

Table 2A-16 (Cont'd)

FILE 20 DOSE CONVERSION FILE
(NUREG/CR-2326. pg 4-2)

.2380e+02	.3250e+03	.2230e+05	.3640e-01		
.9800e+01	.9800e+010	0.	0.	0.	0.
.9700e+01	.1620e+02	.1640e+05	.3060e-01		
.1900e+03	.1900e+040	0.	0.	0.	0.
.5860e+[illegible]	.1160e+04	.6450e+05	.1110e+00		
.6300e+02	.6600e+020	0.	0.	0.	0.
.5190e+02	.7940e+02	.9900e+05	.1790e+00		
.8300e+03	.4000e+05	.2200e+050	0.	0.	0.
.2180e+02	.4560e+03	.2390e+05	.4310e-01		
.6900e+02	.9600e+020	0.	0.	0.	0.
.8500e+01	.5670e+02	.9990e+04	.1820e-01		
.3300e+02	.3400e+020	0.	0.	0.	0.
.3660e+00	.8130e+00	.5280e+03	.9360e-03		
.1400e+03	.2300e+04	.1000e+030	0.	0.	0.
.2530e+01	.5840e+02	.2640e+04	.1100e-02		
.9800e+01	.9800e+010	0.	0.	0.	0.
.1970e+01	.1980e+01	.1020e+05	.1470e-01		
.3400e+03	.3000e+040	0.	0.	0.	0.
.1140e+02	.2460e+03	.7310e+04	.7830e-02		
.3500e+03	.5500e+030	0.	0.	0.	0.
.1500e+03	.9600e+03	.1700e+06	.3140e+00		
.7000e+03	.1500e+040	0.	0.	0.	0.
.1810e+03	.3080e+04	.2520e+05	.4750e-01		
.3600e+03	.7900e+030	0.	0.	0.	0.
.7510e+02	.9200e+03	.8500e+05	.1510e+00		
.1000e+03	.1100e+030	0.	0.	0.	0.
.7460e+02	.1040e+03	.1450e+06	.2680e+00		
.1500e+03	.6000e+030	0.	0.	0.	0.
.4410e+02	.7080e+03	.4910e+05	.8720e-01		
.7000e+02	.7000e+020	0.	0.	0.	0.
.9800e+02	.1070e+03	.2860e+06	.5110e+00		
.1800e+03	.2000e+030	0.	0.	0.	0.
.6870e+02	.3110e+03	.8500e+05	.1540e+00		
.3000e+02	.3000e+020	0.	0.	0.	0.
.4120e+02	.4140e+02	.2860e+06	.5330e+00		
.1500e+03	.1500e+030	0.	0.	0.	0.
.1490e+03	.2850e+03	.2150e+06	.4190e+00		
.6600e+00	.7000e+000	0.	0.	0.	0.
.7500e+01	.1050e+03	.8410e+04	.9060e-02		
.1200e+01	.1200e+010	0.	0.	0.	0.
.2120e+02	.4620e+02	.3090e+05	.5670e-01		
.6100e+03	.4100e+05	.6000e+040	0.	0.	0.
.1760e+03	.3690e+04	.1930e+06	.3500e+00		
.7100e+03	.5900e+040	0.	0.	0.	0.
.2320e+03	.4100e+04	.2570e+06	.4780e+00		
.4000e+03	.3000e+05	.6000e+040	0.	0.	0.
.6170e+02	.1310e+04	.6830e+05	.1220e+00		

Table 2A-16 (Cont'd)

FILE 20 DOSE CONVERSION FILE
(NUREG/CR-2326. pg 4-2)

.4400e+03	.1900e+040	0.	0.	0.	0.	
.4230e+02	.3650e+04	.2770e+05	.4440e-01			
.6200e+03	.9200e+030	0.	0.	0.	0.	
.2450e+03	.1800e+04	.2870e+06	.5670e+00			
.1200e+03	.1100e+040	0.	0.	0.	0.	
.9300e+01	.1820e+03	.1020e+05	.1830e-01			
.2100e+03	.3400e+030	0.	0.	0.	0.	
.3510e+02	.2240e+03	.4160e+05	.6810e-01			
.7200e+03	.3200e+05	.1500e+050	0.	0.	0.	
.5630e+01	.1200e+03	.2810e+04	.4310e-02			
.1600e+03	.8700e+030	0.	0.	0.	0.	
0.	0.	0.	0			
.1800e+03	.7600e+03	.3000e+020	0.	0.	0.	
.1770e+02	.3050e+03	.1960e+05	.3140e-01			
.1400e+03	.2400e+030	0.		.1000e+020	0.	
.2170e+02	.2020e+03	.2490e+05	.3080e-01			
.2000e+05	.2200e+07	.1580e+08	.1700e+08	.1500e+08	.1200e+08	.1100e+08
.2970e+00	.6200e+01	.3250e+03	.5250e-04			
.1800e+05	.2000e+07	.1500e+08	.1800e+08	.1700e+08	.1600e+08	.1400e+08
.1260e+00	.2630e+01	.1370e+03	.2300e-04			
.1800e+05	.2100e+07	.1590e+08	.1800e+08	.1600e+08	.1600e+08	.1500e+08
.2600e+00	.5470e+01	.2860e+03	.4640e-04			
.2900e+01	.1200e+04	.9880e+05	.2800e+06	.2700e+06	.3500e+06	.4000e+06
.4960e-05	.2210e-02	.1090e-01	.4170e-09			
.2000e+05	.2200e+07	.1680e+08	.1900e+08	.1700e+08	.1600e+08	.1500e+08
.6900e+01	.1430e+03	.7530e+04	.4560e-02			
.2200e+05	.1300e+07	.3000e+06	.1000e+06	.1000e+060.		.1000e+06
.2640e+00	.5460e+01	.2890e+03	.5000e-04			
.2100e+05	.2300e+07	.1470e+08	.1100e+08	.8000e+07	.4000e+07	.3000e+07
.1650e+01	.3460e+02	.1800e+04	.1420e-02			
testes						
.9100e+02	.3900e+03	.1000e+020	0.	0.	0.	
.1030e+03	.2090e+04	.1130e+06	.2010e+00			
.2000e+03	.2100e+04	.4000e+04	.2000e+03	.1000e+030	0.	
.2300e+03	.4800e+04	.2510e+06	.4890e+00			
.1800e+00	.1800e+000	0.	0.	0.	0.	
.2690e+00	.5660e+01	.2960e+03	.5280e-03			
.2000e+00	.2000e+000	0.	0.	0.	0.	
.1350e+02	.1910e+02	.2590e+05	.4780e-01			
.9900e+00	.9900e+000	0.	0.	0.	0.	
.1320e+02	.1340e+02	.6450e+05	.1320e+00			
.2100e+01	.2100e+010	0.	0.	0.	0.	
.7000e+02	.8100e+02	.1320e+06	.2830e+00			
.5000e+03	.6500e+040	0.	0.	0.	0.	
.8990e+01	.1660e+03	.9810e+04	.1870e-01			
.2600e+03	.1500e+040	0.	0.	0.	0.	
0.	0.	0.	0.			

Table 2A-16 (Cont'd)

FILE 20 DOSE CONVERSION FILE
(NUREG/CR-2326, Pg. 4-2)

.1800e+03	.2100e+04	.2100e+04	.2000e+030	0.	0.
0.	0.	0.	0		
.1300e+03	.2000e+030	0.	0.	0.	0.
.8790e+02	.2070e+03	.8840e+05	.1640e+00		
.8200e+01	.2100e+020	0.	0.	0.	0.
0.	0.	0.	0.		
.7100e+01	.3800e+03	.1000e+020	0.	0.	0.
.2210e+00	.4440e+01	.2420e+03	.4670e-03		
.6900e+02	.3300e+030	0.	0.	0.	0.
.8130e+02	.1760e+04	.8960e+05	.1610e+00		
.5500e+02	.6400e+020	0.	0.	0.	0.
.1500e+03	.5890e+03	.2040e+05	.3860e-01		
.7100e+02	.2500e+030	0.	[illegible]	0.	0.
.7590e+02	.1490e+04	.8330e+05	.1520e+00		
.2500e+02	.4200e+020	0.	0.	0.	0.
.2490e+02	.2320e+03	.2390e+05	.3920e-01		
.5300e+01	.5300e+010	0.	0.	0.	0.
.9110e+01	.1510e+02	.1520e+05	.2840e-01		
.5100e+02	.2400e+030	0.	0.	0.	0.
.6530e+02	.1300e+04	.7100e+05	.1240e+00		
.5000e+01	.5500e+010	0.	0.	0.	0.
.5720e+02	.8910e+02	.1090e+06	.1970e+00		
.4700e+02	.2300e+04	.1900e+040	0.	0.	0.
.2380e+02	.4970e+03	.2610e+05	.4690e-01		
.6300e+01	.9800e+010	0.	0.	0.	0.
.1110e+02	.7370e+02	.1320e+05	.2360e-01		
.2800e+01	.2900e+010	0.	0.	0.	0.
.4100e+00	.9140e+00	.5950e+03	.1050e-02		
.1600e+02	.2000e+030	0.	0.	0.	0.
.3420e+01	.7770e+02	.3600e+04	.1500e-02		
.5800e+00	.5800e+000	0.	0.	0.	0.
.2150e+01	.2170e+01	.1130e+05	.1610e-01		
.4100e+02	.3300e+030	0.	0.	0.	0.
.1260e+02	.2740e+03	.8270e+04	.8860e-02		
.8300e+02	.1200e+030	0.	0.	0.	0.
.1360e+03	.8860e+03	.1530e+06	.2850e+00		
.3100e+03	.5900e+030	0.	0.	0.	0.
.1840e+03	.3080e+04	.3250e+05	.6140e-01		
.8900e+02	.1700e+030	0.	0.	0.	0.
.8070e+02	.9950e+03	.9090e+05	.1630e+00		
.2200e+02	.2200e+020	0.	0.	0.	0.
.6760e+02	.9570e+02	.1320e+06	.2470e+00		
.5000e+02	.8600e+020	0.	0.	0.	0.
.4940e+02	.7850e+03	.5490e+05	.9750e-01		
.3700e+02	.3700e+020	0.	0.	0.	0.
.9600e+02	.1050e+03	.2800e+06	.5030e+00		
.7000e+02	.7200e+020	0.	0.	0.	0.

Table 2A-16 (Cont'd)

FILE 20 DOSE CONVERSION FILE
(NUREG/CR-2326, pg 4-2)

.7150e+02	.3220e+03	.8880e+05	.1620e+00			
.1400e+02	.1400e+020	0.	0.	0.	0.	
.3750e+02	.3760e+02	.2600e+06	.4830e+00			
.6600e+02	.6600e+020	0.	0.	0.	0.	
.1250e+03	.2520e+03	.1730e+06	.3390e+00			
.3800e+00	.3800e+000	0.	0.	0.	0.	
.7620e+01	.1080e+03	.8520e+04	.9280e-02			
.9000e+00	.9000e+000	0.	0.	0.	0.	
.2740e+02	.6040e+02	.4000e+05	.7360e-01			
.6100e+03	.5000e+05	.6000e+040	0.	0.	0.	
.1760e+03	.3670e+04	.1920e+06	.3500e+00			
.6700e+03	.6600e+040	0.	0.	0.	0.	
.2160e+03	.3820e+04	.2380e+06	.4440e+00			
.3700e+03	.3400e+05	.6000e+040	0.	0.	0.	
.6850e+02	.1460e+04	.7630e+05	.1350e+00			
.2400e+03	.1200e+040	0.	0.	0.	0.	
.4250e+02	.3110e+04	.3150e+05	.5060e-01			
.1200e+03	.1900e+030	0.	0.	0.	0.	
.1990e+03	.1460e+04	.2330e+06	.4580e+00			
.6500e+01	.1700e+020	0.	0.	0.	0.	
.8860e+01	.1730e+03	.9670e+04	.1740e-01			
.1600e+02	.2300e+020	0.	0.	0.	0.	
.4070e+02	.2590e+03	.4840e+05	.7890e-01			
.5500e+01	.1100e+03	.1600e+030	0.	0.	0.	
.4790e+01	.1030e+03	.2730e+04	.4170e-02			
.9300e-01	.1000e+010	0.	0.	0.	0.	
0.	0.	0.	0.			
.1200e+02	.2600e+020	0.	0.	0.	0.	
.1940e+02	.3340e+03	.2150e+05	.3420e-01			
.8500e+01	.1400e+020	0.		.1000e+010	0.	
.2520e+02	.2350e+03	.2900e+05	.3580e-01			
.6300e+02	.2100e+05	.8090e+06	.1170e+07	.1100e+07	.9000e+06	.8000e+06
.2360e+00	.5970e+01	.2600e+03	.4190e-04			
.5900e+02	.2000e+05	.7900e+06	.1290e+07	.1200e+07	.1100e+07	.1100e+07
.1060e+00	.2230e+01	.1160e+03	.1940e-04			
.5900e+02	.2000e+05	.8000e+06	.1280e+07	.1200e+07	.1200e+07	.1000e+07
.2080e+00	.4380e+01	[illegible]	.3700e-04			
.6200e-02	.1900e+[illegible]	[illegible]	.1970e+05	.2700e+05	.2900e+05	.2800e+05
.4380e-05	.1940e-02	[illegible]	.3650e-09			
.7000e+02	.2300e+05	.8570e+06	.1320e+07	.1300e+07	.1200e+07	.1000e+07
.6000e+01	.1260e+03	.6610e+04	.3980e-02			
.7500e+02	.1200e+05	.1000e+05	.4000e+04	.4000e+04	.4000e+04	.3000e+04
.2100e+00	.4350e+01	.2300e+03	.3990e-04			
.7100e+02	.2400e+05	.7260e+06	.8500e+06	.5000e+06	.3000e+06	.3000e+06
.1470e+01	.3080e+02	.1610e+04	.1270e-02			

Table 2A-16 (Cont'd)

FILE 20 DOSE CONVERSION FILE
(NUREG/CR-2326, pg 4-2)

ovaries

.9700e+03	.2300e+040	0.	0.	0.	0.
.6370e+02	.1290e+04	.7010e+05	.1220e+00		
.2100e+04	.8500e+04	.8500e+04	.1000e+040	0.	0.
.2240e+03	.4700e+04	.2440e+06	.4780e+00		
.1800e+00	.1800e+000	0.	0.	0.	0.
.8900e-01	.1860e+01	.9720e+02	.1720e-03		
.2300e+00	.2300e+000	0.	0.	0.	0.
.5700e+01	.8020e+01	.1080e+05	.1980e-01		
.1000e+01	.1000e+010	0.	0.	0.	0.
.1490e+02	.1510e+02	.7220e+05	.1490e+00		
.2300e+01	.2300e+010	0.	0.	0.	0.
.1140e+03	.5340e+03	.2160e+06	.4640e+00		
.5000e+03	.6500e+040	0.	0.	0.	0.
.6090e+01	.1130e+03	.6660e+04	.1260e-01		
.2600e+03	.1500e+040	0.	0.	0.	0.
0.	0.	0.	0.		
.1800e+03	.5900e+04	.2100e+04	.2000e+030	0.	0.
0.	0.	0.	0.		
.2300e+03	.3000e+030	0.	0.	0.	0.
.4460e+02	.1020e+03	.5110e+05	.9440e-01		
.8200e+01	.2100e+020	0.	0.	0.	0.
0.	0.	0.	0.		
.9500e+01	.3900e+03	.1000e+020	0.	0.	0.
.2740e+00	.5480e+01	.3000e+03	.5830e-03		
.7800e+03	.2300e+040	0.	0.	0.	0.
.4120e+02	.9000e+03	.4530e+05	.8170e-01		
.6100e+03	.6900e+030	0.	0.	0.	0.
.5600e+02	.2180e+03	.1490e+05	.2780e-01		
.7800e+03	.1600e+040	0.	0.	0.	0.
.5090e+02	.1010e+04	.5620e+05	.1020e+00		
.2100e+03	.3400e+030	0.	0.	0.	0.
.1030e+02	.1410e+03	.9720e+04	.1580e-01		
.8600e+01	.8600e+010	0.	0.	0.	0.
.4150e+01	.6800e+01	.6920e+04	.1290e-01		
.5300e+03	.1200e+040	0.	0.	0.	0.
.2150e+02	.4200e+03	.2350e+05	.4080e-01		
.5600e+02	.5900e+020	0.	0.	0.	0.
.2100e+02	.3280e+02	.3940e+05	.7190e-01		
.2400e+03	.2800e+04	.2000e+040	0.	0.	0.
.8500e+01	.1770e+03	.9290e+04	.1670e-01		
.5900e+02	.7900e+020	0.	0.	0.	0.
.4640e+01	.3080e+02	.5[illegible]0e+04	.9860e-02		
.3400e+01	.3500e+010	0.	0.	0.	0.
.1380e+00	.3120e+00	.2010e+03	.3530e-03		
.3400e+02	.2500e+030	0.	0.	0.	0.
.9100e+00	.2100e+02	.9460e+03	.3920e-03		

Table 2A-16 (Cont'd)

FILE 20 DOSE CONVERSION FILE
(NUREG/CR-2326, pg 4-2)

.7900e+00	.7900e+000	0.	0.	0.	0.
.8400e+00	.8400e+00	.4380e+04	.6190e-02		
.1000e+03	.4600e+030	0.	0.	0.	0.
.4500e+01	.9800e+02	.2700e+04	.2940e-02		
.5800e+03	.7600e+030	0.	0.	0.	0.
.9800e+02	.6030e+03	.1120e+06	.2090e+00		
.5400e+03	.9700e+030	0.	0.	0.	0.
.9900e+02	.1660e+04	.1380e+05	.2580e-01		
.5300e+03	.8800e+030	0.	0.	0.	0.
.3150e+02	.3920e+03	.3590e+05	.6440e-01		
.7800e+02	.7900e+020	0.	0.	0.	0.
.4570e+02	.6450e+02	.8940e+05	.1670e+00		
.6300e+02	.1000e+030	0.	0.	0.	0.
.1640e+02	.2640e+03	.1840e+05	.3250e-01		
.4300e+02	.4300e+020	0.	0.	0.	0.
.5340e+02	.5860e+02	.1550e+06	.2780e+00		
.7700e+02	.8000e+020	0.	0.	0.	0.
.3230e+02	.1440e+03	.4040e+05	.7250e-01		
.1700e+02	.1700e+020	0.	0.	0.	0.
.2560e+02	.2570e+02	.1780e+06	.3310e+00		
.8300e+02	.8300e+020	0.	0.	0.	0.
.1160e+03	.2170e+03	.1770e+06	.3470e+00		
.4200e+00	.4300e+000	0.	0.	0.	0.
.3080e+01	.4360e+02	.3450e+04	.3720e-02		
.1000e+01	.1000e+010	0.	0.	0.	0.
.1130e+02	.2460e+02	.1650e+05	.3000e-01		
.5800e+03	.4400e+05	.5000e+040	0.	0.	0.
.8900e+02	.1870e+04	.9810e+05	.1750e+00		
.6500e+03	.5900e+040	0.	0.	0.	0.
.1490e+03	.2640e+04	.1650e+06	.3060e+00		
.3600e+03	.3100e+05	.6000e+040	0.	0.	0.
.2270e+02	.4800e+03	.2470e+05	.4440e-01		
.6000e+03	.1900e+040	0.	0.	0.	0.
.2400e+02	.2830e+04	.1080e+05	.1750e-01		
.1100e+04	.1600e+040	0.	0.	0.	0.
.2030e+03	.1490e+04	2370e+06	.4690e+00		
.1100e+03	.2000e+030	0.	0.	0.	0.
.3920e+01	.7700e+02	.4300e+04	.7670e-02		
.2300e+03	.3000e+030	0.	0.	0.	0.
.1800e+02	.1150e+03	.2140e+05	.3500e-01		
.6000e+02	.2400e+03	.1800e+030	0.	0.	0.
.3540e+01	.7610e+02	.1170e+04	.1790e-02		
.9300e-01	.1000e+010	0.	0.	0.	0.
0.	0.	0.	0.		
.1800e+03	.3100e+030	0.	0.	0.	0.
.7100e+01	.1200e+03	.7710e+04	.1250e-01		
.1300e+03	.1900e+030	0.	0.	0.	0.

Table 2A-16 (Cont'd)

FILE 20 DOSE CONVERSION FILE
(NUREG/CR-2326, pg 4-2)

.1080e+02	.1000e+03	.1240e+05	.1540e-01			
.2200e+02	.6800e+04	.2630e+06	.3800e+06	.3400e+06	.3100e+06	.2000e+06
.3320e-01	.7020e+00	.3640e+02	.5920e-05			
.1900e+02	.6400e+04	.2540e+06	.4000e+06	.3400e+06	.4000e+06	.3000e+06
.2020e-01	.4250e+00	.2220e+02	.3690e-05			
.2100e+02	.6400e+04	.2540e+06	.4100e+06	.4300e+06	.3000e+06	.4000e+06
.3220e-01	.6770e+00	.3510e+02	.5720e-05			
.2200e-02	.6000e+01	.1990e+04	.6400e+04	.8600e+04	.9000e+04	.9000e+04
.2970e-05	.1300e-02	.6460e-02	.2460e-09			
.7300e+02	.7400e+04	.2730e+06	.4300e+06	.3900e+06	.4000e+06	.3000e+06
.4060e+01	.8520e+02	.4450e+04	.2690e-02			
.2600e+02	.3700e+04	.3200e+04	.1500e+04	.1300e+04	.1300e+04	.1000e+04
.2690e-01	.5590e+00	.2950e+02	.5110e-05			
.2500e+02	.7500e+04	.2320e+06	.2600e+06	.1700e+06	.1100e+06	.6000e+05
.8900e+00	.1870e+02	.9720e+03	.7720e-03			

Table 2A-17

FILE 27 METEOROLGICAL DATA
(NUREG/CR-2326, pg 4-6)

To be supplied by utility for specific site.

SECTION 3.2

CONTENTS

TABLES

TABLE 3.2-1

CLASSIFICATION SUMMARY

The classification information is presented by System * in the following order:**

Table 3.2-1 Item No.	MPL Number**	Title
B	Nuclear Boiler Supply System	
B1	B11	Reactor Pressure Vessel System*
B2	B21	Nuclear Boiler System*
B3	B31	Reactor Recirculation System
C	Control and Instrument Systems	
C1	C11	Rod Control and Information System
C2	C12	Control Rod Drive System
C3	C31	Feedwater Control System
C4	C41	Standby Liquid Control System
C5	C51	Neutron Monitoring System*
C6	C61	Remote Shutdown System
C7	C71	Reactor Protection System*
C8	C81	Recirculation Flow Control System
C9	C82	Automatic Power Regulator System
C10	C85	Steam Bypass and Pressure Control System
C11	C91	Process Computer (Includes PMCS and PGCS)
C12	C93	Refueling Platform Control Computer
C13	C94	CRD Removal Machine Control Computer
D	Radiation Monitoring Systems	
D1	D11	Process Radiation Monitoring* System
D2	D21	Area Radiation Monitoring System
D3	D23	Containment Atmospheric Monitoring System*
E	Core Cooling Systems	
E1	E11	Residual Heat Removal System*
E2	E22	High Pressure Core Flooder System*

* *These systems or subsystems thereof, have a primary function that is safety-related. As shown in the balance of this Table, some of these systems contain non-safety related components and, conversely, some systems whose primary functions are non-safety related contain components that have been designated safety-related.*

** *Master Parts List Number designated for the system*

*** *Only those systems that are in the ABWR Standard Plant scope are included in this table.*

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Table 3.2-1 Item No.	MPL Number**	Title
E		Core Cooling Systems (Continued)
E3	E31	Leak Detection and Isolation System*
E4	E51	Reactor Core Isolation Cooling System*
F		Reactor Servicing Equipment
F1	F11	Fuel Servicing Equipment
F2	F12	Miscellaneous Servicing Equipment
F3	F13	RPV Servicing Equipment
F4	F14	RPV Internal Servicing EQuipment
F5	F15	Refueling Equipment
F6	F16	Fuel Storage Facility
F7	F17	Under-Vessel Servicing Equipment
F8	F21	CRD Naintenance Facility
F9	F22	Internal Pump Maintenance Facility
F10	F32	Fuel Cask Cleaning Facility
F11	F41	Plant Start-up Test Facility
F12	F51	Inservice Inspection Equipment
G		Reactor Auxiliary Systems
G1	G31	Reactor Water Cleanup System
G2	G41	Fuel Pool Cooling and Cleanup System
G3	G51	Suppression Pool Cleanup System
H		Control Panels
H1	H11	Main Control Room Panels*
H2	H12	Control Room Back Panels*
H3	H14	Radioactive Waste Control Panels
H4	H21	Local Control Panels*
H5	H22	Instrument Racks
H6	H23	Mutiplexing System
H7	H25	Local Control Boxes

* *These systems or subsystems thereof, have a primary function that is safety-related. As shown in the balance of this Table, some of these systems contain non-safety related components and, conversely, some systems whose primary functions are non-safety related contain components that have been designated safety-related.*

** *Master Parts List Number designated for the system*

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Table 3.2-1 Item No.	MPL Number**	Title
J	Nuclear Fuel	
J1	J11	Nuclear Fuel
J2	J12	Fuel Channel
K	Radioactive Waste System	
K1	K17	Radwaste Sys'em
N	Power Cycle Systems	
N1	N11	Turbine Main Steam System
N2	N21	Condensate, Feedwater and Condensate Air Extraction System
N3	N22	Heater, Drain and Vent System
N4	N25	Condensate Purification System
N5	N26	Condensate Filter Facility
N6	N27	Condensate Demineralizer
N7	N31	Main Turbine
N8	N32	Turbine Control Syste
N9	n33	Turbine Glan l Steam System
N10	N34	Turbine Lubricating Oil System
N11	N35	Moisture Separator Heater
N12	N36	Extraction System
N13	N37	Turbine Bypass System
N14	N38	Reactor Feedwater Pump Driver
N15	N39	Turbine Auxiliary STeam System
N16	N41	Generator
N17	N42	Hydrogen Gas Cooling System
N18	N43	Generator Cooling System
N19	N44	Generator Sealing Oil System
N20	N51	Exciter
N21	N61	Main Condenser
N22	N62	Offgas System
N23	N71	Circulating Water System
N24	N72	Cindenser Cleanup System
P	Station Auxilairy Systems	
P1	P11	Make Water System (Purified)
P2	P13	Makeup Water System (Condensate)

* *These systems or subsystems thereof, have a primary function that is safety-related. As shown in the balance of this Table, some of these systems contain non-safety related components and, conversely, some systems whose primary functions are non-safety related contain components that have been designated safety-related.*

** *Master Parts List Number designated for the system*

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Table 3.2-1 Item No.	MPL Number**	Title
P	Station Auxiliary Systems (Continued)	
P3	P21	Reactor Building Cooling Water System*
P4	P22	Turbine Building Cooling Water System
P5	P24	HVAC Normal Cooling Water System
P6	P25	HVAC Emergency Cooling Water System
P7	P32	Oxygen Injection System
P8	P40	Ultimate Heat Sink
P9	P41	Reactor Service Water System
P10	P42	Turbine Service Water System
P11	P51	Station Instrument Air System
P12	P52	Instrument Air System
P13	P54	High Pressure Nitrogen Gas Supply System
P14	P61	Heating Steam and Condensate Water Return System
P15	P62	House Boiler
P16	P63	Hot Water Heating System
P17	P73	Hydrogen Water Chemistry System
P18	P74	Zinc Injection System
P19	P81	Breathing Air System
P20	P91	Sampling System Includes PASS)
P21	P92	Freeze Protection System
P22	P95	Iron Injection System
R	Station Electrical Systems	
R1	R10	Electrical Power Distribution System
R2	R11	Unit Auxiliary Transformer
R3	R13	Isolated Phase Bus
R4	R21	Non-Segreated Phase Bus
R5	R22	Metalclad Switchgear
R6	R23	POwer Center
R7	R24	Motor Control Center
R8	R31	Raceway System
R9	R34	Grounding Wire
R10	R35	Electrical Wiring Penetration

* *These systems or subsystems thereof, have a primary function that is safety-related. As shown in the balance of this Table, some of these systems contain non-safety related components and, conversely, some systems whose primary functions are non-safety related contain components that have been designated safety-related.*

** *Master Parts List Number designated for the system*

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Table 3.2-1 Item No.	MPL Number**	Title
R		Station Electrical Systems (Continued)
R11	R40	Combustion Turbine Gnerera:or
R12	R42	Direct Current Power Supply*
R13	R43	Emeregucy Diesel Generator System*
R14	R46	Vital AC Power Supply
R15	R47	Instrument and Controol Power upply
R16	R51	Communication System
R17	R52	Lighting and Servicing Power Supply
S		Power Transmission Systems
S1	S12	Resreve Transformer
T		Containment and Environmental Control Systems
T1	T11	Primary Containment System
T2	T12	Containment Internal Structures
T3	T13	Reactor Pressure Vessel Pedestal
T4	T22	Standby Gas Treatment System*
T5	T25	PCV Pressure and Leak Testing Facility
T6	T31	Atmospheric Control System
T7	T41	Drywell Cooling System
T8	T49	Flammability Control System
T9	T53	Suppression Pool Temperature Monitoring System*
U		Structures and Servicing Systems
U1	U21	Foundation Work
U2	U24	Turbine Pedestal
U3	U31	Cranes and Hoists
U4	U32	Elevator
U5	U41	Heating, Ventilating and Air Conditioning*
U6	U43	Fire Protection System
U7	U46	Floor Leakage Detection System
U8	U47	Vacuum Sweep System

* *These systems or subsystems thereof, have a primary function that is safety-related. As shown in the balance of this Table, some of these systems contain non-safety related components and, conversely, some systems whose primary functions are non-safety related contain components that have been designated safety-related.*

** *Master Parts List Number designated for the system*

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Table 3.2-1 Item No.	MPL Number**	Title	Table 3.2-1 Item No.	MPL Number**	Title
U	Structures and Servicing Systems (Continued)				
U9	U48	Decontamination System			
U10	U71	Reactor Building*			
U11	U72	Turbine Building*			
U12	U73	Control Building*			
U13	U74	Radwaste Building			
U14	U75	Service Building			
Y	Yard Structures and Equipment				
Y1	Y31	Stack			
Y2	Y52	Oil Storage and Transfer System			
Y3	Y86	Site Security			

* *These systems or subsystems thereof, have a primary function that is safety-related. As shown in the balance of this Table, some of these systems contain non-safety related components and, conversely, some systems whose primary functions are non-safety related contain components that have been designated safety-related.*

** *Master Parts List Number designated for the system*

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
B1 Reactor Pressure Vessel System						
1. Reactor pressure vessel (RPV)	1	C	A	B	I	210.20
2. Reactor vessel support skirt and stabilizer	1	C	A	B	I	
3. RPV appurtances-reactor coolant pressure boundary boundary portions (RCPB)	1	C	A	B	I	
4. Supports for CRD housing and in-core housing	1	C	A	B	I	
5. Reactor internal structure, ECCS spargers, feedwater, RHR shutdown cooling, and pressure core flooder spargers	2	C	B	B	I	
6. Reactor internal structures-safety related components (except ECCS spargers) including core support structures (See Subsection 3.9.5)	3	C	---	B	I	
7. Reactor internal structures-non-safety related components (See Subsection 3.9.5)	N	C	---	E	---	260.4
8. Control Rods	3	C,SC	---	B	I	
9. Deleted						
10. Deleted						
11. Reactor Internal Pump Motor Casing (a part of RPV boundary)	1	C	A	B	I	
B2 Nuclear Boiler System						
1. Vessels - level instrumentation condensing chambers	1	C	A	B	I	

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
B2 **Nuclear Boiler System** (Continued)						
2. Vessel-nitrogen accumulatros (for ADS and SRVs)	3/N	C	C	B	I	
3. Piping including supports-safety/releif valve discharge	2/3	C	B/C	B	I	(h)

210.20

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

	Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes	
B2	**Nuclear Boiler System** (Continued)							
	4. Piping including supports main steamline (MSL) and feedwater (FW) line up to and including the outermost isolation valve	1	C,SC	A	B	I		
210.20	5. Piping including supports-MSL from outermost isolation valve to and including seismic interface restraint and FW from outermost isolation valve to the shutoff valve	2	SC	B	B	I		
	6. Piping including supports-MSL from the seismic interface restraint to the turbine stop valve	N	SC,T	B	B	---	(r)	210.6 210.7 260.4
	7. Piping from FW shutoff valve to seismic interface restraint	3	SC	C	B	I		260.4
	8. Deleted							
	9. Deleted							
	10. Pipe whip restraint - MSL/FW if needed	3	SC,C	---	B	---	(dd)	
210.20	11. Piping including supports-other within outermost isolation valves							
	a. RPV head vent	1	C	A	B	I	(g)	
	b. Main steam drains	1	C,SC	A	B	I	(g)	
210.20	12. Piping including supports-other beyond outermost isolation or shutoff valves							
	a. RPV head vent beyond shutoff valves	N	C	C	E	---		260.4
	b. Main steam drains	2/N	SC	B/C	B	I/---		

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
B2 Nuclear Boiler System (Continued)						
13. Piping including supports-instrumentation up to and beyond outermost isolation valves	2/N	C,SC	B/D	B/E	I/---	(g)
14. Safety/relief valves	1	C	A	B	I	
15. Valves - MSL and FW isolation valves, and other FW valves within containment	1	C,SC	A	B	I	
16. Valves - FW, other beyond outermost isolation valves up to and including shutoff valves	2	SC	B	B	I	
17. Valves - within outermost isolation valves						
a. RPV head vent	1	C	A	B	I	(g)
b. Main steam drains	1	C,SC	A	B	I	(g)
18. Valves, other						
a. RPV head vent	3	C	C	B	I	
b. Main steam drain	N	SC	C	B	---	
19. Deleted						
20. Mechanical modules-instrumentation with safety-related function	3	C,SC	---	B	I	
21. Electrical modules with safety-related function	3	C,SC,X	---	B	I	(i)
22. Cable with safety-related function	3	C,SC,X	---	B	I	

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

	Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes	
	B3 Reactor Recirculation System							
	1. Piping-Purge System, heat exchanger and primary side of recirculation motor cooling system (RMCS)	3	C	C	B	I	(s)	210.8 210.19
	2. Pipe Supports	3	C	C	B	I		
	3. Pump motor cover and hardware	2	C	B	B	I		
	4. Pump non-pressure retaining parts including motor, instruments, electrical cables and seals	N	C	---	E	---		260.4
	5. Valves	3	C	C	B	I	(g)	
	6. ATWS equipment associated with the pump trip function	N	C	---	E	---	(cc)	260.4
	C1 Rod Control and Information System							
	1. Electrical Modules	N	RZ,X	D	E	---		
	2. Cable	N	SC,RZ,X	D	E	---		
	C2 CRD System							
	1. Valves with no safety related function (not part of HCU)	N	SC	D	E	---		
210.20	2. Piping including supports-insert line	2	C,SC	B	B	I	(j)	
	3. Piping-other (pump suction, pump discharge, drive header)	N	SC	D	E	---	(g)	260.4
	4. Hydraulic control unit	2	SC	---	B	I	(k)	
	5. Fine motion drive motor	N	C	---	E	---		260.4

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
C2 CRD System (Continued)						
6. CRD Drive water pumps	N	SC	D	E	---	
7. Control Rod Drive	1/3	C	A/---	B	I	
8. Electrical modules with safety function	3	C,SC	---	B	I	
9. Cable with safety-related	3	C,SC,X	---	B	I	
10. ATWS Equipment associated with the Alternate Rod Insert (ARI) functions	N	C	---	E	---	(cc)
C3 **Feedwater Control System**	N	C,T,SC, X	---	E	---	

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

	Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
	C4 Standby Liquid Control System						
210.20	1. Standby liquid control tank including supports	2	SC	B	B	I	(u)
	2. Pump including supports	2	SC	B	B	I	(u)
	3. Pump motor	2	SC	---	B	I	(u)
	4. Valves - injection	1	SC	A	B	I	(u)
	5. Valves within injection valves	1	C,SC	A	B	I	(u)
	6. Valves beyond injection valves	2	SC	B	B	I	(g,u)
210.20	7. Piping including supports within injection valves	1	C,SC	A	B	I	(g,u)
	8. Piping including supports beyond injection valves	2	SC	B	B	I	(g,u)
260.4	9. Electrical equipment and devices	3/N	SC,X	---	B/E	I/---	(cc)
	10. Cable	3/N	SC,X	---	B/E	I/	(cc)
	C5 Neutron Monitoring System						
	1. Electrical modules - SRNM, LPRM and LPRM	3	SC,X	---	B	I	
	2. Cable - SRNM and LPRM	3	C,SC,X RZ	---	B	I	
	3. Detector and tube assembly	2/3	C	B/C	B	I	

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
C6 **Remote Shutdown System**						
Components of this system are included under B2, E1, E4, G3, H4, and P2.						
1. Electrical modules safety-related function	3	C,SC,RZ, X	---	B	I	
2. Cable with safety related function	3	RZ	---	B	I	
C7 **Reactor Protection System**						
1. Electrical modules with safety-related function	3	SC,X,T, RZ	---	B	I	
2. Cable with safety related function	3	SC,X,T, RZ	---	B	I	
3. Deleted						
4. Deleted						
C8 **Recirculation Flow Control System**	N	X	---	E	---	
C9 **Automatic Power Regulator System**	N	X	---	E	---	
C10 **Steam Bypass and Pressure Control System**	N	X	---	E	---	
C11 **Process Computer (includes PMCS & PGCS)**	N	X	---	E	---	

260.4

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
C12 **Refueling Platform Control Computer**	N	SC	---	E	---	
C13 **CRD Removal Machine Control Computer**	N	SC	---	E	---	
D1 **Process Radiation Monitoring System (includes gaseous and liquid affluent monitoring)**						
1. Electrical modules - with with safey-related functions (including monitors)	3	SC,X,RZ	---	B	I	
2. Cable with safety-related functions	3	SC,X,RZ	---	B	I	
3. Electrical Modules, other	N	T,SC,RZ, X,W	---	E	---	(u)
4. Cables, other	N	T,SC,RZ, X,W	---	E	---	(u)
D2 **Area Radiation Monitoring System**	N	X	---	E	---	

260.4

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
D3 Containment Atmospheric Monitoring System						
1. Component with safety-related function	3	C,SC,X	---	B	I	
E1 RHR System						
1. Heat exchangers-primary side	2	SC	B	B	I	
2. Deleted						
3. Piping including supports* within outermost isolation valves	1	C,SC	A	B	I	(g)
4. Containment spray piping including supports and spargers, within and including the outermost isolation valves	2	C,SC	B	B	I	
4a. Piping including supports beyond outermost isolation valves	2	SC	B	B	I	(g)
5. Main Pumps including supports	2	SC	B	B	I	
6. Main Pump motors	2	SC	B	B	I	
7. Valves - isolation, (LPFL line) including shutdown suction line isolation valves	1	C,SC	A	B	I	(g)
8. Valves - isolation, other (pool suction valves and pool test return valves)	2	SC	B	B	I	(g)
9. Valves beyond isolation valves	2	SC	B	B	I	(g)

210.20

** The RHR/ECCS low pressure flooder spargers are part of the reactor pressure vessel system, see Item B1.5.*

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
F1 RHR System (Continued)						
210.20 10. Jockey pumps and motors including supports	2	SC	B	B	I	

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
E2 High Pressure Core Flooder System						
1. Reactor pressure vessel injection line and connected piping including supports within outermost isolation valve*	1/2	C,SC	A/B	B	I	(g)
2. All other piping including supports**	2/3	SC,O	B/C	B	I	(g)
3. Main Pump	2	SC	B	B	I	
4. Main Pump Motor	3	SC	---	B	I	
5. Valves - other isolation and within the reactor pressure vessel injection line and connected lines	1	C,SC	A	B	I	(g)
6. All other valves 2/3	SC	B/C	B	I	(g)	
7. Electrical modules with safety-related functions	3	C,SC,X	---	B	I	
8. Cable with safety-related function	3	C,SC,X	---	B	I	
E3 Leak Detection and Isolation System						
1. Temperature sensors	3/N	C,SC,T	---	B/E	I/---	(z)
2. Pressure transmitters	3	C,SC	---	B	I/---	(z)
3. Differential pressure transmitters (flow)	3	C,SC	---	B	I/---	(z)

* *The ECCS high pressure core flooder spargers are part of the Reactor Pressure Vessel System, see Item B1.5.*

** *Pool suction piping, suction piping from condensate storage tank, test line to pool, pump discharge piping and return line to pool.*

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

	Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
	E3 Leak Detection and Isolation System (Continued)						
260.4	4. Fission Product M	N	SC	---	E	I	
260.4	5. Isolation Valves	2/N	SC	B/C	B/E	I	
260.4	6. Instrument lines	3	C,SC	B	B	I	
260.4, 430.223	7. Sample lines*	2/N	C,SC	C/D/---	B/E	I/---	
260.4	8. Flow transmitters	N	SC	---	E	---	
260.4	9. Electrical modules	3/N	SC,RZ,X	---	B/E	I/---	
260.4	10. Cables	3/N	SC,RZ,X	---	B/E	I/---	
	E4 RCIC System						
210.20	1. Piping including supports within outermost isolation valves	1/2	C,SC	A/B	B	I	(g)
260.4	2. Piping including supports - discharge line from vacuum pump to containment isolation valves, and discharge line from condensate pump to the first globe valve	N	SC	D	E	---	(g)
210.20	3. Piping including supports beyond outermost isolation valves up to the turbine exhaust line to the suppression pool, including turbine inlet and outlet drain lines	2/3	C,SC	B/C	B	I	(g)

* *These sample lines are totally within containment and the fission product monitor provides no isolation function.*

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

	Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
	E4 RCIC System (Continued)						
260.4	4. RCIC Pump and piping including support, CST suction line from the first RCIC motorized valve, S/P suction line to the pump, discharge line up to the FW line "B" thermal sleeve	2	SC	B	B	I	(g)
260.4	5. Pump motors	N	SC	---	E	I	
	6. Valves - outer isolation and within	1/2	C,SC	A/B	B	I	(g)
260.4	7. Valves - outside the PCV*	2	SC	B	B	I	(g)
	8. Valves - beyond turbine inlet second shutoff	3	SC	C	B	I	(g)
260.4	9. Turbine including supports	2	SC	---	B	I	(m)
	10. Electrical modules with safety-related function	3	C,SC,X	---	B	I	
	11. Cable with safety function	3	C,SC,X	---	B	I	
	12. Other mechanical and electrical modules	N	SC,X	---	E	---	
260.4	**F1 Fuel Servicing Equipment**	N	SC	---	E	---	
	F2 Miscellaneous Servicing Equipment	N	SC,RZ	---	E	---	

* *Except item 8.*

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
F3 RPV Servicing Equipment	N	SC	---	E	---	
F4 RPV Internal Servicing Equipment	N	SC	---	E	---	
F5 Refueling Equipment						
1. Refueling equipment platform assembly	N	SC	---	E	I	(bb)
2. Refueling bellows	N	SC	---	E	---	
F6 Fuel Storage Equipment						
1. Fuel storage racks - new and spent	N	SC	---	E	I	(bb)
2. Defective fuel storage	N	SC	---	E	---	(bb)
F7 Under-Vessel Servicing Equipment	N	SC	---	E	---	(bb)
F8 CRD Maintenance Facility	N	SC	---	E	---	
F9 Internal Pump Maintenance Facility	N	SC	---	E	---	
F10 Fuel Cask Cleaning Facility	N	SC	---	E	---	
F11 Plant Start-up Test Equipment	N	M	---	E	---	

260.4

210.13
430.195a

210.15

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
F12 Inservice Inspection Equipment	N	M	---	E	---	
G1 Reactor Water Cleanup System						
1. Vessels including supports (filter/deminrealizer)	N	SC	C	E	---	
2. Regenerative heat exchangers including supports carrying reactor water	N	SC	C	E	---	
3. Cleanup recirculation pump, motors	N	SC	C	E	---	

260.4
210.29

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
G1 Reactor Water Cleanup System (Continued)						
4. Piping including supports and valves within and including outermost containment isolation valves on pump suction	1	C,SC	A	B	I	(g)
5. Pump suction and discharge piping including supports and valves from containment isolation valves back to shut-off valves at feedwater line connections	N	SC	C	E	---	(g)
6. Piping including supports and valves from feedwater lines to and including shut-off valves	2	SC	B	B	I	(g)
7. Piping including supports and valves to main condenser	N	SC,T	C	E	---	(g)
8. Non-regenerative heat exchanger tube inside and piping including supports and valves carrying process water	N	SC	C	E	---	(g)
9. Non-regenerative heat exchanger shell and piping including supports carrying closed cooling water	N	SC	D	E	---	
10. Filter/demineralizer precoat subsystem	N	SC	D	E	---	
11. Filter demin holding pumps including supports - valves and piping including supports	N	SC	C	E	---	

260.4

210.16

260.4

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
G1 **Reactor Water Cleanup System** (Continued)						
12. Sample station	N	SC	D	E	---	
13. Electrical modules and cable with no safety-related function	N	SC,X	D	E	---	
14. Electrical modules and cable for isolation valves	3	SC	---	B	I	

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
G2 Fuel Pool Cooling and Cleanup System						
1. Vessels including supports - filter/demineralizers	N	SC	C	E	---	
2. Vessels including supports - drain tanks	N	SC	C	E	---	
3. Heat exchangers including supports	N	SC	C	E	---	
4. Pumps including support and pump motors	N	SC	C	E	---	
5. Piping including supports and valves	N	SC	C	E	---	
6. Normal Makeup system components	N	SC,O,T	C	E	---	
7. RHR connections for safety-related makeup	3	SC	C	B	I	
8. Electrical modules and cables with no safety-related function	N	SC,X	---	E	---	
G3 Suppression Pool Cleanup System						
1. Isolation valves and piping including supports within outermost isolation valves	2	C	B	B	I	
2. Pumps	N	SC	D	E	---	
3. Pump motors	N	SC	---	E	---	
4. Other piping including supports	N	SC	D	E	---	
5. Electrical modules and Cables	N	SC,X	---	E	---	

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
H1 Main Control Room Panels						
260.4 1. Panels	3/N	X	---	B/E	I/---	(aa)
2. Electrical Modules with safety-related function	3	X	---	B	I	
3. Cable with safety-related function	3	X	---	B	I	
260.4 4. Other mechanical and electrical modules	N	X	---	E	---	
H2 Control Room Back Panels						
260.4 1. Panels	3/N	C,SC,X	---	B/E	I/---	(aa)
2. Electrical modules with safety-related function	3	C,SC,X	---	B	I	
3. Cable with safety-related function	3	C,SC,X	---	B	I	
260.4 4. Other mechanical and electrical modules	N	C,SC,X	---	E	---	
H3 Radioactive Waste Control Panels	N	W	---	E	---	(p)
H4 Local Control Panels						
260.4 1. Panels and Racks	3/N	RZ,SC,X	---	B/E	I/---	(aa)
2. Electrical modules with safety-related function	3	RZ,SC,X	---	B	I	

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
H4 Local Control Panels (Continued)						
3. Cable with safety-related function	3	RZ,SC,X	---	B	I	
4. Other mechanical and electrical modules	N	RZ,SC,X	---	E	---	268-4
H5 Instrument Racks						
1. Mechanical and electrical with safety-related function	3	SC,RZ,X Y,W,M	---	B	---	
2. Other mechanical and selected modules	N	SC,RZ,X, T	---	E	---	
H6 Multiplexing System						
1. Electrical module with safety-related function (Essential)	3	SC,RZ,X	---	B	I	
2. Cable with safety-related function (Essential)	3	SC,RZ,X	---	B	I	
3. Other electrical modules and cables (Non-essential)	N	SC,RZ,X, W	---	E	---	
H7 Local Control Boxes						
1. Electrical modules with safety-related function	3	SC,RZ,X, H,T,W,M	---	B	I	
2. Other electrical modules	N	SC,RZ,X, H,T,W,M	---	E	---	
J1 Nuclear Fuel						
1. Fuel assemblies	3	C,SC	---	B	I	
J2 Fuel Channel	3	C,SC	---	B	I	

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Components[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
K1 Radwaste System						
1. Drain piping including supports and valves - radioactive	N	ALL (except RZ,X)	D	E	--	(p)
2. Drain piping including supports and valves - nonradioactive	N	ALL	D	E	---	(p)
3. Piping and valves - containment isolation	2	C,SC	B	B	I	
4. Piping including supports and valves forming part of containment boundary	N	C,SC	B	B	I	
5. Pressure vessels including supports	N	W	---	E	---	(p)
6. Atmospheric tanks including supports	N	C,SC,H, T,W	---	E	---	(p)
7. 0-15 PSIG Tanks and supports	N	W	---	E	---	(p)
8. Heat exchangers and supports	N	C,SC,W	---	E	---	(p)
9. Piping including supports and valves	N	C,SC,H, T,W	---	E	---	(p)
10. Other mechanical and electrical modules	N	ALL	--- E	---	(p)	
N1 Turbine Main Steam System						
1. Deleted						

210.20

260.4

260.4

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

	Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
N1	**Turbine Main Steam System** (Continued)						
	2. Branch line of MSL including supports between the second isolation valve and the turbine stop valve from branch point at MSL to and including the first valve in the branch line	N	SC,T	B	B	---	(r)
N2	**Condensate, Feedwater and Condenser Air Extraction System**						
	1. Main feedwater line (MFW) including supports from second isolation valve branch lines and components beyond up to outboard shutoff valves	[illegible]	SC	B	[illegible]	I	
	2. Feedwater system components beyond outboard shutoff valve	N	T	D	E	---	
N3	**Heater, Drain and Vent System**	N	T	---	E	---	
N4	**Condensate Purification System**	N	T	---	E	---	
N5	**Condensate Filter Facility**	N	T	---	E	---	
N6	**Condensate Demineralizer**	N	T	---	E	---	
N7	**Main Turbine**	N	T	---	E	---	
N8	**Turbine Control System**						
	1. Turbine stop valve, turbine bypass valves, and the main steam leads from the turbine control valve to the turbine casing	N	T	D	E	---	(1)(n)(o)

2601.4

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
N9 Turbine Gland Steam System	N	T	D	E	---	
N10 Turbine Lubricating Oil System	N	T	---	E	---	
N11 Moisture Separator Heater	N	T	---	E	---	
N12 Extraction System	N	T	---	E	---	
N13 Turbine Bypass System	N	T	B	B	---	(r)
N14 Reactor Feedwater Pump Driver	N	T	---	E	---	
N15 Turbine Auxiliary Steam System	N	T	---	E	---	
N16 Generator	N	T	---	E	---	
N17 Hydrogen Gas Cooling System	N	T	---	E	---	
N18 Generator Cooling System	N	T	---	E	---	
N19 Generator Sealing Oil System	N	T	---	E	---	
N20 Exciter	N	T	---	E	---	
N21 Main Condenser	N	T	---	E	---	(cc)
N22 Offgas System	N	T	---	E	---	
N23 Circulating Water System	N	T	D	E	---	
N24 Condenser Cleanup Facility	N	T	---	E	---	

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes	
P1 Makeup Water System (Purified)							
1. Piping including supports and valves forming part of thew containment boundary	2	C	B	B	I		
2. Demineralizer water storage tank including supports	N	O	D	E	---		260.4
3. Demineralizer water header-piping including supports and valves	2	SC	B	B	I		
4. Piping including supports and valves	N	O	D	E	---		260.4
5. Other components	N	O	D	E	---		
P2 Makeup Water System (Condensate)							
1. Condensate storage tank including supports	N	O	D	E	---	(w)	
2. Condensate header - piping including supports, level instrumentation and valves	2	SC	B	B	I		
3. Piping including supports and valves and other components	N	O	D	E	---		260.4
P3 Reactor Building Cooling Water System							
1. Piping and valves forming part of primary containment boundary	2	SC,C	B	B	I	(g)	
2. Other safety-related piping, including supports, pumps and valves	3	SC,C	C	B	I		

210.20

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

	Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
	P3 Reactor Building Cooling Water System (Continued)						
	3. Electrical modules with with safety-related function	3	SC,C,X	---	B	I	
	4. Cable with safety-related function	3	SC,C,X	---	B	I	
260.4	5. Other mechanical and electrical modules	N	SC,C,X,M	---	E	---	
	P4 Turbine Building Cooling Water System	N	T	D	E	---	
	P5 HVAC Normal Cooling Water System	N	C,SC,RZ, T,X	---	E	---	
	P6 HVAC Emergency Cooling Water System						
	1. Chillers, pumps, valves, and piping including supports	3	SC,X	C	B	I	
210.20	2. Piping including supports and valves forming part of containment boundary	2	C,SC	B	B	I	
	3. Electrical modules and cable with safety-related function	3	SC,X	---	B	I	
260.4	4. Other mechanical and electrical modules	N	C,SC,RZ, T,X	---	E	---	
	P7 Oxygen Injection System	N	T	---	E	---	
	P8 Ultimate Heat Sink	N	O	---	E	---	
	P9 Reactor Service Water System						
	1. Safety-related piping including supports, piping and valves	3	U,O,X	C	B	I	

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

	Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
P9	**Reactor Service Water System** (Continued)						
	2. Electrical modules and cables with safety-related function	3	U,O,X	---	B	I	
	3. Other non-safety related mechanical and electrical modules	N	U,O,X	---	E	---	260.4
P10	**Turbine Service Water System**						
	1. Non-safety related piping including supports, piping and valves	N	P,O,T	---	E	---	260.4
	2. Electrical modules and cables with non-safety related function	N	P,O,T	---	E	---	260.4
P11	**Station Service Air System**						
	1. Containment isolation including supports, valves and piping	2	C	B	B	I	210.20
	2. Other non-safety related mechanical and electrical components	N	SC,RZ, X,T,H, W,C	---	E	---	260.4
P12	**Instrument Air Service**						
	1. Containment isolation including supports, valves and piping	2	C	B	B	I	210.20
	2. Other non-safety related mechanical and electrical components	N	SC,RZ, X,T,H, W,C	---	E	---	260.4
P13	**High Pressure Nitrogen Gas Supply Systems**						
	1. Containment isolation including supports, valves and piping	2	C	B	B	I	

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
P13 High Pressure Nitrogen Gas Supply System (Continued)						
2. Piping including supports with safety-related function	3	SC,C	C	B	I	
3. Electric modules with safety-related function	3	RZ,X	---	B	I	
4. Cable with safety-related function	3	SC,RZ,	---	B	I	
5. Other non-safety related mechanical and electrical components	N	SC,RZ, X	---	E	---	
P14 Heating Steam and Condensate Water Return System	N	ALL	---	E	---	
P15 House Boiler	N	T	---	E	---	
P16 Hot Water Heating System	N	ALL	---	E	---	
P17 Hydrogen Water Chemistry System	N	T	---	E	---	
P18 Zinc Injection System	N	T	---	E	---	
P19 Breathing Air System	N	C,SC,T	---	E	---	
P20 Sampling System (Includes PASS)	N	SC,RZ,T	---	E	---	
P21 Freeze Protection System	N	O	---	E	---	
P22 Iron Injection System	N	T	---	E	---	

260.4

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

	Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
R1	**Electrical Power Distribution System**						
	1. 120VAC safety related distribution equipment including inverters	3	SC,X,RZ	---	B	I	
	2. Motors	3	SC,C,,X, RZ	---	B	I	
	3. Load Sequencers	3	SC,X,RZ	---	B	I	
	4. Protective relays and control panels	3	SC,X,RZ	---	B	I	435.54
	5. Valve operators	3	SC,X,RZ	---	B	I	
R2	**Unit Auxiliary Transformer**						
	1. Transformers	3	SC,C,X,RZ	---	B	I	
R3	**Isolated Phase Bus**	3	RZ	---	B	I	
R4	**Non-Segrated Phase Bus**	3	RZ	---	B	I	
R5	**Metalclad Switchgear**						
	1. 6900 volt switchgear	3	SC,X,RZ	---	B	I	
R6	**Power Center**						
	1. 480 volt load centers	3	SC,X,RZ	---	B	I	
R7	**Motor Control Center**						
	1. 480 volt motor control centers	3	SC,X,RZ	---	B	I	435.54
R8	**Raceway System**						
	1. Control and power cables (including underground cable systems, cable splices, connectors and terminal blocks)	3	SC,C,X RZ	---	B	I	

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
R8 Raceway System (Continued)						
2. Conduit and cable trays and their supports	3	SC,C,X RZ	---	B	I	
R9 Grounding Wire	3	O	---	B	I	435.54
R10 Electrical Wiring Penetration	#	SC,C,X, RZ	---	B	I	
R11 Combustion Turbine Generator	N	T	---	E	---	
R12 Direct Current Power Supply						
1. 125 volt batteries, battery racks, battery chargers, and distribution equipment	3	SC,C,X RZ	---	B	I	
2. Protective relays and control panels	3	SC,X,RZ	---	B	I	435.54
3. Motors	3	SC,C,X RZ	---	B	I	

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
R13 Emergency Diesel Generator System						
1. Starting air receiver tanks piping including supports from and including check valve and downstream piping including supports and valves	3	RZ	C	B	I	(y)
2. Starting air compressor and motors	N	RZ	--	E	---	
3. Combustion air intake and exhaust system	3	RZ,O	C	B	I	
4. Safety-related piping including supports, valves - fuel oil system, diesel cooling water system, and lube oil system	3	RZ,O	C	B	I	
5. Pump motors - fuel oil system, diesel cooling water system and lube oil system	3	RZ,O	---	B	I	
6. Diesel generators	3	RZ	---	B	I	(y)
7. Mechanical and electrical modules with safety-related functions	3	RZ,O,X	---	B	I	
8. Cable with safety-related functions	3	RZ,O,X	---	B	I	
9. Other mechanical and electrical modules	N	RZ,O	---	E	---	

260.4

260.4

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
R14 Vital AC Power Supply	3	X	---	B	I	
R15 Instrument and Control Power Supply	3	X	---	B	I	
R16 Communication System	N	X	---	B	I	
R17 Lighting and Servicing Power Supply						
1. Normal Lighting	N	ALL	---	E	---	26G.4
2. Standby Lighting	3/N	ALL	C/---	B/E	I/---	
3. DC Emergency Lighting	3/N	SC,X,W	C/---	B/E	I/---	
S1 Reserve Transformer	N	T	---	E	---	
T1 Primary Containment System						
1. Primary containment vessel (PCV)-reinforced concrete containment vessel (RCCV)	2	C	B	B	I	
2. Vent system (vertical flow channels and horizontal discharges	2	C	B	B	I	
3. Suppression chamber/drywell vacuum breakers	2	C	B	B	I	
4. PCV penetrations and drywell steel head	2	C	B	B	I	
5. Upper and lower drywell airlocks	2	C,SC	---	B	I	
6. Upper and lower drywell equipment hatches	2	C,SC	---	B	I	
7. Lower drywell access tunnels	2	C	---	B	I	

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

	Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
T1	**Primary Containment System** (Continued)						
	8. Suppression chamber access hatch	2	C,SC	---	B	I	
	9. Safety related instrumentation	2	C,SC	---	B	I	
T2	**Containment Internal Sytructures**						
	1. Reactor vessel stabilizer truss	3	C	---	B	I	

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

	Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes	
T2	**Containment Internal Structures** (Continued)							
	2. Support structures for safety-related piping including supports and equipment	3	C	---	B	1		250.20
T3	**RPV Pedestal**							
	1. RPV pedestal and shield wall	3	C	---	B	1		
	2. Diaphragm floor	3	C	---	B	1		
T4	**Standby Gas Treatment System**							
	1. All equipment except deluge piping and valves	3	SC,RZ	C	B	1		
	2. Deluge piping and valves	N	SC	---	E	---		260.4
T5	**PCV Pressure and Leak Testing Facility**	N	SC	---	E	---		

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

	Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
T6	**Atmospheric Control System**						
	1. Nitrogen Storage Tanks	N	O	---	E	---	260.4
	2. Vaporizers and controls	N	O	---	E	---	260.4
	3. Piping including supports and valves forming part of containment boundary	2	C,SC	B	B	I	
	4. Electrical modules with safety-related function	3	C,SC	---	B	I	
	5. Cables with safety-related function	3	C,SC,X	---	B	I	
	6. Other piping and valves	N	SC,RZ,O	---	E	---	
T7	**Drywell Cooling System**						
	1. Motors	N	C	---	E	---	260.4
	2.Fans	N	C	---	E	---	260.4
	3. Coils. cooling	N	C	---	E	---	260.4
	4. Other mechanical and electrical modules	N	C,X	---	E	---	260.4
T8	**Flammibility Control System**	2	SC	B	B	I	
T9	**Suppression Pool Temperature Monitoring System**						
	1. Electrical modules with safety-related function	3	C,X,SC RZ	---	B	I	
	2. Cable with dsafety-related function	3	C,X,Sc RZ	---	B	I	

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

	Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
U1	**Foundation Work**	3	C,SC,RZ	---	B	I	
U2	**Turbine Pedestal**	N	T	---	E	---	
U3	**Cranes and Hoists**						
	1. Reactor Building crane	N	SC	---	E	---	(x)
	2. Refueling Bridge crane	N	SC	---	E	---	(x)
	3. Fuel handling jib crane	N	SC	---	E	---	(x)
	4. Upper Drywell Servicing	N	C	---	E	I	
	5. Lower Drywell Servicing	N	C	---	E	I	
	6. Main Steam Tunnel Servicing	N	M	---	E	---	
	7. Special Service Rooms	N	SC,RZ, T,W,X	---	E	---	
U4	**Elevator**	3/N	SC,RZ	---	B/E	I/E	
U5	**Heating, Ventilating and Air Conditioning***						
	1. Safety-related equipment**						
	a. Fan-coil cooling units	3	SC,X	---	B	I	
	b. Heating units - electrical or water	3	SC,RZ,X	---	B	I	
	c. Blowers - Air supply or	3	SC,RZ,X	---	B	I	
	d. Ductwork	3	SC,RZ,X	---	B	I	
	e. Filters - Equipment areas	3	SC,RZ,X	---	B	I	
	f. HEPA Filters, Charcoal Absorbers - Control Rooms and Secondary Containment	3	SC,X	---	B	I	

* *Includes Reactor Building, Control Building, and Service Building thermal and radiological environmental control functions within the ABWR Standard Plant.*

** *Controls environment in Main and Local control rooms, diesel-generator rooms, battery rooms, ECCS-, RCIC-, pump rooms within the ABWR Standard Plant.*

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes
U5 Heating, Ventilating, and Air Conditioning Systems* (Continued)						
g. Valves and Dampers-secondary containment isolation	2	SC,RZ	---	B	I	
h. Other safety-related valves and dampers	3	H,Z	---	B	I	
i. Electrical modules with safety-related function	3	SC,RZ H,X	---	B	I	
j. Cable with safety-related function	3	SC,RZ H,X	---	B	I	
2. Non-safety related equipment**						
a. HVAC mechanical or electrical components with non-safety related functions	N	SC,RZ,H X,W,T	---	E	---	260.4
U6 Fire Protection System						
1. Piping including supports and valves forming part of the primary containment boundary	2	C	B	B	I	
2. Other piping including supports and valves	N	SC,C,X RZ,H,T, W,O	D	E	---	(t) (u)
3. Pumps	N	F	D	E	---	(t) (u)
4. Pump motors	N	F	---	E	---	(t) (u)

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* *Includes thermal and radiological environmental control functions within the ABWR Standard Plant scope.*

** *Controls environment in rooms or areas containing non-safety related equipment within the ABWR Standard Plant.*

TABLE 3.2-1

CLASSIFICATION SUMMARY (Continued)

Principal Component[a]	Safety Class[b]	Location[c]	Quality Group Classification[d]	Quality Assurance Requirement[e]	Seismic Category[f]	Notes	
U6 Fire Protection System (Continued)							
5. Electrical Modules	N	C,SC,X RZ,H, T,W	---	E	---	(t) (u)	
6. CO_2 actuation modules	N	R	---	E	---	(t) (u)	260.4
7. Cables	N	SC,C,X	---	E	---	(t) (u)	
8. Sprinklers or deluge water	N	H,R,Z.SC X,RZ,T	D	E	---	(t) (u)	
9. Foam, preaction or deluge	N	RZ,T	---	E	---	(t) (u)	
U7 Floor Leakage Detection System							
U8 Vacuum Sweep System							
U9 Decontamination System							
U10 Reactor Building	3	SC,RZ	---	B	I		
U11 Turbine Building	N	T	---	E	---	(v)(ee)	260.4
U12 Control Building	3	X	---	B	I		
U13 Radwaste Building	N	W	---	E	---		260.4
1. Radwaste Building Substructure	3	W	---	E	I		
U14 Service Building	N	H	---	E	---		260.4
Y1 Stack	3	RZ	---	E	I		
Y2 Oil Storage and Transfer System	2/N	O	---	B/E	I/---		
Y3 Site Security	N	ALL	---	E	---		

NOTES

a. A module is an assembly of interconnected components which constitute an identifiable device or piece of equipment. For example, electrical modules include sensors, power supplies, signal processors and mechanical modules include turbines, strainers, and orifices.

b. 1, 2, 3, N + Nuclear safety-related function designation defined in Subsections 3.2.3 and 3.2.5.

c. C = Primary Containment
H = Service building
M = any other location
O = Outside onsite
RZ = Reactor Building Clean Zone (balance portion of the reactor building outside the Secondary Containment Zone)

SC = Secondary Containment portion of the reactor building
T = Turbine Building
W = Radwaste Building
X = Control Building
F = Firewater Pump House*
U = Ultimate Heat Sink Pump House*
P + Power Cycle Heat Sink Pump House*
* Pump House structures are out of the ABWR Standard Plant scope.

d. A,B,C,D = Quality groups defined in Regulatory Guide 1.26 and Subsection 3.2.2. The structures, systems and components are designed and constructed in accordance with the requirements identified in Tables 3.2-2 and 3.2-3.

--- = Quality Group Classification not applicable to this equipment.

e.B = the quality assurance requirements of 10CFR50, Appendix B are applied in accordance with the quality assurance program described in Chapter 17.

2604 E = Elements of 10CFR50, Appendix B are generally applied, commensurate with the importance of the equipment's function.

f. I = The design requirements of Seismic Category I structures and equipment are applied as described in Section 3.7, Seismic Design.

--- = The seismic design requirements for the safe shutdown earthquake (SSE) are not applicable to the equipment. However, the equipment that is not safety related but which could damage Seismic Category I equipment if its structural integrity failed is checked analytically and design to assure its integrity under seismic loading resulting from the SSE.

g. 1. Lines one inch and smaller which are part of the reactor coolant pressure boundary and are ASME Code Section III, Class 2 and Seismic Category I.

2. All instrument lines which are connected to the reactor coolant pressure boundary and are utilized to actuate and monitor safety systems shall be Safety Class 2 from the outer isolation valve or the process shutoff valve (root valve) to the sensing instrumentation.

3. All instrument lines which are connected to the reactor coolant pressure boundary and are not utilized to actuate and monitor safety systems shall be Code Group D from the outer isolation valve or the process shutoff valve (rot valve) to the sensing instrumentation.

NOTES (Continued)

4. All other instrument lines:

   i Through the root valve the lines shall be of the same classification as the system to which they are attached.

   ii Beyond the root valve, if used to actuate a safety system, the lines shall be of the same classification as the system to which they are attached.

   iii Beyond the root valve, if not used to actuate a safety system, the lines may be Code Group D.

5. All sample lines from the outer isolation valve or the process root valve through the remainder of the sampling system may be Code Group D.

6. All safety-relaed instrument sensing lines shall be in conformance with the criteria of Regulatory Guide 1.151.

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h. Relief valve discharge piping shall be Quality Group B and Seismic Category I.

   Safety/relief valve discharge line (SRVDL) piping from the safety/relief valve to the quenchers in the suppression pool consists of two parts: the first part is attached at one end to the safety/relief valve and attached at its other end to the diaphragm floor penetration. This first portion of the safety/relief valve discharge piping is analyzed with the main steam piping as a complete system. The second part of the safety/relief valve discharge piping extends from the penetration to the quenchers in the suppression pool. Because of the penetration on this part of the line, it is physically decoupled from the main steam piping and the first part of the SRVDL piping and is, therefore, analyzed as a separate piping system.

i. Electrical devices include components such as switches, controllers, solenoids, fuses, junction boxes, and transducers which are discrete components of a larger subassembly/module. Nuclear safety-related devices are Seismic Category I. Fail-safe devices are non-Seismic Category I.

j. The control rod drive insert lines from the drive flange up to and including the first valve on the hydraulic control unit are Safety Class 2, and non-safety related beyond the first valve.

k. The hydraulic control unit (HCU) is a factory-assembled engineered module of valves, tubing, piping, and stored water which controls two control rod drives by the application of pressures and flows to accomplish rapid insertion for reactor scram.

   Although the hydraulic control unit, as a unit, is field installed and connected to process piping, many of its internal parts differ markedly from process piping components because of the more complex functions they must provide. Thus, although the codes and standards invoked by Groups A, B, C, and D pressure integrity quality levels clearly apply at all levels to the interfaces between the HCU and the connection to conventional piping components (e.g., pipe nipples, fittings, simple hand valves, etc.), it is considered that they do not apply to the specialty parts (e.g., solenoid valves, pneumatic components and instruments).

NOTES (Continued)

The design and construction specifications for the HCU do invoke such codes and standards as can be reasonably applied to individual parts in developing required quality levels, but of the remaining parts and details. For example: (1) all welds are LP inspected; (2) all socket welds are inspected for gap between pipe and socket bottom; (3) all welding is performed by qualified welders; and (4) all work is done per written procedures. Quality Group D is generally applicable because the codes and standards invoked by that group contain clauses which permit the use of manufacturer standards and proven design techniques which are not explicitly defined within the codes for Quality Groups A, B, or C. This is supplemented by the QC technique described.

l. The turbine stop valve is designed to withstand the SSE and maintain its integrity.

m. The RCIC turbine is not included in the scope of standard codes. To assure that the turbine is fabricated to the standards commensurate with safety and performance requirements, General Electric has established specific design requirements for this component which are as follows:

1. All welding shall be qualified in accordance with Section IX, ASME Boiler and Pressure Vessel Code.

2. All pressure-containing castings and fabrications shall be hydrotested at 1.5 times the design pressure.

3. All high-pressure castings shall be radiographed according to:

ASTM E-94	
E-141	
E-142	maximum feasible volume
E-446, 186 or 280	Severity level 3

4. As-cast surfaces shall be magnetic-particle or liquid-penetrant tested according to ASME Code, Section III, Paragraphs NB-2545, NC-2545, or NB-2546, and NC-2546.

5. Wheel and shaft forgings shall be ultrasonically tested according to ASTM A-388.

6. Butt welds in forgings shall be radiographed and magnetic particle or liquid penetrant tested according to the ASME Boiler and Pressure Vessel Code, Section III paragraph NB-2575, NC-2575, NB-2545, NC-2545, NB-2546, NC-2546 respectively. Acceptance standards shall be in accordance with ASME Boiler and Pressure Vessel Code Section III, Paragraph NB-5320, NC-5320, NB-5340, NC-5340, NB-5350, NC-5350, respectively.

7. Notification shall be made on major repairs and records maintained thereof.

8. Record system and traceability shall be according to ASME Section III, NCA-4000.

9. Quality control and identification shall be according to ASME Section III, NCA-4000.

10. Authorized inspection procedures shall conform to ASME Section III, NB-5100 and NC-5100.

11. Non-destructive examination personnel shall be qualified and certified according to ASME Section III, NB-5500 and NC-5500.

NOTES (Continued)

n. All cast pressure-retaining parts of a size and configuration for which volumetric methods are effective are examined by radiographic methods by qualified personnel. Ultrasonic examination to equivalent standards is used as an alternate to radiographic methods. Examination procedures and acceptance standards are at least equivalent to those defined in Paragraph 136.4, Nonboiler External Piping, ANSI B31.1.

o. The following qualifications are met with respect to the certification requirements:

1. The manufacturer of the turbine stop valves, turbine control valves, turbine bypass valves, and main steam leads from turbine control valve to turbine casing utilizes quality control procedures equivalent to those defined in GE Publication GEZ-4982A, General Electric Large Steam Turbine Generator Quality Control Program.

2. A certification obtained from the manufacturer of these valves and steam loads demonstrates ? . . the quality control program as defined has been accomplished.

The following requirements shall be met in addition to the Quality Group D requirements:

1. All longitudinal and circumferential butt weld joints shall be radiogrphed (or ultrasonically tested to equivalent standards). Where size or configuration does not permit effective volumetric examination, magnetic particle or liquid penetrate examination may be substituted. Examination procedures and acceptance standards shall be at least equivalent to those specified as supplementary types of examinations, Paragraph 136.4 in ANSI B31.1.

2. All fillet and socket welds shall be examined by either magnetic particle or liquid penetrate methods. All structural attachment welds to pressure retaining materials shall be examined by either magnetic particle or liquid penetrate methods. Examination procedures and acceptance standards shall be at least equivalent to those specified as supplementary types of examinations, Paragraph 136.4 in ANSI B31.1

3. All inspection records shall be maintained for the life of the plant. These records shall include data pertaining to qualification of inspection personnel, examination procedures, and examination results.

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p. A quality assurance program meeting the guidance of Regulatory Guide 1.143 will be applied during design and construction.

q. Detailed seismic design criteria fro the offgas system are provided in Subsection 11.3.4.8.

r. The main steam lines from the containment outboard isolation valves and all branch lines 2-1/2 inches in diameter and larger, up to and including the first valve (including lines and valve supports) are designed by the use of an appropriate dynamic seismic system analysis to withstand the operating bases earthquake (OBE) and safe shutdown earthquake (SSE) design loads in combination with other appropriate loads, within the limits specified for Class 2 pipe in the ASME Section III. The mathematical model for the dynamic seismic analyses of the main steam lines and branch line piping includes the turbine stop valves and piping to the turbine casing. The dynamic input loads for design of the main steam lines are drived from a time history model analysis or am equivalent method as described in Section 3.7.

NOTES (Continued)

s. The recirculation motor cooling system (RMCS) is classified Quality Group C and Safety Class 3 which is consistent with the requirements of 10CFR50.55a. The RMCS, which is part of the reactor coolant pressure boundary (RCPB) meets 10CFR50.55a (c)(2). Postulated failure of the RMCS piping cannot cause a loss of reactor coolant in excess of normal makeup (CRD return or RCIC flow), and the RMCS is not an engineered safety feature. Thus, in the event of a postulated failure of the RMCS piping during normal operation, the reactor can be shutdown and cooled down in an orderly manner, and reactor coolant makeup can be provided by a normal make up system (e.g., CRD return or RCIC system). Thus, per 10CFR50.55a(c)(2), the RMCS need not be classified Quality Group A or Safety Class 3, however, the system is designed and constructed in accordance with ASME Boiler and Pressure Vessel Code, Section III, Class 1 criteria as specified in Subsection 3.9.3.1.4 and Figure 5.4-4.

260.4 t. A quality assurance program for the Fire Protection System meeting the guidance of Branch Technical Position CMEB 9.5-1 (NUREG-0800), is applied.

u. Special seismic qualification and quality assurance requirements are applied.

v. See Subsection 11.3.4.8 and Reg Guide 1.143, paragraph C.5 for the offgas vault seismic requirements.

w. The condensate storage tank will be designed, fabricated, and tested to meet the intent of API Standard API 650. In addition, the specification for this tank will require: (1) 100% surface examination of the side wall to bottom joint and (2) 100% volumetric examination of the side wall weld joints.

x. The cranes are designed to hold up their loads under conditions of OBE and to maintain their positions over the units under conditions of SSE.

y. All off-engine components are constructed to the extent possible to the ASME Code, Section III, Class 3.

z. Components associated with safety-related function (e.g., isolation) are safety-related.

aa. Structures which support or house safety-related mechanical or electrical components are safety-related.

bb. All quality assurance requirements shall be applied to ensure that the design, construction and testing requirements are met.

260.4 cc. A quality assurance program, which meets or exceeds the guidance of Generic Letter 85-06, is applied to all non-safety related ATWS equipment.

dd. The need for pipe whip restraints on the MSL/FW piping will be determined by a "leak-before-break" evaluation.

ee. The condenser anchorage and turbine procedure is given in Subsection 3.7.4.16 and the codes, load combinations, and structural acceptance criteria are given in Table 3.2-4.

Table 3.2-2

MINIMUM DESIGN REQUIREMENTS FOR AN ASSIGNED SAFETY DESIGNATION

	Minimum Design Requirements(6)			
Safety Designation(1)	Quality Group(2)	Seismic Category(3)	Electrical Classification(4)	Quality Assurance(5)
SC-1	A	I	---	B
SC-2	B	I	---	B
SC-3	C	I	1E	B
NNS	...(2)	...(3)	...(4)	...(5)

NOTES

1. Safety designations are defined in Subsections 3.2.3 and 3.2.5.

2. Table 3.2-3 shows applicable codes and standards for components and structures in accordance with their quality group identified in Table 3.2-1.

   Non-nuclear safety (NNS) related structures, systems and equipment that are not assigned a Quality Group in Table 3.2-1 are designed to requirements of applicable industry codes and standards (See Subsection 3.2.5.2).

   Some NNS structures, system, and component are optionally designed to Quality Group C or D requirements of Table 3.2-3, per Quality Group designation on Table 3.2-1.

3. Seismic Category I structures, systems and components meet design and analysis requirements of Subsection 3.7.

   Some NNS structures, systems and components are optionally designed to Seismic Category I design criteria as noted on Table 3.2-1. Some safety-related components (e.g., Pipe whip restraints) have no safety-related function in the event of an SSE, and are not Seismic Category I.

4. Safety-related electrical equipment and instrumentation are designated SC-3 and are designed to meet IEEE Class 1E (as well Seismic Category I) design requirements.

   Some NNS electrical equipment and instrumentation are optionally designed to IEEE Class 1E requirements as noted on Table 3.2-1.

5. Safety-related structures, systems and components meet the quality assurance requirements of 10CFR50, Appendix B, as described in Chapter 17.

   Some NNS structures, systems, and components meet the QA requirements as noted on Table 3.2-1.

6. For structural design requirements that are not covered here and in Table 3.2-3, see Section 3.8.

Table 3.2-3

QUALITY GROUP DESIGNATIONS - CODES AND INDUSTRY STANDARDS

Applicable Standards or Subsections of the ASME Code Section III

Quality Group Classi-fication	ASME Section III Code Classes	Pressure Vessels and Heat Exchangers	Pipes, Valves, and Pumps	Storage Tanks 0-15 psig	Storage Tanks Atmos-pheric	ASME Section III Component Supports	Core Support Structures	Primary Containment Boundary
A	1	NCA and NB TEMA C	NCA and NB	---	---	NCA and NF	---	---
B	2	NCA and NC TEMA C	NCA and NC	NCA and NC	NCA and NC	NCA and NF	---	---
	CC and MC	---	---	---	---	---	---	NCA, CC and NE
C	3	NCA and ND TEMA C	NCA and ND	NCA and ND	NCA and ND	NCA and NF	---	---
	CS	---	---	---	---	---	NG	---
D	---	ASME Section VIII Div 1 TEMA C	Piping & valves B31.1.0 Pumps(1)	API-620 or equiva-lent(2)	API-650 AWWA-D100 ANSI B96.1 or equivalent	---	---	---

NOTES

(1) For pumps classified in Group D, ASME Code Section VIII, Division 1, shall be used as a guide in calculating the wall thickness for pressure-retaining parts and in sizing the cover bolting.

(2) Tanks shall be dr to meet the intent of API, AWWA, and/or ANSI B96.1 Standards as applicable.

Table 3.2-4

CODES AND SPECIFICATIONS, LOADS AND LOAD COMBINATIONS, AND STRUCTURAL ACCEPTANCE CRITERIA FOR NONSEISMIC STRUCTURES

1.0 Applicable Codes and Specifications

(1) ACI318, Code Requirements for Concrete Structures

(2) AISC, Specification for Design, Fabrication and Erection of Structural Steel Buildings

2.0 Loads and Load Combinations*

2.1 Load Combinations For Concrete Members

For any load combination in this subsection, where any load reduces the effects of other loads, the corresponding coefficient for that load shall be taken as 0.9 if it can be demonstrated that the load is always present or occurs simultaneously with the other loads. Otherwise, the coefficient shall be taken as zero.

The strength design method will be used and the following load combinations will be satisfied:

$$U = 1.4D + 1.7L + 1.7H + 1.7B$$

$$U = 1.05D + 1.28L + 1.28H + 1.28B + 1.28W$$

$$U = 1.05D + 1.28L + 1.28H + 1.28B + 1.4E$$

2.2 Load Combinations For Steel Members

The elastic working stress design method is used for the following load combinations:

$$S = D + L$$

$$S = D + L + W$$

$$1.6S = D + L + E$$

In all of these load combinations, both cases of L having its full value or being completely absent are checked.

3.0 Structural Acceptance Criteria

The structural acceptance criteria are defined in ACI 318 Code and the AISC Specification. In addition, all acceptance criteria as defined in the static seismic analysis section apply.

*All abbreviations for loads have been taken from SSAR Subsection 3.8.4.3.1.1.

3.4.1.1.2.1.6 Evaluation of Floor 600 (3F)

Flooding events at this floor level may involve fuel oil as well as water. Those divisional rooms associated with the emergency diesel generator fuel tank and cooling system, have the potential of leakage from the fuel storage tanks. These rooms must accommodate leakage of 11.4 cubic meter (3000 gallons) for each division. Twenty cm (8 inches) sills on entry to these areas successfully contain all the volume in the tanks. Leakage from these tanks will also be monitored through safety grade level indication and alarm equipment so that protracted leakage as well as gross leakage can be identified. The rooms are protected by CO_2 firefighting system. Water flooding may occur from the cooling system at about .15 cubic meter/minutes (41 gpm). If undetected for several hours water may begin cascading down the nearest stairwell but is prevented from entering other division areas by raised sills.

In the SGTS areas, the room cooling equipment may cause flooding at a rate .15 cubic meter (41 gpm). Raised sills prevent intrusion of water into rooms of another division. Flooding may also occur from manual firefighting in equipment maintenance areas or from leakage from the standby liquid control tanks. Maximum tank leak rate will be .1 cubic meter/minute (25 gpm) so that a response to tank level alarms within 10 minutes will limit loss to one cubic meter (or 250 gallons). Large floor areas permit spread of water at limited depth.

3.4.1.1.2.1.7 Evaluation of Floor 700 (M4F)

Flooding in the FMCRD panel rooms may occur from firefighting activities at an input rate of .57 cubic meters/minute (150 gpm). Since these activities are manually controlled, any excessive depth of water will be noted and action taken to mitigate water intrusion to other areas.

Flooding on this level may also occur from room cooling systems or from firefighting efforts. Cooling system failures in air supply, exhaust or filter rooms may allow flooding at the rate of .3 cubic meter/minute (80 gpm) which will flow out into adjacent corridor areas. If undetected for 10 minutes, the approximate 3 cubic meter (800 gallons) released may create a depth of a few millimeters over the available floor area; a very limited amount of water will cascade down the stairwells. Divisional areas encompassing the three emergency electric supply fans and the RIP A exhaust will include raised sills to preclude water intrusion although water depth will be slight. Equipment pedestals will also minimize flooding impact on all equipment.

Firefighting activities in this area would cause water inflow of .57 cubic meter/minute (150 gpm) under controlled conditions and expected water intrusion is no more than that above.

3.4.1.1.2.1.8 Evaluation of Floor 800 (4F)

Flooding on this floor can be caused by rupture of the RCW surge tanks A, B & C piping. However, each tank and its associated piping is located in a separate compartment which can be sealed off in the event of accidental flooding. The use of raised sills on entry ways will contain the seepage to the flooded area. Also, the use of pedestals for equipment installation of the RIP supply and exhaused fans and for the DG-C exhaust fans will guard against flooding this equipment.

Flooding in the main reactor hall may occur from reactor service operations, but will be drained into service pools. Firefighting water expended into this area would occur at a maximum rate of .57 cubic meter second (150 gpm) but will spread over the large service area available. Minor amounts of water may find the way to stairwells, but would not impede operations.

3.4.1.1.2.1.9 Flooding Summary Evaluation

Floor-by-floor analysis of potential pipe failure generated flooding events in the reactor building shows the following:

(1) Where extensive flooding may occur in a division rated compartment, propagation to other divisions is prevented by watertight doors or sealed hatches. Flooding in one division is limited to that division and flood water cannot propagate to other divisions.

(2) Leakage of water from large circulating water lines, such as reactor building cooling water lines may flood rooms and corridors, but through sump alarms and leakage detection systems the control room is alerted and can control flooding by system isolation. Divisional areas are protected by watertight doors, or where only limited water depth can occur, by raised sills with pedestal mounted equipment within the protected rooms.

(3) Limited flooding that may occur from manual firefighting or from lines and tanks having limited inventory is restrained from entering division areas by raised sills and elevation differences.

Therefore, within the reactor building, internal flooding events as postulated will not prevent the safe shutdown of the reactor.

3.4.1.1.2.2 Evaluation of Control Building Flooding Events

The control building is a seven story building. It houses in separate areas, the control room proper, control and instrument cabinets with power supplies, closed cooling water pumps and heat exchangers, mechanical equipment (HVAC and chillers) necessary for building occupation and environmental control for computer and control equipment, and the steam tunnel.

The only high energy lines in the control building are the mainsteam lines and feedwater lines which pass through the steam tunnel connecting the reactor building to the turbine building. There are no openings into the control building from the steam tunnel. The tunnel is sealed at the reactor building end and open at the turbine building end. It consists of reinforced concrete with 2 meter thick walls. Any break in a mainsteam or a feedwater line will flood the steam tunnel with steam. The rate of blowdown will cause most of the steam to vent out of the tunnel into the turbine building. Water or steam cannot enter the control building. See Section 3.6.1.3.2.3 for a description of the subcompartment pressurization analysis performed for the steam tunnel.

Moderate energy water services in the control building comprise 28-inch service water lines, 18-inch cooling water lines, 6-inch cooling water lines to the chiller condenser, 6-inch fire protection lines, and 6-inch chilled water heater lines. Smaller lines supply drinking water, sanitary water and makeup for the chilled water system. Areas with water pipe routed through are supplied with floor drains and curbs to route leakage to the basement floor so that control or computer equipment is not subjected to water. In those areas where water infusion cannot be tolerated, the access sills are raised.

Maximum flooding may occur from leakage in a 28-inch service water line at a maximum rate of 12.0 cubic meters/minute (3150 gpm). Early detection by alarm to control room personnel will limit the extent of flooding which will also be mitigated by drainage to exterior of the building. The expected release of a service water leak is limited to line volume plus operator response time times leakage rate. The assumed operator response time is 30 minutes to close isolation valves and turn off the pump in the affected service water division. Water will be contained inside a division of closed cooling water equipment rooms in the bottom level of the control building. A maximum of 2.15 meters of water in a divisional room is expected. Water tight doors will confine the water to a division.

The failure of a cooling water line in the mechanical rooms of the turbine building may result in a leak of 0.6 cubic meter/minute (160 gpm). Early detection by control room personnel will limit the extent of flooding. Total release from the chilled water system will be limited to line inventory and surge tank volume, spillage of more than 6 cubic meters (1500 gallons) is unlikely. Elevation differences and separation of the mechanical functions from the remainder of the control building prevent propagation of the water to the control area.

Table 3.4-1

STRUCTURES, PENETRATIONS, AND ACCESS OPENINGS DESIGNED FOR FLOOD PROTECTION

Structure	Reactor Building	Service Building	Control Building	Radwaste Building	Turbine Building
Design Flood Level (mm)	11,700	11,700	11,700	11,700	11,700
Reference Plant Grade (mm)	12,000	12,000	12,000	12,000	12,000
Base Slab (mm)	-8,200	-2150 & 3500	-8,200	-1,500	5,300
Actual Plant Grade (mm)	12,000	12,000	12,000	12,000	12,000
Building Height (mm)	49,700	16,700	19,700	23,000	49,350
Penetrations Below Design Flood Level	Refer to Table 6.2-9	None	Refer to Table 6.2-9 for RCW lines	None	None
Access Openings Below Design Flood Level	Tunnel from S/B @ 3,500mm TMSL	Main Entrance @ grade level	Tunnel from S/B @ 7,900 mm.HX Area Access from S/B @ 12,050mm	Pipe Tunnel from R/B&T/B @1,450mm Note 3	Tunnel from [illegible] @ 7,900mm

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Notes:

1. Water tight doors (bulkhead type) are provided at all reactor and control building access ways that are below grade.

2. Water tight penetrations will be provided for all reactor and control building penetrations that are below grade.

3. The lines that run through the radwaste building tunnel are not exposed to outside ground flooding

Table 3.4-2

(DELETED)

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3.5.1.1 Internally Generated Missiles (Outside Containment)

These missiles are considered to be those missiles resulting internally from plant equipment failures within the ABWR Standard Plant but outside containment.

3.5.1.1.1 Rotating Equipment

3.5.1.1.1. Missile Characterization

Equipment within the general categories of pumps, fans, blowers, diesel generators, compressors, and turbines and, in particular, components in systems normally functioning during power reactor operation, has been examined for any possible source of credible and significant missiles.

3.5.1.1.1.2 RCIC Steam Turbine

The RCIC steam turbine driving the pump is not a credible source of missiles. It is provided with mechanical overspeed protection as well as automatic governing; very extensive industrial and nuclear experience with this model of turbine has never resulted in a missile which penetrated the turbine casing.

3.5.1.1.1.3 Main Steam Turbine

251.13

Acceptance criteria 1 of SRP Section 3.5.1.3 considers a plant with a favorable turbine generator placement and orientation and adhering to the guidelines of Regulatory Guide 1.115 adequately protected against turbine missile hazards. Further, this criterion specifies that exclusions of safety-related structures, systems or components from low trajectory turbine missile strike zones constitutes adequate protection against low trajectory turbine missiles. The turbine generator placement and orientation of the ABWR Standard Plant meets the guidelines of Regulatory Guide 1.115 as illustrated in Figure 3.5-2.

In addition, the applicant referencing the ABWR design shall;

(1) Submit for NRC approval, within three years of obtaining an operating license, a turbine system maintenance program including probability calculations of turbine missile generation based on the NRC approved methodology (such as Reference 10), or

(2) Volumetrically inspect all low pressure turbine rotors at the second refueling outage and every other (alternate) refueling outage thereafter until a maintenance program is approved by the staff.

(3) Meet the minimum requirement for the probability of turbine missile generation given in Table 3.5-1.

3.5.1.1.1.4 Other Missile Analysis

No remaining credible missiles meet the significance criteria of having a probability (P_4) greater than 10^{-7} per year for rotating or pressurized equipment, because either:

(1) The equipment design and manufacturing criteria mentioned previously result in (P_1) being less than 10^{-7} per year; or

(2) Sufficient physical separation (barriers and/or distance) of safety-related and redundant equipment exists so that the combined probability ($P_1 \times P_2$) is less than 10^{-7} per year.

These conclusions are arrived at by noting that pumps, fans, and the like are AC powered. Their speed is governed by the frequency of the AC power supply. Since the AC power supply frequency variation is limited to a narrow range, it is not likely they will attain an overspeed condition. At rated speed, if a piece such as a fan blade breaks off, it will not penetrate the casing. As an example, a containment high purge exhaust fan has been analyzed for a thrown blade at rated speed conditions using an analytical expression from Reference 1. It is determined, based on maximum thickness this blade could penetrate, that the blade would not escape the fan casing and consequently (P_1) is less than 10^{-7} per year.

3.5.1.1.2 Pressurized Components

3.5.1.1.2.1 Missile Characterization

Potential missiles which could result from the failure of pressurized components are analyzed in this subsection. These potential missiles may be categorized as contained fluid-energy missiles or stored strain-energy (elastic) missiles. These potential missiles have been conservatively evaluated against the design criteria in Subsection 3.5.1.

Examples of potential contained fluid-energy missiles are valve bonnets, valve stems, and retaining bolts. Valve bonnets are considered jet-propelled missiles and have been analyzed as such. Valve stems have been analyzed as piston-type missiles, while retaining bolts are examples of stored strain-energy missiles.

3.5.1.1.2.2 Missile Analyses

Pressurized components outside the containment capable of producing missiles have been reviewed. Although piping failures could result

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including deadweight and SSE (inertial) components.

Shielded Metal Arc (SMAW) and Submerged Arc (SAW) Welds:

The flow stress used to construct the master curve is 51 ksi

The value of SI used to enter the master curve for SMAW and SAW is

$$SI = M (P_m + P_b + P_e) Z \quad (8)$$

where

P_b = the combined primary bending stress, including deadweight and seismic components.

P_e = combined expansion stress at normal operation.

$$Z = 1.15 [1.0 + 0.013 (OD-4)] \text{ for SMAW,} \quad (9)$$

$$Z = 1.30 [1.0 + 0.010 (OD-4)] \text{ for SAW,} \quad (10)$$

and

OD = pipe outer diameter in inches.

When the allowable flaw length is determined from the master curve at the appropriate SI value, it can be used to determine if the required margins on load and flaw size are met using the following procedure.

For the method of load combination described in item (5), let M = 1.4, and if the allowable flaw length from the master curve is at least equal to the leakage size flaw, then the margin on load is met.

3.6.4 Interfaces

3.6.4.1 Details of Pipe Break Analysis Results and Protection Methods

The following shall be provided by the applicant referencing the ABWR design (See Subsection 3.6.2.5):

(1) A summary of the dynamic analyses applicable to high-energy piping systems in accordance with Subsection 3.6.2.5 of Regulatory Guide 1.70. This shall include:

(a) Sketches of applicable piping systems showing the location, size and orientation of postulated pipe breaks and the location of pipe whip restraints and jet impingement barriers.

(b) A summary of the data developed to select postulated break locations including calculated stress intensities, cumulative usage factors and stress ranges as delineated in BTP MEB 3-1.

(2) For failure in the moderate-energy piping systems listed in Table 3.6-6, descriptions showing how safety-related systems are protected from the resulting jets, flooding and other adverse environmental effects. 410.21

(3) Identification of protective measures provided against the effects of postulated pipe failures in each of the systems listed in Tables 3.6-1, 3.6-2 and 3.6-4. 410.22

(4) The details of how the MSIV functional capability is protected against the effects of postulated pipe failures. 410.26

(5) Typical examples, if any, where protection for safety-related systems and components against the dynamic effects of pipe failures include their enclosure in suitably designed structures or compartments (including any additional drainage system or equipment environmental qualification needs). 410.28

(6) The details of how the feedwater line check and feedwater isolation valves functional capabilities are protected against the effects of postulated pipe failures.

3.6.4.2 Leak-Before-Break Analysis Report

As required by Reference 1, an LBB analysis report shall be prepared for the piping systems proposed for inclusion from the analyses for the dynamic effects due to their failure. The report shall include only the piping stress analysis results for the piping systems analyzed and reported for LBB in Appendix 3F in order to show that the piping stresses are within the stress levels assumed in Appendix 3F (See Subsection 3.6.3).

3.6.5 References

1. *Modification of General Design Criterion 4 Requirements for Protection Against Dynamic Effects of Postulated Pipe Rupture,* Federal Register, Volume 52, No. 207, Rules and Regulations, Pages 41288 to 41295, October 27, 1987

2. *RELAP 3, A Computer Program for Reactor Blowdown Analysis,* IN-1321, issued June 1970, Reactor Technology TID-4500.

3. Moody, F. J., *Fluid Reactor and Impingement Loads,* Vol. 1, ASCE Specialty Conference on Structural Design of Nuclear Plant Facilities pp. 219-262, December 1973.

4. *Standard Review Plan; Public Comments Solicited,* Federal Register, Volume 52, No. 167, Notices, Pages 32626 to 32633, August 28, 1987.

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TABLES

3.7.3.8.2.2 Effect of Differential Building Movements

The relative displacement between anchors is determined from the dynamic analysis of the structures. The results of the relative anchor-point displacement are used in a static analysis to determine the additional stresses due to relative anchor-point displacements. Further details are given in Subsection 3.7.3.8.1.8.

3.7.3.9 Multiple Supported Equipment Components With Distinct Inputs

The procedure and criteria for analysis are described in Subsections 3.7.2.1.3 and 3.7.3.3.1.3.

3.7.3.10 Use of Constant Vertical Static Factors

All Seismic Category I subsystems and components are subjected to a vertical dynamic analysis with the vertical floor spectra or time histories defining the input. A static analysis is performed in lieu of dynamic analysis if the peak value of the applicable floor spectra times a factor of 1.5 is used in the analysis. A factor of 1.0 instead of 1.5 can be used if the equipment is simple enough such that it behaves essentially as a single degree of freedom system. If the fundamental frequency of a component in the vertical direction is greater than or equal to 33 Hz, it is treated as seismically rigid and analyzed statically using the zero-pe-sponse spectrum.

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3.7.3.11 Torsional Effects of Eccentric Masses

Torsional effects of eccentric masses are included for Seismic Category I subsystems similar to that for the piping systems discussed in Subsection 3.7.3.3.1.2.

3.7.3.12 Buried Seismic Category I Piping and Tunnels

For buried Category I buried piping systems and tunnels the following items are considered in the analysis:

(1) The inertial effects due to an earthquake upon buried systems and tunnels will be adequately accounted for in the analysis. In case of buried systems sufficiently flexible relative to the surrounding or underlying soil, it is assumed that the systems will follow essentially the displacements and deformations that the soil would have if the systems were absent. When applicable, procedures, which take into account the phenomena of wave travel and wave reflection in compacting soil displacements from the ground displacements, are employed.

(2) The effects of static resistance of the surrounding soil on piping deformations or displacements, differential movements of piping anchors, bent geometry and curvature changes, etc., are considered. When applicable, procedures utilizing the principles of the theory of structures on elastic foundations are used.

(3) When applicable, the effects due to local soil settlements, soil arching, etc., are also considered in the analysis.

3.7.3.13 Interaction of Other Piping with Seismic Category I Piping

In certain instances, non-Seismic Category I piping may be connected to Seismic Category I piping at locations other than a piece of equipment which, for purposes of analysis, could be represented as an anchor. The transition points typically occur at Seismic Category I valves which may or may not be physically anchored. Since a dynamic analysis must be modeled from pipe anchor point to anchor point, two options exist:

(1) specify and design a structural anchor at the Seismic Category I valve and analyze the Seismic Category I subsystem; or, if impractical to design an anchor,

(2) analyze the subsystem from the anchor point in the Seismic Category I subsystem through the valve to either the first anchor point in the non-Seismic Category I subsystem; or to sufficient distance in the non-Seismic Category I Subsystem so as not to significantly degrade the accuracy of analysis of the Seismic Category I piping.

Where small, non-Seismic Category piping is directly attached to Seismic Category I piping, its effect on the Seismic Category I piping is accounted for by lumping a portion of its mass with the Seismic Category I piping at the point of attachment.

Furthermore, non-Seismic Category I piping (particularly high energy piping as defined in Section 3.6) is designed to withstand the SSE to avoid jeopardizing adjacent Seismic Category I piping if it is not feasible or practical to isolate these two piping systems.

3.7.3.14 Seismic Analysis for Reactor Internals

The modeling of RPV internals is discussed in Subsection 3.7.2.3.2. The damping values are given in Table 3.7-1. The seismic model of the RPV and internal in shown in Figure 3.7-32.

3.7.3.15 Analysis Procedures for Damping

The modeling of RPV internals is discussed in Subsection 3.7.2.3.2. The damping values are given in Table 3.7-1. The seismic model of the RPV and internals is shown in Figure 3.7-32.

3.7.3.16 Analysis Procedure for NonSeismic Structures in Lieu of Dynamic Analysis

The method described here can be used for non-seismic structures in lieu of a dynamic analysis.

Structures designed to this method should be able to do the following:

(1) Resist minor levels of earthquake ground motion without damage.

(2) Resist moderate levels of earthquake ground motion without structural damage, but possibly experience some nonstructural damage.

(3) Resist major levels of earthquake ground motion having an intensity equal to the strongest either experienced or forecast at the building site, without collapse, but possibly with some structural as well as nonstructural damage.

3.7.3.16.1 Lateral Forces

Seismic loads are characterized as a force profile that varies with the height of the structure. These forces are applied at each floor of the structure and the resulting forces and moments are calculated from static equilibrium.

The buildings total base shear is characterized by the following equation:

$V = Z*I*C*W/R_w$; where,

V	=	Total lateral force or shear at the base.
$F_x F_i F_n$	=	Lateral force applied to level i, n, or x respectively.
F_t	=	That portion of V considered to be concentrated at the top of the structure in addition to F_n
Z	=	Seismic zone factor
I	=	Importance factor
C	=	Numerical Coefficient
R_w	=	Numerical Coefficient
S	=	Coefficient for site soil characteristics
T	=	Fundamental period of vibration of the structure in the direction under consideration, as determined by using the properties and deformation characteristics of the resisting elements in a properly substantiated analysis.
W	=	Total dead load of building including the partition load where applicable.
$w_i w_x$	=	That portion of W which is located at or is assigned to level i or x, respectively
$h_i h_x$	=	Height in feet above the base to level i or x, respectively

The ABWR design will fix Z and I and leave R and C as variables for each building and site.

The value of I has been selected for power generating facilities.

$I = 1.0$

The site coefficient Z will be selected to provide enveloping coverage for most of the U.S. east of rocky mountains.

$Z = 0.15$

The value of C is calculated based upon the following formula:

$C = 1.25*ST^{2/3}$

Where: C need not exceed 2.75

The value of S is dependent on the site soil characteristics. The value of S shall be selected from Table 3.7-11.

The value of R_w shall be selected from Table 3.7-12 according to the type of construction material and framing system under consideration.

3.7.3.16.2 Lateral Force Distribution

The concentrated force at the top of the structure shall be determined according to the following formula:

$F_t = 0.07*T*V$ where,

F_t need not exceed 0.25V and may be considered as 0 where T is 0.7 seconds or less. The remaining portion of the total base shear V shall be distributed over the rest of the structure including level n according to the following formula:

$$F_x = \frac{(V - F_t)\, w_x h_x}{\sum_{i=1}^{n} w_i h_i}$$

At each level designated x, the force F_x shall be applied over the area of the building in accordance with the mass distribution on that level.

3.7.3.16.3 Accident Torsion

In addition, the vertical resisting elements depend on diaphragm action for shear distribution at any level, the shear resisting elements shall be capable of resisting torsional moment assumed to be equivalent to the story shear acting with an eccentricity of not less than 5 percent of the maximum building dimension at that level.

3.7.3.16.4 Lateral Displacement Limits

Lateral deflections or drift of a story relative to its adjacent stories shall not exceed 0.005 times the story height nor $0.04/R_w$ for buildings less than 65 feet in height. For buildings greater in height, the calculated story drift shall not exceed 0.004 times the story height nor $0.04/R_w$. These drift limits may be exceeded when it is demonstrated that greater drift can be tolerated by both structural elements and nonstructural elements that could effect life or safety. For designs using working stress methods, this capacity may be determined using an allowable stress increase of 1.7. The rigidity of other elements shall also be considered.

3.7.3.16.5 Ductility Requirements

All framing not required by design to be part of the lateral force-resisting system shall be investigated and shown to be adequate for vertical load-carrying capacity and induced moment due to $3R_w/8$ times the distortions resulting from the code required lateral forces.

Connections shall be designed to develop the full capacity of the members or shall be based upon the above forces without the one-third increase usually permitted for stresses resulting from earthquake forces.

3.7.4 Seismic Instrumentation

3.7.4.1 Comparison with NRC Regulatory Guide 1.12

The seismic instrumentation program is consistent with Regulatory Guide 1.12.

3.7.4.2 Location and Description of Instrumentation

The following instrumentation and associated equipment are used to measure plant response to

earthquake motion:

(1) three triaxial time-history accelerographs (THA);

(2) three peak-recording accelerographs (PRA);

(3) two triaxial seismic triggers;

(4) one seismic switch (SS);

(5) four response spectrum recorders;

(6) recording and playback equipment; and

(7) annuciators.

The location of seismic instrumentation is outlined in Table 3.7-7.

3.7.4.2.1 Time-History Accelerographs

Time-history accelerographs produce a record of the time-varying acceleration at the sensor location. This data is used directly for analysis and comparison with reference information and may be, by calculational methods, converted to response spectra form for spectra comparisons with design parameters.

Each triaxial acceleration sensor unit contains three accelerometers mounted in an orthogonal array (two horizontal and one vertical). All acceleration units have their principal axes oriented identically. The mounted units are oriented so that their axes are aligned with the building major axes used in development of the mathematical models for seismic analysis.

One THA is located on the reactor building (RB) foundation mat, El (-) 13.2 M, at the base of an RB clean zone for the purpose of measuring the input vibratory motion of the foundation mat. A second THA is located in an RB clean zone at El (+) 26.7 M on the same azimuth as the foundation mat THA. They provide data on the frequency, amplitude, and phase relationship of the seismic response of the reactor building structure. A third THA is located in the free field at the finished grade approximately 160 M from any station structures with axes oriented in the same direction as the reactor building accelerometers.

Two seismic triggers, connected to form redundant triggering, are provided to start the THA recording system. They are located in the free field at the finished grade 160 M from the reactor building. The trigger unit consists of orthogonally mounted acceleration sensors that actuate relays whenever a threshold acceleration is exceeded for any of the three axes. The trigger in engineered to discriminate against false starts from other operating inputs such as traffic, elevators, people, and rotating equipment.

Table 3.7-11

SITE COEFFICIENTS

Type	Description	S Factor
S_1	A soil profile with either; (a) A rock like material characterized by a shear wave velocity greater than 2,500 fps or by other suitable means of classification. or (b) Stiff or dense soil condition where soil depth is less than 200 ft.	1.0
S_2	A soil profile with dense or stiff soil conditions, where the soil depth exceeds 200 feet.	1.2
S_3	A soil profile 40 feet or more in depth and containing more than 20 feet of soft to medium stiff clay but not more than 40 feet of soft clay.	1.5
S_4	A soil profile containing more than 40 feet of soft clay.	2.0

Table 3.7-12

STRUCTURAL SYSTEMS

Basic Structural	Lateral Load Resisting System Description	R_w
A Bearing wall	1. Shear walls - concrete	6
	2a. Braced frames where bracing carries gravity loads - steel	6
	2b. Braced frames where bracing carries gravity loads - concrete	4
B Building frame	1. Steel eccentric braced frame	10
	Shear walls - concrete	8
	Concentric braced frames - steel	8
	Concentric braced frames - concrete	8
C Moment resisting frame	Special moment resisting space frames	12
	Concrete intermediate moment-resisting space frames (OMRSF)	7
	Ordinary moment resisting space frames (OMRSF) - steel	6
	Ordinary moment resisting space frames (OMRSF) - concrete	5
D Dual		
1. Shear walls	a. Concrete with SMRSF	12
	b. Concrete with concrete IMRSF	9
2. Steel EBF with steel SMRSF		12
3. Concentric braced frames	a. Steel with steel SMRSF	10
	b Concrete with concrete SMRSF	9
	c. Concrete with concrete IMRSF	6

interaction of the substructure with the underlying foundation medium. For a mat foundation supported on soil or rock, the pertinent aspects in the design are to maintain the bearing pressures within allowable limits, particularly due to overturning forces, and to ensure that there is adequate frictional and passive resistance to prevent sliding of the structure when subjected to lateral loads.

The design loads considered in analysis of the foundations are the worst resulting forces from the superstructures and loads directly applied to the foundation mat due to static and dynamic load combinations.

The capability of the foundation to transfer shear with waterproofing will be evaluated (see Subsection 3.8.6.1).

The standard ABWR design is developed using a range of soil conditions as detailed in Appendix 3A. The variations of physical properties of the site-specific subgrade materials will be determined (see Subsection 3.8.6.2). Settlement of the foundations, differential settlement between foundations for the site-specific foundations medium will be calculated and safety-related systems (i.e., piping, conduit, etc.) will be designed for the calculated settlement of the foundations. The effect of the site-specific subgrade stiffness and calculated settlement on the design of the seismic Category I structures and foundations will be evaluated (see Subsection 3.8.6.2).

A detailed description of the analytical and design methods for the reactor building foundation mat including the containment foundation, is included in Section 3.8.1.4.

3.8.5.5 Structural Acceptance Criteria

The main structural criteria for the containment portion of the foundation are adequate strength to resist loads and sufficient stiffness to protect the containment liner from excessive strain. The acceptance criteria for the containment portion of the foundation mat are presented in Subsection 3.8.1.5. The structural acceptance criteria for the reactor building foundations are described in Subsection 3.8.4.5.

The calculated and allowable factors of safety of the ABWR structures for overturning, sliding, and flotation are shown in Sections 3H.1 and 3H.2.

3.8.5.6 Materials, Quality Control, and Special Construction Techniques

The foundations of seismic Category I structures are constructed of reinforced concrete using proven methods common to heavy industrial construction. For further discussion see Subsections 3.8.1.6 and 3.8.4.6.

3.8.5.7 Testing and Inservice Inspection Requirements

A formal program of testing and inservice inspection is not planned and is not required for the seismic Category I structures of the ABWR.

3.8.6 Interfaces

3.8.6.1 Foundation Waterproofing

The capability of foundations to transfer shear loads where foundation waterproofing is used will be evaluated (see Subsection 3.8.5.4).

3.8.6.2 Site Specific Physical Properties and Foundation Settlement

Physical properties of the site specific subgrade medium shall be determined and the settlement of foundations and structures including seismic Category I will be evaluated see Subsection 3.8.5.4).

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TABLES

sizing of each minimum recirculation flow path is evaluated to assure that its use under all analyzed conditions will not result in degradation of the pump. The flow rate through minimum recirculation flow paths can also be periodically measured to verify that flow is in accordance with the design specification.

The safety-related pumps are provided with instrumentation to verify that the net positive suction head (NPSH) is greater than or equal to the NPSH required during all modes of pump operation. These pumps can be disassembled for evaluation when the Code Section XI testing results in a deviation which falls within the "required action range." The Code provides criteria limits for the test parameters indentified in Table 3.9-8. A program will be developed by the applicant referencing the ABWR design to establish the frequency and the extent of disassembly and inspection based on suspected degradation of all safety related pumps, including the basis for the frequency and the extent of each disassembly. The program may be revised throughout the plant life to minimize disassembly based on past disassembly experience. (See Subsection 3.9.7.3(1) for interface requirements.)

3.9.6.2 Inservice Testing of Safety-Related Valves

3.9.6.2.1 Check Valves

All ABWR safety-related piping systems incorporate provisions for testing to demonstrate the operability of the check valves under design conditions. In-service testing will incorporate the use of advance non-intrusive techniques to periodically assess degradation and the performance characteristics of the check valves. The Code Section XI tests will be performed, and check valves that fail to exhibit the required performance can be disassembled for evaluation. The Code provides criteria limits for the test parameters identified in Table 3.9-8. A program will be developed by the applicant referencing the ABWR design to establish the frequency and the extent of disassembly and inspection based on suspected degradation of all safety related pumps, including the basis for the frequency and the extent of each disassembly. The program may be revised throughout the plant life to minimize disassembly based on past disassembly experience. (See Subsection 3.9.7.3(1) for interface requirements.)

3.9.6.2.2 Motor Operated Valves

The motor operated valve (MOV) equipment specifications require the incorporation of the results of either in-situ or prototype testing with full flow and pressure or full differential pressure to verify the proper sizing and correct switch settings of the valves. Guidelines to justify prototype testing are contained in Generic Letter 98-10, Supplement 1, Questions 22 and 24 through 28. The applicant referencing the ABWR design will provide a study to determine the optimal frequency for valve stroking during in-service testing such that unnecessary testing and damage is not done to the valve as a result of the testing. (See Subsection 3.9.7.3 for interface requirements).

The concerns and issues identified in Generic Letter 89-10 for MOVs will be addressed prior to plant startup. The method of assessing the loads, the method of sizing the actuators, and the setting of the torque and limit switches will be specifically addressed. (See Subsection 3.9.7.3 for interface requirements).

The in-service testing of MOVs will rely on diagnostic tecniques that are consistent with the state of the art and which will permit an assessment of the performance of the valve under actual loading. Periodic testing will be conducted under adequate differential pressure and flow conditions that allow a justifiable demonstration of continuing MOV capability for design basis conditions, including recovery from inadvertent valve positioning. MOVs that fail the acceptance criteria, and are "declared inoperable," for stroke tests and leakage rate can be disassembled for evaluation. The Code provides criteria limits for the test parameters identified in Table 3.9-8. A program will be developed by the apllicant referencing the ABWR design to establish the frequency and the extent of disassembly and inspection based on suspected degradation of all safety related "MOV's", including the basis for the frequency and the extent of each disassembly. The program may be revised throughout the plant life to minimize disassembly based on past disassembly exper-

ience. (See Subsection 3.9.7.3(1) for interface requirements.)

3.9.6.2.3 Isolation Valve Leak Tests

The leak-tight integrity will be verified for each valve relied upon to provide a leak-tight function. These valves include:

(1) pressure isolation valves - valves that provide isolation of pressure differential from one part of a system from another or between systems;

(2) temperature isolation valves - valves whose leakage may cause unacceptable thermal loading on supports or stratification in the piping and thermal loading on supports or whose leakage may cause steam binding of pumps; and

(3) containment isolation valves - valves that perform a containment isolation function in accordance with the Evaluation Against Criterion 54, Subsection 3.1.2.5.5.2, including valves that may be exempted from Appendix J, Type C, testing but whose leakage may cause loss of suppression pool water inventory.

Leakage rate testing of valves will be in accordance with the Code Section XI. An example is the fusible plug valves that provide a lower drywell flood for severe accidents described in Subsection 9.5.12. The valves are safety-related due to the function of retaining suppression pool water as shown in Figure 9.5-3. These special valves are noted here and not in Table 3.9-8. The fusible plug valve is a nonreclosing pressure relief device and the Code requires replacement of each at a maximum of 5 year intervals.

3.9.7 Interfaces

3.9.7.1 Reactor Internals Vibration Analysis, Measurement and Inspection Program

The first applicant referencing the ABWR design will provide, at the time of application, the results of the vibration assessment program for the ABWR prototype internals. These results will include the following information specified in Regulatory Guide 1.20.

R.G. 1.20	Subject
C.2.1	Vibration Analysis Program
C.2.2	Vibration Measurement Program
C.2.3	Inspection Program
C.2.4	Documentation of Results

NRC review and approval of the above information on the first applicants docket will complete the vibration assessment program requirements for prototype reactor internals.

In addition to the information tabulated above, the first applicant referencing the ABWR design will provide the information on the schedules in accordance with the applicable portions of position C.3 of Regulatory Guide 1.20 for non-prototype internals.

Subsequent applicants need only provide the information on the schedules in accordance with the applicable portions of position C.3 of Regulatory Guide 1.20 for non-prototype internals. (See Subsection 3.9.2.4 for interface requirements).

3.9.7.2 ASME Class 2 or 3 or Quality Group Components with 60 Year Design Life

Applicants referencing the ABWR design will identify ASME Class 2 or 3 or Quality Group D components that are subjected to loadings which could result in thermal or dynamic fatigue and provide the analyses required by the ASME Code, Subsection NB. These analyses will include the appropriate operating vibration loads and for the effects of mixing hot and cold fluids. (See Subsection 3.9.3.1 for interface requirements).

3.9.7.3 Pump and Valve Inservice Testing Program

Applicants referncing the ABWR design will provide a plan for the detailed pump and valve inservice testing and inspection program. This plan will

(1) Include baseline pre-service testing to support the periodic in-service testing of the components required by technical specifications. Provisions are included to disassemble and inspect the pump, check valves, and MOVs within the Code and safety-related classification as necessary, depending on test results. (See Subsections 3.9.6, 3.9.6.1, 3.9.6.2.1 and 3.9.6.2.2)

(2) Provide a study to determine the optimal frequency for valve stroking during inservice testing. (See Subsection 3.9.6.2.2)

(3) Address the concerns and issues identified in Generic Letter 89-10; specifically the method of assessment of the loads, the method of sizing the actuators, and the setting of the torque and limit switches (See Subsection 3.9.6.2.2)

3.9.7.4 Audit of Design Specification and Design Reports

Applicants referencing the ABWR design will make available to the NRC staff design specification and design reports required by ASME Code for vessels, pumps, valves and piping systems for the purpose of audit. (See Subsection 3.9.3.1)

3.9.8 References

1. *BWR Fuel Channel Mechanical Design and Deflection*, NEDE-21354-P, September 1976.

2. *BWR/6 Fuel Assembly Evaluation of Combined Safe Shutdown Earthquake (SSE) and Loss-of-Coolant Accident (LOCA) Loadings*, NEDE-21175-P, November 1976.

3. NEDE-24057-P (Class III) and NEDE-24057 (Class I) Assessment of Reactor Internals. Vibration in BWR/4 and BWR/5 Plants,

Table 3.9-8 (Continued)

INSERVICE TESTING SAFETY-RELATED PUMPS AND VALVES

E51 Reactor Core Isolation Cooling System (Continued)

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F006	1	Suppression Pool (CSP) suction line MOV	2	A	I,A	L P,S	2 yrs 3 mo	5.4-8a
F007	1	Suppression Pool (CSP) suction line check vlv	2	C	A	P,S	3 mo	5.4-8a
F008	1	RCIC Sys suppr pool test return line MOV	2	B	A	P,S	3 mo	5.4-8a
F009	1	RCIC Sys suppr pool test return line MOV	2	B	I,A	L P,S	2 yrs 3 mo	5.4-8a
F010	1	RCIC Sys minimum flow bypass line check vlv	2	C	A	P,S	3 mo	5.4-8a
F011	1	RCIC Sys minimum flow bypass line MOV	2	B	I,A	L P,S	2 yrs 3 mo	5.4-8a
F012	1	RCIC turbine accessories cooling water line MOV	2	B	A	P,S	3 mo	5.4-8c
F013	1	RCIC turbine accessories cooling water line PCV	2	B	A	E1		5.4-8c
F015	1	Barometric condenser condensate pump discharge line valve	2	B	P	E1		5.4-8c
F016	1	Barometric condenser condensate pump discharge line check valve	2	C	P	P,S	3 mo	5.4-8c
F017	1	RCIC pump suction line relief valve	2	C	A	L,S	2 yrs	5.4-8a
F018	1	Valve in the bypass line around check valve E51-F003	2	B	P	E1		5.4-8a
F019	1	Pump discharge line test line valve	2	B	P	E1		5.4-8a
F020	1	Pump discharge line test line valve	2	B	P	E1		5.4-8a
F021	1	Pump discharge line fill line shutoff valve	2	B	P	E1		5.4-8a
F022	1	Pump discharge line fill line check valve	2	C	A	P,S	3 mo	5.4-8a
F023	1	Pump discharge line fill line check valve	2	C	A	P,S	3 mo	5.4-8a
F024	1	Pump discharge line test line valve	2	B	P	E1		5.4-8a
F025	1	Pump discharge line test line valve	2	B	P	E1		5.4-8a
F026	1	Valve in pressure equalizing line around E51-F005	2	B	P	E1		5.4-8a
F027	1	Suppression Pool (S/P) suction line test line valve	2	B	P	E1		5.4-8a
F028	1	Minimum flow bypass line test line valve	2	B	P	E1		5.4-8a
F029	1	Minimum flow bypass line test line valve	2	B	P	E1		5.4-8a
F030	1	Turbine accessories cooling water line relief valve	2	C	A	L,S	2 yrs	5.4-8c
F031	1	Barometric condenser condensate discharge line AOV to HCW	2	B	P	E1		5.4-8c
F032	1	Barometric condenser condensate discharge line AOV to HCW	2	B	P	E1		5.4-8c
F033	1	Discharge line fill line bypass line shutoff valve	2	B	P	E1		5.4-8a

Table 3.9-8 (Continued)

INSERVICE TESTING SAFETY-RELATED PUMPS AND VALVES

E51 Reactor Core Isolation Cooling System (Continued)

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F034	1	Barometric condenser condensate pump discharge line test line valve	2	B	P	E1		5.4-8c
F035	1	Steam supply line isolation valve	1	A	I,A	L P,S	2 yrs 3 mo	5.4-8b
F036	1	Steam supply line isolation valve	1	A	I,A	L P,S	2 yrs 3 mo	5.4-8b
F037	1	Steam admission valve	2	B	A	P,S	3 mo	5.4-8a
F038	1	Turbine exhaust line check valve	2	C	I,A	L P,S	2 yrs 3 mo	5.4-8a
F039	1	Turbine exhaust line MOV	2	A	I,A	L P,S	2 yrs 3 mo	5.4-8a
F040	1	Steam supply line drain pot drain line AOV	2	B	P			5.4-8b
F041	1	Steam supply line drain pot drain line AOV	2	B	P			5.4-8b
F044	1	Steam admission valve bypass line maintenance valve	2	B	P			5.4-8b
F045	1	Steam admission valve bypass line MOV	2	B	A	P,S	3 mo	5.4-8b
F046	1	Barometric condenser vacuum pump discharge line check valve	2	C	A	L P,S	2 yrs 3 mo	5.4-8a
F047	1	Barometric condenser vacuum pump discharge line MOV	2	A	I,A	L P,S	2 yrs 3 mo	5.4-8a
F048	1	Steam supply line warm-up line valve	1	A	I,A	L P,S	2 yrs 3 mo	5.4-8b
F049	1	Steam supply line test line valve	2	B	P	E1		5.4-8b
F050	1	Steam supply line test line valve	2	B	P	E1		5.4-8b
F051	1	Turbine exhaust line drain line valve	2	B	P	E1		5.4-8c
F052	1	Turbine exhaust line drain line valve	2	B	P	E1		5.4-8c
F053	1	Turbine exhaust line test line valve	2	B	P	E1		5.4-8a
F054	1	Turbine exhaust line vacuum breaker	2	C	A	P,S	3 mo	5.4-8a
F055	1	Turbine exhaust line vacuuum breaker	2	C	A	P,S	3 mo	5.4-8a
F056	1	Steam supply line drain pot drain line test line valve	2	B	P	E1		5.4-8b
F057	1	Steam supply line drain pot drain line test drain line	2	B	P	E1		5.4-8b
F059	1	Baromemtric condener vacuum pump discharge line test line valve	2	B	P	E1		5.4-8a
F500	1	Pump discharge line vent line valve	2	B	P	E1		5.4-8a
F501	1	Pump discharge line vent line valve	2	B	P	E1		5.4-8a
F502	1	Pump discharge line drain line valve	2	B	P	E1		5.4-8a
F503	1	Pump discharge line drain line valve	2	B	P	E1		5.4-8a
F700	1	Pump suction line pressure instrumentation instrument root valve	2	B	P	E1		5.4-8a

Table 3.9-8 (Continued)

INSERVICE TESTING SAFETY-RELATED PUMPS AND VALVES

E51 Reactor Core Isolation Cooling System (Continued)

No.	Qty	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F701	1	Pump suction line pressure instrumentation instrument root valve	2	B	P	E1		5.4-8a
F702	1	Pump discharge line pressure instrumentation instrument root valve	2	B	P	E1		5.4-8a
F703	1	Pump discharge line pressure instrumentation instrument root valve	2	B	P	E1		5.4-8a
F704	1	Pump discharge line pressure instrumentation instrument root valve	2	B	P	E1		5.4-8a
F705	1	Pump discharge line pressure instrumentation instrument root valve	2	B	P	E1		5.4-8a
F706	1	Pump discharge line flow instrument root valve	2	B	P	E1		5.4-8a
F707	1	Pump discharge line flow instrument root valve	2	B	P	E1		5.4-8a
F708	1	Pump discharge line flow instrument root valve	2	B	P	E1		5.4-8a
F709	1	Pump discharge line flow instrument root valve	2	B	P	E1		5.4-8a
F710	1	Pump discharge line pressure instrument root valve	2	B	P	E1		5.4-8a
F711	1	Pump discharge line pressure instrument root valve	2	B	P	E1		5.4-8a
F712	1	Turbine accessories cooling water line instrument root valve	2	B	P	E1		5.4-8c
F713	1	Turbine accessories cooling water line instrument root valve	2	B	P	E1		5.4-8c
F714	1	Turbine accessories cooling water line instrument root valve	2	B	P	E1		5.4-8c
F716	1	Steam supply line pressure instrument root valve	2	B	P	E1		5.4-8b
F717	1	Steam supply line pressure instrument root valve	2	B	P	E1		5.4-8b
F718	1	Steam supply line drain pot instrument root valve	2	B	P	E1		5.4-8b
F719	1	Steam supply line drain pot instrument root valve	2	B	P	E1		5.4-8b
F720	1	Steam supply line drain pot instrument root valve	2	B	P	E1		5.4-8b
F721	1	Steam supply line drain pot instrument root valve	2	B	P	E1		5.4-8b
F722	1	Turbine exhaust pressure instrument root valve	2	B	P	E1		5.4-8c

Table 3.9-8 (Continued)

INSERVICE TESTING SAFETY-RELATED PUMPS AND VALVES

E51 Reactor Core Isolation Cooling System Valves (Continued)

No.	Quan	Description	Safety Class (a)	Code Cat. (c)	Valve Func. (d)	Test Para (e)	Test Freq. (f)	SSAR Fig.
F723	1	Turbine exhaust pressure instrument root valve	2	B	P	E1		5.4-8c
F724	1	Turbine exhaust pressure between rupture disk instrument root valve	2	B	P	E1		5.4-8c
F725	1	Turbine exhaust pressure between rupture disk instrument root valve	2	B	P	E1		5.4-8c
D014	1	Turbine exhaust pressure rupture disk	2	D	A	Rplc.	5 yrs	5.4-8c
D015	1	Turbine exhaust pressure rupture disk	2	D	A	Rplc.	5 yrs	5.4-8c

G31 Reactor Water Cleanup System Valves

F001	1	Line inside containment from RHR system maintenance valve	1	B	P	E1		5.4-12a
F002	1	CUW System suction line inboard isolation valve	1	A	I,A	L P,S	2 yrs 3 mo	5.4-12a
F003	1	CUW System suction line outboard isolation valve	1	A	I,A	L P,S	2 yrs 3 mo	5.4-12a
F017	1	CUW System RPV head spray line outboard isolation valve	1	A	I,A	L P,S	2 yrs 3 mo	5.4-12a
F018	1	CUW System RPV head spray line inboard check valve	1	C	I,A	L P,S	2 yrs 3 mo	5.4-12a
F019	1	CUW Sys bottom head drain line maintenance valve	1	B	P	E1		5.4-12a
F050	1	Test line off the suct line outboard isolation valve G31-F003	2	B	P	E1		5.4-12a
F058	1	Test line off RPV head spray line outboard isolation valve	2	B	P	E1		5.4-12a
F060	1	RPV bottom head drain line sample line test line valve	2	B	P	E1		5.4-12a
F070	1	RPV bottom head drain line sample line maintenance valve	2	B	P	E1		5.4-12a
F071	1	RPV bottom head drain line sample line vlv	2	A	I,A	L P,S	2 yrs 3 mo	5.4-12a
F072	1	RPV bottom head drain line sample line vlv	2	A	I,A	L P,S	2 yrs 3 mo	5.4-12a
F500	1	CUW Sys bottom head drain line drain vlv	2	B	P	E1		5.4-12a

3E.2.2.2.2 Material J/T Curve For 420°F

Since the test temperature of 350°F can be considered reasonably close to the 420°F, the test J-R curves for 350°F were used in this case. A review of the test matrix in Table 3E.2-4 shows that three tests were conducted at 350°F. The [illegible] T_{mod} data for all three tests were re[illegible]. The flow stress value used in the tearing modulus calculation was 54 ksi based on Table 3E.2-3. Also reviewed were the data on SA106 carbon steel at 300°F reported by Gudas [14].

Consistent with the trend of the 550°F data, the 350°F weld metal (J-T) data fell below the plate and pipe base metal data. This probably reflects the slightly lower toughness of the SAW weld in the plate. The (J/T) data for the pipe base metal fell between the plate base metal and the plate weld metal. Based on the considerations similar to those presented in the previous section, the pipe base metal J-T data, although they may lie above the weld J-T data, were used for selecting the appropriate (J-T) curve. Accordingly, the curve shown in Figure 3E.2-9 was developed for using the (J-T) methodology in evaluations at 420°F.

3E.2.3 References

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12. ASTM Standard E399, *Plane-Strain Fracture Toughness of Metallic Materials*.

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3E.4.2.3 Crack Opening Area Formulation

The crack opening areas were calculated using LEFM procedures with the customary plastic zone correction. The loadings included in the crack opening area calculations were: pressure, weight and thermal expansion.

The mathematical expressions given by Paris and Tada [7] are used in this case. The crack opening areas for pressure (A_p) and bending stresses (A_b) were separately calculated and then added together to obtain the total area, A_c.

For simplicity, the calculated membrane stresses from weight and thermal expansion loads were combined with the axial membrane stress, σ_p, due to the pressure.

The formulas are summarized below:

$$A_p = \frac{\sigma_p}{E} (2\pi Rt)\, G_p\ (\lambda) \qquad (3E.4\text{-}2)$$

where,

σ_p = axial membrane stress due to pressure, weight and thermal expansion loads.

E = Young's modulus

R = pipe radius

t = pipe thickness

λ = shell parameter = $a/\sqrt{Rt}$

a = half crack length

(3E.4-3)

$$G_p(\lambda) = \lambda^2 + 0.16\,\lambda^4 \quad (o \le \lambda \le 1)$$

$$= 0.02 + 0.81\,\lambda^2 + 0.30\,\lambda^3 + 0.03\,\lambda^4 \quad (1 \le \lambda \le 5)$$

$$A_b = \frac{\sigma_b}{E} \cdot \pi \cdot R^2 \cdot \frac{(3 + \cos\theta)}{4}\, I_t(\theta) \qquad (3E.4\text{-}4)$$

where,

σ_b = bending stress due to weight and thermal expansion loads

θ is half crack angle

(3E.4-5)

$$I_t(\theta) = 2\theta^2 \left[1 + \left(\frac{\theta}{\pi}\right)^{3/2} \left\{8.6 - 13.3\left(\frac{\theta}{\pi}\right) + 24\left(\frac{\theta}{\pi}\right)^2\right\} + \left(\frac{\theta}{\pi}\right)^3 \left\{22.5 - 75\left(\frac{\theta}{\pi}\right) + 205.7\left(\frac{\theta}{\pi}\right)^2 - 247.5\left(\frac{\theta}{\pi}\right)^3 + 242\left(\frac{\theta}{\pi}\right)^4\right\}\right]$$

$(0 < \theta < 100^0)$

The plastic zone correction was incorporated by replacing a and θ in these formulas by a_e and θ_e which are given by

$$\theta_{eff} = \theta + \frac{K_{total}^2}{2\pi R \sigma_Y^2} \qquad (3E.4\text{-}6)$$

$$a_e = \theta_e \cdot R$$

The yield stress, σ_y, was conservatively assumed as the average of the code specified yield and ultimate strength. The stress intensity factor, K_{total}, includes contribution due to both the membrane and bending stress and is determined as follows:

$$K_{total} = K_m + K_b \qquad (3E.4\text{-}7)$$

where,

$$K_m = \sigma_p \sqrt{a} \cdot F_p(\lambda)$$

$$F_p(\lambda) = (1 + 0.3225\lambda^2)^{\frac{1}{2}}$$

$$= 0.9 + 0.25\lambda \qquad (0 \leq \lambda \leq 1)$$
$$(1 \leq \lambda \leq 5)$$

$$K_b = \sigma_b \cdot \sqrt{\pi a} \cdot F_b(\theta)$$

$$F_b(\theta) = 1 + 6.8\left(\frac{\theta}{\pi}\right)^{3/2}$$
$$- 13.6\left(\frac{\theta}{\pi}\right)^{5/2} + 20\left(\frac{\theta}{\pi}\right)^{7/2}$$
$$(0 \leq \theta \leq 100^{\circ})$$

The steam mass flow rate, M, shown in Table 3E.4-1 is a function of parameter, ft/2δ. Once the mass flow rate is determined corresponding to the calculated value of this parameter, the leak rate in gpm can then be calculated.

3E.4.3 References

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SECTION 3H.3

TABLES

SECTION 3H.3

TABLES (Continued)

ILLUSTRATIONS

3H.3 Fugures and Tables Supporting Section 3.8

The following figures and tables present the details of a structural evaluation of the reactor building, containment and internal structures as presented in Section 3.8

References to section numbers in the following Tables correspond to those in Figure 3.8-14.

Table 3H.3-18

FORCES AND MOMENTS IN THE REACTOR BUILDING EXTERIOR WALL

AT SECTIONS 10, 11, AND 12

SECTION	ELEMENT	LOCATION	LOADING	Fx (k/ft)	Fy (k/ft)	Fxy (k/ft)	Mx (k-ft/ft)	My (k-ft/ft)	Mxy (k-ft/ft)	Fz_1 (k/ft)	Fz_2 (k/ft)
10	601	180°	$D+L_1+Pt^{(52/52)}$	-6.4	-130.9	8.0	14.4	60.4	2.5	0.3	-5.8
			$D+L_1+Pa+CO$ (10 min)	-7.8	-121.9	7.6	14.6	62.4	2.3	0.4	-6.4
			$D+L_1+Pa+CO+SRV$ (6hrs)	-5.8	-125.4	8.1	13.5	57.5	2.1	0.3	-5.7
			SSE	100.9	181.0	158.0	29.0	114.7	3.7	0.7	17.6
			Ta (10min)	94.8	2.4	-5.0	-84.7	-143.2	-1.6	-2.0	8.1
			Ta (6hrs)	110.0	9.5	-6.3	-90.2	-169.5	-2.2	-2.6	11.4
			Soil Pressure (L_2)	0.0	0.0	0.0	0.0	377.0	0.0	0.0	106.0
11	681	180°	$D+L_1+Pt^{(52/52)}$	7.5	-122.7	3.2	3.2	11.0	-1.6	1.5	-1.8
			$D+L_1+Pa+CO$ (10min)	3.0	-155.1	3.9	2.1	8.6	-1.8	1.5	-2.5
			$D+L_1+Pa+CO+SRV$ (6hrs)	2.4	-119.3	4.2	2.5	10.3	-1.7	1.5	-2.9
			SSE	33.6	191.5	214.0	1.3	12.9	7.5	2.2	1.5
			Ta (10min)	18.1	12.9	-1.5	-57.1	-88.8	4.1	-3.2	5.2
			Ta (6hrs)	34.6	25.4	-2.3	-56.9	-95.1	5.7	-4.2	6.8
			Soil Pressure (L_2)	0.0	0.0	0.0	0.0	-136.0	0.0	0.0	0.0
12	841	180°	$D+L+Pt^{(52/52)}$	13.4	-97.4	-4.4	-0.7	-23.9	-0.7	-3.3	3.2
			D+L+Pa+CO (10min)	9.9	-91.8	-2.4	-0.2	-18.5	-0.6	-3.6	3.2
			D+L+Pa+CO=SRV (6hrs)	7.3	-95.7	-1.3	-0.3	-17.5	-0.7	-4.8	4.4
			SSE	34.1	178.8	137.0	10.3	23.0	3.2	4.3	4.7
			Ta (10min)	544.8	168.0	32.1	405.5	300.3	-4.1	-1.1	24.9
			Ta (6hrs)	578.8	177.2	24.6	415.3	328.8	-2.4	2.7	3.7
			Soil Pressure (L_2)	0.0	0.0	0.0	0.0	83.0	0.0	0.0	51.0

Note: For exterior walls live load. $L = L_1 + L_2$

Table 3H.3-19

REBAR AND CONCRETE STRESS IN THE REACTOR BUILDING EXTERIOR WALL

AT SECTIONS 10, 11 AND 12

LOAD COMB	ELEMENT NUMBER	AZIMUTH (degrees)	REINFORCING STEEL CALCULATED STRESSES (ksi) INSIDE FACE VERT.	INSIDE FACE HORIZ	OUTSIDE FACE VERT.	OUTSIDE FACE HORIZ.	SHEAR TIES	CONCRETE ALLOWABLE STRESS (ksi)	CONCRETE CALCULATED STRESS (ksi)	CONCRETE ALLOWABLE STRESS (ksi)
Section: 10										
Location: REACTOR BUILDING EXTERIOR WALL NEAR THE FOUNDATION MAT										
15	601	180	8.21	18.67	22.64	17.54	16.23	54.0	-0.46	-3.4
15a, 15b	601	180	16.39	16.96	13.80	21.12	16.76	54.0	-0.26	-3.4
Sections: 11										
Location: REACTOR BUILDING EXTERIOR WALL AT EL. -16.7FT										
15	681	180	49.28	34.28	23.26	27.05	6.44	54.0	-0.82	-3.4
15a, 15b	681	180	44.99	34.46	25.35	27.22	7.52	54.0	-0.81	-3.4
Section: 12										
Location: REACTOR BUILDING EXTERIOR WALL NEAR GRADE										
15	841	180	31.88	32.93	39.85	47.02	22.15	54.0	-1.11	-3.4
15a, 15b	841	180	15.72	6.28	47.86	44.67	15.80	54.0	-0.76	-3.4

Table 3H.3-20

FORCES AND MOMENTS IN THE REACTOR BUILDING FLOOR SLABS

AT SECTIONS 18, 19, AND 40)

SECTION	ELEMENT	LOCATION	LOADING	Fx (k/ft)	Fy (k/ft)	Fxy (k/ft)	Mx (k-ft/ft)	My (k-ft/ft)	Mxy (k-ft/ft)	Fz_1 (k/ft)	Fz_2 (k/ft)
18	2386	180°	D+L+Pt(52/52)	-58.2	70.8	-20.9	-6.2	-3.8	-5.1	-7.6	6.6
			D+L+1.5 (Pa+CO)+1.25 SRV(6hrs)	-60.6	67.2	-21.5	-19.1	-8.6	-6.5	-12.2	5.7
			D+L+Pa+CO+SRV (6hrs)	-43.1	51.5	-16.0	-7.6	-5.2	-4.5	3.2	5.3
			SSE	29.2	30.7	7.2	173.6	44.8	20.6	56.1	3.9
			Ta (10min)	-67.6	149.7	-23.7	12.8	4.5	2.7	9.3	-0.5
			Ta (6hrs)	-86.1	177.2	-29.7	6.6	3.6	2.3	8.5	-1.3
19	2686	180°	D+L+Pt(52/52)	-37.4	53.4	-12.7	10.5	2.1	0.3	6.9	2.9
			D+L+1.5 (Pa+CO)+1.25 SRV(6hrs)	-34.3	46.8	-11.6	7.0	1.3	0.1	5.6	2.3
			D+L+Pa+CO+SRV (6hrs)	-20.2	35.6	-8.4	6.9	1.2	0.1	5.0	2.2
			SSE	42.6	17.9	6.0	30.5	7.1	3.4	11.3	1.3
			Ta (10min)	-64.6	72.9	-16.8	-7.8	-1.1	-0.2	-4.9	-2.3
			Ta (6hrs)	-90.8	97.2	-22.1	-13.5	-2.0	-0.6	-7.3	-3.4
40	2986	180°	D+L+Pt(52/52)	-26.9	72.4	-10.8	100.9	23.9	9.0	14.4	8.5
			D+L+1.5 (Pa+CO) (10min)	-31.7	80.1	-12.7	100.2	21.9	8.6	14.3	8.9
			D+L+1.5 (Pa+CO)+1.25 SRV(6hrs)	-20.6	68.3	-10.4	76.4	16.8	6.9	10.1	6.3
			D+L+Pa+CO+SRV (6hrs)	-11.6	50.8	-7.2	59.4	13.8	6.1	6.7	4.3
			SSE	37.6	55.3	26.2	249.2	58.8	23.5	137.6	-88.1
			Ta (10min)	-187.7	314.4	-77.2	36.3	7.9	-3.3	16.9	11.5
			Ta (6hrs)	-317.1	461.5	-114.5	111.9	25.4	1.3	30.4	19.9

Table 3H.3-21

REBAR AND CONCRETE STRESS IN THE REACTOR BUILDING SLABS

AT SECTIONS 18, 19 AND 40

LOAD COMB	ELEMENT NUMBER	AZIMUTH (degrees)	REINFORCING STEEL CALCULATED STRESSES (ksi) TOP FACE X DIR.	TOP FACE Y DIR.	BOTTOM FACE X DIR.	BOTTOM FACE Y DIR.	SHEAR TIES	REINFORCING STEEL ALLOWABLE STRESS (ksi)	CONCRETE CALCULATED STRESS (ksi)	CONCRETE ALLOWABLE STRESS (ksi)
Section: 18										
Location: REACTOR BUILDING FLOOR SLAB AT EL. -21.98 FT NEAR THE CONTAINMENT WALL										
1	2386	180	38.52	4.10	37.94	4.02	20.48	45.0	-0.21	-2.4
8A, 8B	2386	180	17.36	-1.60	15.38	-1.30	13.01	54.0	-0.37	-3.4
Section: 19										
Location: REACTOR BUILDING FLOOR SLAB AT EL. -0.67 FT NEAR CONTAINMENT WALL										
1	2686	180	14.33	-1.29	16.66	4.16	0.77	45.0	-0.36	-2.4
8a, 8b	2686	180	34.30	-2.55	34.80	-0.78	1.10	54.0	-0.67	-3.4
Section: 40										
Location: REACTOR BUILDING FLOOR SLAB AT EL. 23.95 FT NEAR CONTAINMENT WALL										
1	2986	180	16.04	-5.35	19.84	-1.92	11.16	45.0	-0.72	-2.4
8	2986	180	29.08	-2.32	37.40	11.98	6.58	54.0	-0.83	-3.4

Note: x is the direction parallel to the R/B fuel pool girders.

y direction is normal to R/B fuel pool girders.

3H.4 References

1. Bechtel Topical Report BC-TOP-4, August 1980, *Seismic Analysis of Structures and Equipment for Nuclear Power Plants.*

2. Tseng, W.S., D.D., *Simplified Method of Predicting Seismic Basemat Uplift of Nuclear Power Plant Structures*, Transactions of the 6th International Conference on SMIRT, Paris France, August 1981.

4. REACTOR

4.1 SUMMARY DESCRIPTION

The reactor assembly consists of the reactor pressure vessel, pressure containing appurtenances including CRD housings, in-core instrumentation housing and the head vent and spray assembly plus the reactor internal components described in Subsection 4.1.2. Figure 5.3-2 shows the arrangement of the reactor assembly components. A summary of the important design and performance characteristics is given in Subsection 1.3.1.1. Loading conditions for reactor assembly components are specified in Subsection 3.9.5.2.

For the purpose of this SSAR, a typical fuel and control rod design and core loading pattern was used as the basis for the system response studies in Section 6.3 and Chapter 15. The actual fuel and control rod designs and core loading pattern to be used at a plant will either have been approved or will meet criteria approved by the USNRC, and will be provided to the USNRC for information. The fuel and control blade design and core loading pattern used for the system response studies are documented in this chapter; information to be provided by the utility referencing the ABWR design is contained in the interface subsection of each of these sections.

4.1.1 Reactor Pressure Vessel

The reactor pressure vessel includes the reactor internal pump (RIP) casing and flow restrictors in each of the steam outlet nozzles and the shroud support and pump deck which form the partition between the RIP suction and discharge. The reactor pressure vessel design and description are covered in Section 5.3.

4.1.2 Reactor Internal Components

The major reactor internal components are the core (fuel, channels, control blades and instrumentation), the core support structure (including the shroud, top guide and core plate), the shroud head and steam separator assembly, the steam dryer assembly, the feedwater spargers, the core spray and core flooding spargers. Except for the Zircaloy in the reactor core, these reactor internals are stainless steel or other corrosion-resistant alloys. The fuel assemblies (including fuel rods and channel), control blades, shroud head and steam separator assembly, and steam dryers and in-core instrumentation dry tubes are removable when the reactor vessel is opened for refueling or maintenance.

4.1.2.1 Reactor Core

Important features of the reactor core are:

(1) The bottom-entry cruciform control rods. Rods of this design were first introduced in the Dresden-1 reactor in April 1961 and have accumulated thousands of hours of service.

(2) Fixed in-core fission chambers (LPRMs) provide continuous local power range neutron flux monitoring. A guide tube in each in-core assembly provides for a traversing ion chamber (TIP) for calibration and axial detail. Start-up range neutron monitors (SRNMs) are located at fixed locations between the (LPRMs) as shown on Figure 4.1-1. The in-core location of the start-up and source range instruments provides coverage of the large reactor core and provides an acceptable signal-to-noise ratio and neutron-to-gamma ratio. All in-core instrument leads enter from the bottom and the instruments are in service during refueling. In-core instrumentation is presented in Subsection 7.6.1.

(3) As shown by experience obtained at Dresden-1 and all other BWR plants, utilizing the in-core flux monitor system, the desired power distribution can be maintained within a large core by proper control rod scheduling.

(4) The fuel channels provide a fixed flow path for the boiling coolant, serve as a guiding surface for the control rods and protect the fuel during handling operations.

(5) The mechanical reactivity control permits

criticality checks during refueling and provides maximum plant safety. The core is designed to be subcritical at any time in it's operating history with any one control rod fully withdrawn and the other control rods full inserted.

(6) The selected control rod pitch represents a practical value of individual control rod reactivity worth, and allows adequate clearance below the pressure vessel between CRD mechanisms for ease of maintenance and removal.

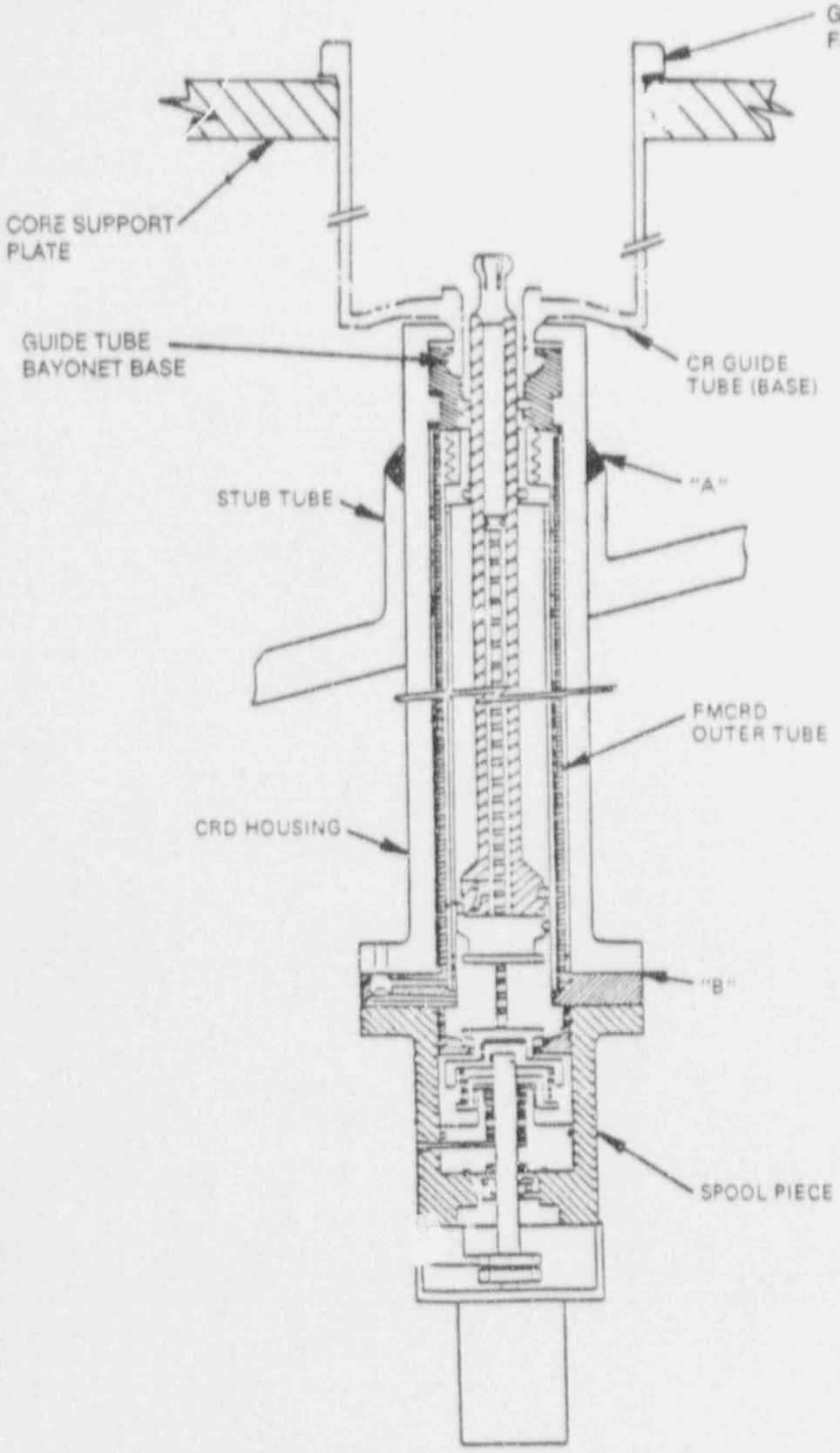

Figure 4.6-7 INTERNAL BLOWOUT SUPPORT SCHEMATIC

GE PROPRIETARY - provided under separate cover

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(1) those components where, in the event of postulated failure of the component during normal reactor operation, the reactor can be shut down and cooled down in an orderly manner, assuming makeup is provided by the reactor coolant makeup system only; and

(2) components which are or can be isolated from the reactor coolant system by two valves (both closed, both open, or one closed and one open). Each such open valve is capable of automatic actuation and if the other valve is open its closure time is such that, in the event of postulated failure of the component during normal reactor operation, each valve remains operable and the reactor can be shut down and cooled down in an orderly manner assuming makeup is provided by the reactor coolant makeup system only.

5.2.4.2 Accessibility

All items within the Class 1 boundary are designed, to the extent practicable, to provide access for the examinations required by ASME Section XI, IWB-2500. Items for which the design is known to have inherent access restrictions are described in Subsection 5.2.4.8.

5.2.4.2.1 Reactor Pressure Vessel Access

Access for examinations of the reactor pressure vessel (RPV) is incorporated into the design of the vessel, biological shield wall and vessel insulation as follows:

(1) RPV Welds Below the Top Biological Shield Wall

The shield wall and vessel insulation behind the shield wall are spaced away from the RPV outside surface to provide access for remotely operated ultrasonic examination devices as described in Subsection 5.2.4.3.2.1 Access for the insertion of automated devices is provided through removable insulation panels at the top of the shield wall and at access ports at reactor vessel nozzles. Platforms are attached to the bioshield wall to provide access for installation of remotely operated nozzle examination devices.

(2) RPV Welds Above Top of the Biological Shield Wall

Access to the reactor pressure vessel welds above the top of the biological shield wall is provided by removable insulation panels. This design provides reasonable access for both automated as well as manual ultrasonic examination.

(3) Closure Head, RPV Studs, Nuts and Washers

The closure head is dry stored during refueling. Removable insulation is designed to provide access for manual ultrasonic examinations of closure head welds. RPV nuts and washers are dry stored and are accessible for surface and visual (VT-1) examination. RPV studs may be volumetrically examined in place or when removed.

(4) Bottom Head Welds

Access to the bottom head to shell weld and bottom head seam welds is provided through openings in the RPV support pedestal and removable insulation panels around the cylindrical lower portion of the vessel. This design provides access for manual or automated ultrasonic examination equipment. Sufficient access is provided to partial penetration nozzle welds, i.e., CRD penetrations, instrumentation nozzles and recirculation internal pump penetration welds, for performance of the visual, VT-2, examination during the system leakage and system hydrostatic examinations.

(5) Reactor Vessel Support Skirt

The integral attachment weld from the number four shell course forging to the RPV skirt will be examined ultrasonically. Sufficient access is provided for either manual or automated ultrasonic examination. Access is provided to the balance of the support skirt for performance of visual, VT-3, examination.

5.2.4.2.2 Piping, Pumps Valves and Supports

Physical arrangement of piping pumps and valves provide personnel access to each weld location for performance of ultrasonic and surface (magnetic particle or liquid penetrant) examinations and sufficient access to supports for performance of visual, VT-3, examination. Working platforms are provided in some

areas to facilitate servicing of pumps and valves. Platforms and ladders are provided for access to piping welds including the pipe-to-reactor vessel nozzle welds. Removable thermal insulation is provided on welds and components which require frequent access for examination or are located in high radiation areas. Welds are located to permit ultrasonic examination from at least one side, but where component geometries permit, access from both sides is provided.

Restrictions: For piping systems and portions of piping systems subject to volumetric and surface examination, the following piping designs are not used:

(1) Valve to Valve
(2) Valve to Reducer
(3) Valve to Tee
(4) Elbow to Elbow
(5) Elbow to Tee
(6) Nozzle to elbow
(7) Reducer to elbow
(8) Tee to tee
(9) Pump to valve

Straight sections of pipe and spool pieces shall be added between fittings. The minimum length of the spool piece has been determined by using the formulate L = 2T + 152mm, where L equals the length of the spool piece (not including weld preparation) and T equals the pipe wall thickness.

5.2.4.3 Examination Categories and Methods

5.2.4.3.1 Examination Categories

The examination category of each item is listed in Table 5.2-8. The items are listed by system and line number where applicable. Table 5.2-8 also states the method of examination for each item. The preservice and inservice examination plans will be supplemented with detailed drawings showing the examination areas, such as Figures 5.2-7a and 5.2-7b.

5.2.4.3.2 Examination Methods

5.2.4.3.2.1 Ultrasonic Examination of the Reactor Vessel

Ultrasonic examination of the RPV will be conducted in accordance with ASME Section XI, IWA-2232 (a), and Section V, Article 4. In addition the ultrasonic examination system shall meet the requirements of Regulatory Guide 1.150 as described in Table 5.2-9. RPV welds and nozzles subject to examination are shown in Figure 5.2-7a.

The GE reactor vessel inspection system (GERIS) meets the detection and sizing requirements of Regulatory Guide 1.150, as cited in Table 5.2-9. Inner radius examinations are performed from the outside of the nozzle using several compound angle transducer wedges to obtain complete coverage of the required examination volume. Electronic gating used in GERIS system records up to 8 different reflectors simultaneously to assure that all relevant indications are recorded. Appendix 5A demonstarted compliance with Regulatory Guide 1.150.

5.2.4.3.2.2 Visual Examination

Visual examination methods, VT-1, VT-2 and VT-3, shall be conducted in accordance with ASME Section XI, IWA-2210. In addition, VT-2 examinations shall meet the requirements of IWA-5240.

Direct visual, VT-1, examinations shall be conducted with sufficient lighting to resolve a 0.8mm black line on an 18% neutral grey card. Where direct visual, VT-1, examinations are conducted without the use of mirrors or with other viewing aids, clearance (of at least 610mm of clear space) is provided where feasible for the head and shoulders of a man within a working arm's length (508mm) of the surface to be examined.

At locations where leakages are normally expected and leakage collection systems are located, (e.g., valve stems and pump seals), the visual, VT-2, examination shall verify that the leakage collection system is operative.

Piping runs shall be clearly identified and laid out such that insulation damage, leaks and structural distress will be evident to a trained visual examiner.

5.2.4.3.2.3 Surface Examination

Magnetic particle and liquid penetrant examination techniques shall be performed in accordance with ASME Section XI, IWA-2221 and IWA-2222, respectively. Direct examination access for magnetic particle (MT) and penetrant (PT) examination is the same as that required for direct visual (VT-1) examination (Subsection 5.2.4.3.2.3), except that

unidentified leakage rate might be emitted from a single crack in the nuclear system process barrier.

An allowance for leakage that does not compromise barrier integrity and is not identifiable is established for normal plant operation.

The unidentified leakage rate limit is established at 3.785 liters/min to allow time for corrective action before the process barrier could be significantly compromised. This unidentified leakage rate is a small fraction of the calculated flow from a critical crack in a primary system pipe (Appendix 3E).

5.2.5.5.2 Margins of Safety

The margins of safety for a detectable flaw to reach critical size are presented in Subsection 5.2.5.5.3. Figure 5.2-8 shows general relationships between crack length, leak rate, stress, and linesize using mathematical models.

5.2.5.5.3 Criteria to Evaluate the Adequacy and Margin of Leak Detection System

For process lines that are normally open, there are at least two different methods of detecting abnormal leakage from each system comprising the nuclear system process barrier, located both inside the primary containment (drywell) and external to the drywell, in the reactor building the steam tunnel and the turbine building (Tables 5.2-6 and 5.2-7). The instrumentation is designed so it can be set to provide alarms at established leakage rate limits and isolate the affected system if necessary. The alarm points are determined analytically or based on measurements of appropriate parameters made during startup and preoperational tests.

The unidentified leakage rate limit is based, with an adequate margin for contingencies, on the crack size large enough to propagate rapidly.

The established limit is sufficiently low so that, even if the entire unidentified leakage rate were coming from a single crack in the nuclear system process barrier, corrective action could be taken before the integrity of the barrier would be threatened.

The leak detection system will satisfactorily detect unidentified leakage of 3.785 liters/min within the drywell.

5.2.5.6 Differentiation Between Identified and Unidentified Leaks

Subsection 5.2.5.1 describes the leak detection methods utilized by the leak detection system. The ability of the leak detection system to differentiate between identified and unidentified leakage is discussed in Subsections 5.2.5.4 and 5.2.5.5.

5.2.5.7 Sensitivity and Operability Tests

Sensitivity, including sensitivity tests and response time of the leak detection system, and the criteria for shutdown if leakage limits are exceeded are covered in Subsections 5.2.5.1.1, 5.2.5.1.2, 5.2.5.2.1(1) and 7.3.1.1.2.

Testability of the LDS is contained in Subsection 7.3.1.1.2(10).

5.2.5.8 Testing and Calibration

Provisions for testing and calibration of the leak detection and isolation system are covered in Chapter 14.

5.2.5.9 Regulatory Guide 1.45: Compliance

These guidelines are prescribed to assure that leakage detection and collection systems provide maximum practical identification of leaks from the RCPB.

Leakage is separated into identified and unidentified categories and each is independently monitored, thus meeting Position C.1 requirements.

Leakage from unidentified sources from inside the drywell is collected into the floor drain sump and monitored with an accuracy better than

3.785 liters/min thus meeting Position C.2 requirements.

By monitoring (1) floor drain sump fillup and pumpout rate, (2) airborne particulates, and (3) air coolers condensate flow rate, Position C.3 is satisfied.

Monitoring of the reactor building cooling water heat exchanger coolant return lines for radiation due to leaks within the RHR, RIP and CUW and the fuel pool cooling system heat exchangers satisfies Position C.4. For system detail, see Subsection 7.6.1.2.

The floor drain sump monitoring, air particulates monitoring, and air cooler condensate monitoring are designed to detect leakage rates of 3.785 liters/min within one hour, thus meeting Position C.5 requirements.

The fission products monitoring subsystem is qualified for SSE. The containment floor drain sump monitor, air cooler, and condensate flow meter are qualified for OBE, thus meeting Position C.6 requirements.

Leak detection indicators and alarms are provided in the main control room. This satisfies Position C.7 requirements. Procedures and graphs will be provided by the applicant to plant operators for converting the various indicators to a common leakage equivalent, when necessary, thus satisfying the remainder of Position C.7 (See interface requirements Subsection 5.2.6.2). The leakage detection system is equipped with provisions to permit testing for operability and calibration during the plant operation using the following methods:

(1) simulation of trip signal;

(2) comparing channel to channel of the same leak detection method (i.e., area temperature monitoring);

(3) operability checked by comparing one method versus another (i.e., sump fillup rate versus pumpout rate and particulate monitoring or air cooler condensate flow versus sump fillup rate); and

(4) continuous monitoring of floor drain sump level, and a source of water for calibration and testing is provided.

These satisfy Position C.8 requirements.

Limiting unidentified leakage to the 3.785 liters/min and identified to 95 liters/min satisfies Position C.9.

5.2.6 Interfaces

5.2.6.1 Conversion of Indications

Procedures and graphs will be provided to operations for converting the various indicators into a common leakage equivalent (See Subsection 5.6.5.9).

5.2.7 References

1. (Deleted)

2. (Deleted)

3. D.A. Hale, *The Effect of BWR Startup Environments on Crack Growth in Structural Alloys*, Trans. of ASME, vol 108, January 1986.

4. F.P. Ford and M. J. Povich, *The Effect of Oxygen/Temperature Combinations on the Stress Corrosion Susceptibility of Sensitized T-304 Stainless Steel in High Purity Water*, Paper 94 presented at Corrosion 79, Atlanta, GA, March 1979.

5. *BWR Normal Water Chemistry Guidelines: 1986 Revision*, EPRI NP-4946-SR, July 1988.

6. B.M. Gordon, *The Effect of Chloride and Oxygen on the Stress Corrosion Cracking of Stainless Steels: Review of Literature*, Material Performance, NACE, Vol. 19, No. 4, April 1980.

7. W.J. Shack, et al, *Environmentally Assisted Cracking in Light Water Reactors: Annual Report, October 1983 - September 1984*,

Table 5.2-1

REACTOR COOLANT PRESSURE BOUNDARY COMPONENTS

APPLICABLE CODE CASES

Number	Title	Applicable Equipment	Remarks
N-71-15	(1)	Component Support	Accepted per RG 1.85
N-122	(2)	Piping	Accepted per RG 1.84
N-247	(3)	Component Support	Accepted per RG 1.84
N-249-9	(4)	Component Support	Conditionally Accepted per RG 1.85
N-309-1	(5)	Component Support	Accepted per RG 1.84
N-313	(6)	Piping	Accepted per RG 1.84
N-316	(7)	Piping	Accepted per RG 1.84
N-318-3	(8)	Piping	Conditionally Accepted per RG 1.84
N-319	(9)	Piping	Accepted per RG 1.84
N-391	(10)	Piping	Accepted per RG 1.84
N-392	(11)	Piping	Accepted per RG 1.84
N-393	(12)	Piping	Accepted per RG 1.84
N-411-1	(13)	Piping	Conditionally Accepted per RG 1.84
N-414	(14)	Component Support	Accepted per RG 1.84
N-430	(15)	Component Support	Accepted per RG 1.84
N-236-1	(16)	Containment	Conditionally Accepted Per RG 1.147
N-307-1	(17)	RPV Studs	Accepted per RG 1.147

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Table 5.2-1

REACTOR COOLANT PRESSURE BOUNDARY COMPONENTS

APPLICABLE CODE CASES (Continued)

Number	Title	Applicable Equipment	Remarks
(Deleted)	(18)		
(Deleted)	(19)		
N-416	(20)	Piping	Accepted Per RG 1.147
N-432	(21)	Class 1 Components	Accepted Per RG 1.147
N-435-1	(22)	Class 2 Vessels	Accepted Per RG 1.147
N-457	(23)	Bolts and Studs	Accepted Per RG 1.147
N-463	(24)	Piping	Accepted Per RG 1.147
N-460	(25)	Class 1 & 2 Components and Piping	Accepted Per RG 1.147
N-472	(26)	Pumps	Accepted Per RG 1.147
N-479	(27)	Main Steam System	Not Listed in RG 1.147
N-491	(28)	Component Supports	Not Listed in RG 1.147

Table 5.2-1

REACTOR COOLANT PRESSURE BOUNDARY COMPONENTS

APPLICABLE CODE CASES (Continued)

(1) *Additional Materials for Subsection NF, Classes 1, 2, 3 and MC Component Supports Fabricated by Welding, Section III, Division 1.*

(2) *Stress Indices for Structure Attachments, Class 1, Section III, Division 1.*

(3) *Certified Design Report Summary for Component Standard Supports, Section III, Division 1, Class 1, 2, 3 and MC.*

(4) *Additional Material for Subsection NF, Classes 1, 2, 3 and MC Component Supports Fabricated Without Welding, Section III, Division 1.*

(5) *Identification of Materials for Component Supports, Section III, Division 1.*

(6) *Alternate Rules for Half-Coupling Branch Connections, Section III, Division 1.*

(7) *Alternate Rules for Fillet Weld Dimensions for Socket Welded Fittings, Section III, Division 1, Class 1, 2, 3.*

(8) *Procedure for Evaluation of the Design of Rectangular Cross Section Attachments on Class 2 or 3 Piping, Section III, Division 1.*

(9) *Alternate Procedure for Evaluation of Stress in Butt Weld Elbows in Class 1 Piping, Section III, Division 1.*

(10) *Procedure for Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Class 1 Piping, Section III, Division 1.*

(11) *Procedure for Evaluation of the Design of Hollow Circular Cross Section Welded Attachments on Classes 2 and 3 Piping, Section III, Division 1.*

(12) *Repair Welding Structural Steel Rolled Shapes and Plates for Component Supports, Section III, Division 1.*

(13) *Alternative Damping Values for Seismic Analysis of Classes 1, 2, 3 Piping Sections, Section III, Division 1.*

(14) *Tack Welds for Class 1, 2, 3 and MC Components and Piping Supports.*

(15) *Requirements for Welding Workmanship and Visual Acceptance Criteria for Class 1, 2, 3 and MC Linear-Type and Standard Supports.*

(16) *Repair and Replacement of Class MC Vessels*

(17) *Revised Examination Volume for Class 1 Bolting, Table IWB-2500-1, Examination Category B-G-1, When the Examinations Are Conducted from the Drilled Hole*

(18) *(Deleted)*

(19) *(Deleted)*

(20) *Alternative Rules for Hydrostatic Testing of Repair or Replacement of Class 2 Piping*

(21) *Repair Welding Using Automatic Or Machine Gas Tungsten-Arc Welding (GTAW) Temperbead Technique*

(22) *Alternative Examination Requirements for Vessels With Wall Thicknesses 2 in. or Less*

(23) *Qualification Specimen Notch Location for Ultrasonic Examination of Bolts and Studs*

(24) *Evaluation Procedures and Acceptance Criteria for Flaws in Class 1 Ferritic Piping That Exceed the Acceptance Standards of IWB-3514-2*

Table 5.2-1

REACTOR COOLANT PRESSURE BOUNDARY COMPONENTS

APPLICABLE CODE CASES (Continued)

(25) *Alternate Examination Coverage for Class 1 and 2 Welds*

(26) *Use of Digital Readout and Digital Measurement Devices for Performing Pump Vibration Testing.*

(27) *Boiling Water Reactor (BWR) Main Steam Hydrostatic Test*

(28) *Alternate Rules for Examination of Class 1, 2, 3 and MC Component Supports of Light Water Cooled Power Plants.*

Table 5.2-3

NUCLEAR SYSTEM SAFETY/RELIEF VALVE SETPOINTS

Set Pressures and Capacities

Number * of Valves	Spring Set Pressure (kg/cm^2g)	ASME Rated Capacity at 103% Spring Set Pressure (kg/hr each)	Relief Pressure Set Pressure (kg/cm^2g)
1	80.8	395,000	76.6
1	80.8	395,000	77.3
4	81.5	399,000	78.0
4	82.2	402,000	78.7
4	82.9	406,000	79.4
4	83.6	409,000	80.1

* *Eight of the SRVs serve in the automatic depressurization function.*

Table 5.2-4

REACTOR COOLANT PRESSURE BOUNDARY MATERIALS

Component	Form	Material	Specification (ASTM/ASME)
Main Steam Isolation Valves			
Valve Body	Cast	Carbon steel	SA352 LCB
Cover	Forged	Carbon Steel	SA350LF2
Poppet	Forged	Carbon Steel	SA350LF2
Valve stem	Rod	17-4 pH	SA 564 630 (H1100)
Body bolt	Bolting	Alloy steel	SA 540 B23 CL4 or 5
Hex nuts	Bolting Nuts	Alloy steel	SA 194 GR7
Main Steam Safety/Relief Valve			
Body	Forging	Carbon steel	ASME SA 350 LF2
	or Casting	Carbon steel	ASME SA 352 LCB
Bonnet (yoke)	Forging	Carbon steel	ASME SA 350 LF2
	or Casting	Carbon steel	ASME SA 352 LCB
Nozzle (seat)	Forging	Stainless steel	ASME SA 182 Gr F316
	or Casting	Carbon steel	ASME SA 350 LF2
Body to bonnet stud	Bar/rod	Low-alloy steel	ASME SA 193 Gr B7
Body to bonnet nut	Bar/rod	Alloy steel	ASME SA 194 Gr 7
Disc	Forging	Alloy steel	ASME SA 637 Gr 718
	or Casting	Stainless steel	ASME SA 351 CF 3A
Spring washer &	Forging	Carbon steel	ASME SA 105
Adjusting Screw or		Alloy steel	ASME SA 193 Gr B6 (Quenched + tempered or normalized & tempered
Set point adjustment assembly	Forgings	Carbon and alloy steel parts	Multiple specifications
Spindle (stem)	Bar	Precipitation-hardened steel	ASTM A564 Type 630 (H 1100)
Spring	Wire or	Steel	ASTM A304 Gr 4161 N
	Bellville washers	Alloy steel	45 Cr Mo V67
Main Steam Piping			
Pipe	Seamless	Carbon steel	SA 333 Gr. 6
Contour nozzle	Forging	Carbon steel	SA 350 LF 2
200A 1500# large groove flange	Forging	Carbon steel	SA 350 LF 2

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depressurization systems perform adequate core cooling to prevent excessive fuel clad temperature during LOCA event. Detailed discussion of RCIC meeting this GDC is described in Subsection 3.1.2.

Compliance with GDC 36. The RCIC system is designed such that in-service inspection of the system and its components is carried out in accordance with the intent of ASME Section XI. The RCIC design specification requires layout and arrangement of the containment penetrations, process piping, valves, and other critical equipment outside the reactor vessel, to the maximum practical extent, permit access by personnel and/or appropriate equipment for testing and inspection of system integrity.

Compliance with GDC 37. The RCIC system is designed such that system and its components can be periodically tested to verify operability. Systems operability is demonstrated by preoperational and periodic testings in accordance with RG 1.68. Preoperational test will ensure proper functioning of controls, instrumentation, pumps and valves. Periodic testings confirm systems availability and operability through out the life of the plant. During normal plant operation, a full flow pump test is being performed periodically to assure systems design flow and head requirements are attained. All RCIC systems components are capable of individual functional testings during plant operation. This includes sensors, instrumentation, control logics, pump, valves, and more. Should the need for RCIC operation occur while the system is being tested, the RCIC system and its components will automatically re-aligned to provide cooling water into the reactor. The above test requirements satisfy GDC 37.

5.4.6.1 Design Basis

The reactor core isolation cooling (RCIC) system is a safety system which consists of a turbine, pump, piping, valves, accessories, and instrumentation designed to assure that sufficient reactor water inventory is maintained in the reactor vessel to permit adequate core cooling to take place. This prevents reactor fuel overheating during the following conditions:

(1) a loss-of-coolant (LOCA) event;

(2) vessel isolated and maintained at hot standby;

(3) vessel isolated and accompanied by loss of coolant flow from the reactor feedwater system;

(4) complete plant shutdown with loss of normal feedwater before the reactor is depressurized to a level where the shutdown cooling system can be placed in operation; or

(5) loss of AC power.

The RCIC system is designed to perform its function without AC power for at least 8 hours. Supporting systems such as DC power and the water supply will support the RCIC system during this time period. Without AC power, RCIC room cooling will not be available. However, room temperature will not reach the equipment maximum environmental temperature within 8 hours. (also see Subsection 19E.2.1.2.2 for additional information)

During loss of AC power, RCIC when started at water level 2 is capable of preventing water level from dropping below the level which ADS mitigates (Level 1). This accounts for decay heat boil-off and primary system leakages.

Following a reactor scram, steam generation will continue at a reduced rate due to the core fission product decay heat. At this time the turbine bypass system will divert the steam to

directly into the reactor pressure vessel to the drywell spray header degraded plant conditions when AC power is not available from either onsite or offsite sources. The RHR provides the piping and valves which connect the FPS piping with the RHR loop C pump discharge piping. The manual valves in this line permit adding water from the FPS to the RHR system if the RHR is not operable. The primary means for supplying water through this connection is by use of the diesel-driven pump in the FPS. A backup to this pump is provided by a connection on the outside of the reactor building which allows hookup of the FPS to a fire truck pump.

The vessel injection mode is intended to prevent core damage during station blackout after RCIC has stopped operating, and to provide an in-vessel core melt prevention mechanism during a severe accident condition. If the AC-independent water addition mode is not actuated in time to prevent core damage, core melting and vessel failure, then it covers the corium in the lower drywell when initiated and adds water to containment, thereby slowing the pressure rise.

The drywell spray mode prevents high gas temperatures in the upper drywell and adds additional water to the containment, which increases the containment thermal mass and slows the pressurization rate. Additionally, the drywell spray provides fission product scrubbing to reduce fission product release in the event of failure of the drywell head.

Operation of the AC-independent water addition mode is entirely manual. All of the valves which must be opened or closed during fire water addition are located within the same ECCS valve room. The connection to add water using a fire truck pump is located outside the reactor building at grade level.

5.4.7.1.2 Design Basis for Isolation of RHR System from Reactor Coolant System

The low pressure portions of the RHR system are isolated from full reactor pressure whenever the primary system pressure is above the RHR system design pressure. (See Subsection 5.4.7.1.3 for further details.) In addition, automatic Isolation occurs for reasons of maintaining water inventory which are unrelated to line pressure rating. A low water level signal closes the RHR containment isolation valves that are provided for the shutdown cooling suction. Subsection 5.2.5 provides an explanation of the leak detection system and the isolation signals; see Subsection 5.2.5.2.1 (12) and Table 5.2-6.

The RHR pumps are protected against damage from a closed discharge valve by means of automatic minimum flow valves which open on low mainline flow and close on high mainline flow.

5.4.7.1.3 Design Basis for Pressure Relief Capacity

The relief valves in the RHR system are sized on the basis of thermal relief and valve bypass leakage only.

Redundant interlocks prevent opening valves to the low pressure suction piping when the reactor pressure is above the shutdown range. These same interlocks initiate valve closure on increasing reactor pressure.

Overpressure protection is achieved during system operation when the system is not isolated from the reactor coolant pressure. The RHR system is operational and not isolated from the reactor coolant system only when the reactor is depressurized. Two modes of operation are applicable; the flooder mode and the shutdown cooling mode. For the flooder mode, the injection valve opens through interlocks only for reactor pressure less than approximately 500 psig. For the shutdown cooling mode, the suction valves can be opened through interlocks only for reactor pressures less than approximately 135 psig. Once the system is operating in these lower pressure modes, events are not expected that would cause the pressure to increase. If for some unlikely event the pressure would increase, the pressure interlocks that allowed the valves to initially open would cause the valves to close on increasing pressure. The RHR system piping would then be protected from overpressure. The valves close at low pressure, and the rate of pressure increase would be low. During the time period while the valves are closing at these low pressure conditions, the RHR system design and margins that satisfy the interfacing system LOCA provide ample overpressure protection.

In addition, a high pressure check valve will close to prevent reverse flow if the pressure should increase. Relief valves in the discharge piping are sized to account for leakage past the check valve and are coded in accordance with the ASME Boiler and Pressure Coded Vessel Code, Section III.

5.4.7.1.4 Design Basis With Respect to General Design Criterion 5

The RHR system for this unit does not share equipment or structures with any other nuclear unit.

5.4.7.1.5 Design Basis for Reliability and Operability

The design basis for the shutdown cooling mode of the RHR System is that this mode is controlled by the operator from the control room. The only operations performed outside of the control room for a normal shutdown are manual operation of local flushing water admission valves, which are the means of providing clean water to the shutdown portions of the RHR system.

Three separate shutdown cooling loops are provided; and although the three loops are required for shutdown under normal circumstances, the reactor coolant can be brought to 100°C in less than 36 hours with only two loops in operation. The RHR system is part of the ECCS and therefore is required to be designed with redundancy, piping protection, power separation, etc., as required of such systems. (See Section 6.3 for an explanation of the design bases for ECCS Systems.)

Shutdown suction and discharge valves are required to be powered from both offsite and standby emergency power for purposes of isolation and shutdown following a loss of offsite power.

5.4.7.1.6 Design Basis for Protection from Physical Damage

The design basis for protection from physical damage, such as internally generated missiles, pipe break, seismic effects, and fires, are discussed in Sections 3.5, 3.6, 3.7, and Subsection 9.5.1

5.4.7.2 Systems Design

5.4.7.2.1 System Diagrams

All of the components of the RHR system are shown in the P&ID (Figure 5.4-10). A description of the controls and instrumentation is presented in Subsection 7.3.1.1.1 emergency core cooling systems control and instrumentation.

Figure 5.4-11 is the RHR process diagram and data. All of the sizing modes of the system are shown in the process data. The interlock block diagram (IBD) for the RHR system is provided in Section 7.3.

6.1 ENGINEERED SAFETY FEATURE MATERIALS

Materials used in the engineered safety feature (ESF) components have been evaluated to ensure that material interactions do not occur that can potentially impair operation of the ESF. Materials have been selected to withstand the environmental conditions encountered during normal operation and any postulated accident. Their compatibility with core and containment spray solutions has been considered, and the effects of radiolytic decomposition products have been evaluated.

Coatings used on exterior surfaces within the primary containment are suitable for the environmental conditions expected. Only metallic insulation is used inside containment, except for duct and antisweat insulation. All nonmetallic thermal insulation employed is required to have the proper ratio of leachable sodium plus silicate ions to leachable chloride plus fluoride (Regulatory Guide 1.36), in order to minimize the possible contribution to stress corrosion cracking of austenitic stainless steel.

6.1.1 Metallic Materials

6.1.1.1 Materials Selection and Fabrication

6.1.1.1.1 Material Specifications

Table 5.2-4 lists the principal pressure-retaining materials and the appropriate materials specifications for the reactor coolant pressure boundary (RCPB) components. Table 6.1-1 lists the principal pressure-retaining materials and the appropriate material specifications of the primary containment system, the emergency core cooling systems and their auxiliary systems and the standby liquid control system. The ESF materials selected satisfy Appendix I to Section III of the ASME Code and Parts A, B, and C of Section II of the code.

6.1.1.1.2 Compatibility of Construction Materials with Core Cooling Water and Containment Sprays

All materials of construction used in essential portions of these systems are resistant to corrosion, both in the medium contained and the external environment. General corrosion of all materials, except carbon and low-alloy steel, is negligible. Conservative corrosion allowances are provided for all exposed surfaces of carbon and low-alloy steel. Special allowances are made for the standby liquid control system which contains sodium pentaborate solution.

Demineralized water, with no additives, is employed in BWR core cooling water and containment sprays. (See Subsections 9.2.6 and 9.2.9 for a description of the water quality requirements.) Leaching of chlorides from concrete and other substances is not significant. No detrimental effects occur on any of the ESF construction materials from allowable containment levels in the high-purity water. Thus, the materials are compatible with the post-LOCA environment.

6.1.1.1.3 Controls for Austenitic Stainless Steel

6.1.1.1.3.1 Control of the Use of Sensitized Stainless Steel

Controls to avoid severe sensitization are discussed in Subsection 5.2.3.4.1.1

6.1.1.1.3.2 Process Controls to Minimize Exposure to Contaminants

Process controls for austenitic stainless steel are discussed in Subsection 5.2.3.4.1.2.

6.1.1.1.3.3 Use of Cold Worked Austenitic Stainless Steel

Austenitic stainless steel with a yield strength greater than 90,000 psi is not used in essential coolant systems.

6.1.1.1.3.4 Thermal Insulation Requirements

Nonmetallic thermal insulation materials used on ESF systems were selected, procured, tested and stored in accordance with Regulatory Guide 1.36. Insulation is required to have the proper ratio of leachable sodium plus silicate ions to leachable chloride plus fluoride ions as specified in Regulatory Guide 1.36.

6.1.1.1.3.5 Avoidance of Hot Cracking of Stainless Steel

Process controls to avoid hot cracking are discussed in Subsection 5.2.3.4.2.1

6.1.1.2 Composition, Compatibility and Stability of Containment and Core Spray Coolants

Demineralized water from the condensate storage tank or the suppression pool, with no additives is employed in the core cooling water and containments sprays. One exception is that the sodium pentaborate liquid control solution if used, enters through the core flooder system (see Subsections 9.2.6 and 9.2.9). In addition, following an accident, the containment and drywell atmospheres are maintained below 4% (by volume) hydrogen in accordance with Regulatory Guide 1.7. Material compatibility is thus demonstrated.

The post-LOCA ESF coolant, which is high-purity water, comes from one of two sources. Water in the 304L stainless steel-lined suppression pool is maintained at high purity (low corrosion attack) by the suppression pool cleanup (SPCU) system. See Subsection 9.5.9 for further details. Since the pH range (5.3 - 8.6) is maintained, corrosive attack on the pool liner (304L SS) will be insignificant over the life of the plant (see subsection 3.8.2.4.3). ESF coolant may also be obtained from the condensate storage tank, if available (see Subsection 9.2.6).

Because of the methods described above (coolant storage provisions, insulation materials requirements, and the like), as well as the fact that the containment has no significant stored quantities of acidic or basic materials, the post-LOCA aqueous phase pH in all areas of containment will have a flat time history. In other words, the liquid coolant will remain at its design basis pH throughout the event.

6.1.2 Organic Materials

6.1.2.1 Protective Coatings

The use of organic protective coatings within the containment has been kept to a minimum. The major use of such coatings is on the carbon steel containment liner, internal steel structures and equipment inside the drywell and wetwell.

The epoxy coatings are specified to meet the requirements of Regulatory Guide 1.54 and are qualified using the standard ANSI tests, including ANSI N101.4. However, because of the impracticability of using these special coatings on all equipment, certain exemptions are allowed. The exemptions are restricted to small-size equipment where, in case of a LOCA, the paint debris is not a safety hazard. Exemptions include such items as electronic/electrical trim, covers, face plates and valve handles. Other than these minor exemptions, all coatings within the containment are qualified to Regulatory Guide 1.54. See Subsection 6.1.3.1 for interface requirements.

Where decontamination or light reflection are not a consideration, carbon steel components are protected with an inorganic zinc primer only, with no organic top coat.

6.1.2.2 Other Organic Materials

Materials used in or on the ESF equipment have been reviewed and evaluated in respect to radiolytic and pyrolytic decomposition and attendant effects on safe operation of the system. For example fluorocarbon plastic (Teflon) is not permitted in environments that obtain temperatures greater than 300°F, or radiation exposures above 10^4 rads. The 10 reactor internal pump motors each contain less than 10 pounds of polyacrylic and polyethylene motor winding insulation. This material has a design life of 20 years in the environment of less than $6x10^7$ rads at 140°F maximum.

Other organic materials in the containment are qualified to environmental conditions in the containment. See Subsection 6.1.3.1 for interface requirements.

6.1.2.3 Safety Analysis

For each application the materials have been specified to withstand an appropriate radiation dose for their design life, without suffering

operation within acceptable limits for equipment operation as described to detail in Subsection 9.4.5.1.

The drywell is protected against the dynamic effects of plant-generated missiles (see Section 3.5), and the jet forces and pipe whip associated with postulated line breaks (see Section 3.6). Protection is provided by the massive structure of the drywell and by providing restraints that prevents pipes from impacting on the drywell walls. For additional information, see Subsection 3.8.3.1.

The drywell is provided with an equipment hatch for removal of equipment during maintenance and an air lock for entry of personnel into the drywell. These access openings are sealed under normal plant operation and are only opened when the plant is shut down for refueling and/or maintenance.

During normal operation, a nitrogen make-up subsystem automatically supplies nitrogen to the wetwell and the drywell to maintain a slightly positive pressure to preclude air inleakage from the reactor building. Before personnel can enter the drywell, it is necessary to deinert the drywell atmosphere. The ACS provides the purge supply and exhaust systems for deinerting as discussed in detail in Subsection 9.4.5.2.

6.2.1.1.2.2 Wetwell

The suppression pool water is located inside the wetwell annular region between the cylindrical RPV pedestal wall and the outer wall of the wetwell. The horizontal vent system communicates the drywell to the suppression pool. The nominal submergence to the centerline of the top row of horizontal vents is 3.5M. The vertical spacing between the centerlines of the horizontal vents is 1.37M. The centerline of the bottom horizontal vent is 0.76M above the bottom of the suppression pool.

In the event of a pipe break within the drywell, the increased pressure inside the drywell forces a mixture of air, steam and water through the drywell connecting vents, DCVs, and horizontal vents into the suppression pool where the steam is rapidly condensed. The noncondensable gases transported with the steam escape to and are contained in the free air volume of the wetwell. There is sufficient water volume in the suppression pool to provide a minimum of two feet of submergence over the top to the upper row of horizontal vents when water is removed from the pool during post LOCA drawdown by the ECCS. This drawdown floods the RPV to the steam lines, floods the lower drywell to its drain to the DCV, and provides for water in transit from the break on its gravity drain back to the suppression pool.

The wetwell chamber design pressure is 45 psig and design temperature is 219°F.

Performance of the pressure suppression pool concept in condensing steam under water (main steam lines through the SRVs) has been demonstrated by the horizontal vent system tests as described in Appendix 3B.

The SRVs discharge steam from the relief valves through their exhaust piping and quenchers into the suppression pool. Operation of the SRVs is intermittent and closure of the valves with subsequent condensation of steam in the exhaust piping can produce a partial vacuum, thereby sucking suppression pool water into the exhaust pipes. Vacuum relief valves are provided on the exhaust piping to control the maximum SRV discharge bubble pressure resulting from high water levels in the SRV discharge pipe.

Under normal plant operating conditions, the maximum suppression pool water and wetwell airspace temperature is 95°F or less. Under blowdown conditions following an isolation event or LOCA, the initial pool water temperature may rise to a maximum of 170°F. The continued release of decay heat after the initial blowdown may result in suppression pool temperatures as high as 207°F. The residual heat removal (RHR) system is available in the suppression pool cooling mode to control the pool temperature. Heat is removed via the RHR heat exchanger(s) to the reactor building cooling water system (RCWS) and finally to the reactor service water system (RSWS). The RHR system is described in Subsection 5.4.7.

6.2.1.1.3 Design Evaluation

6.2.1.1.3.1 Summary Evaluation

The key design parameter and the maximum calculated accident parameters for the pressure suppression containment are shown in Table 6.2-1.

The maximum drywell pressure would occur during a feedwater line break. The maximum drywell temperature condition would result from a main steam line break. All of the analyses assume that the primary system and containment system are initially at the maximum normal operating conditions.

6.2.1.1.3.2 Containment Design Parameters

Table 6.2-2 provides a listing of the key design parameters of the primary containment system including the design characteristics of the drywell, suppression pool and the pressure suppression vent system.

Table 6.2-2a provides the performance parameters of the related engineered safety feature systems which supplement the design conditions of Table 6.2-2 for containment cooling purposes during post-blowdown long-term accident operation. Performance parameters given include those applicable to full capacity operation and reduced capacities assumed for containment analyses.

6.2.1.1.3.3 Accident Response Analysis

The containment functional evaluation is based upon the consideration of several postulated accident conditions which would result in the release of reactor coolant to the containment. These accidents include:

(1) an instantaneous guillotine rupture of a feedwater line;

(2) an instantaneous guillotine rupture of a main steam line; or

(3) small break accidents.

The containment design pressure and temperature were established based on enveloping the results of this range of analyses plus providing NRC prescribed margins.

6.2.1.1.3.3.1 Feedwater Line Break

Immediately following a double-ended rupture in one of the two main feedwater lines just outside the vessel (Figure 6.2-1), the flow from both sides of the break will be limited to the maximum allowed by critical flow considerations. The effective flow area on the RPV side is given in Figure 6.2-2. During the inventory depletion period, subcooled blowdown occurs and the effective flow area at saturated condition is much less than the actual break area. The detailed calculational method is provided in Reference 1. The RPV blowdown through the break is prevented by the check valves.

The feedwater system side of the feedwater line break (FWLB) was modeled by adding a time variant feedwater mass flow rate and enthalpy directly to the drywell airspace. The time histories of the mass flow and enthalpy were determined from the operating characteristics of a typical feedwater system.

The maximum possible feedwater flow rate was calculated to be 164% of nuclear boiler rated (NBR), based on the response of the feedwater pumps to an instantaneous loss of discharge pressure. Since the feedwater control system will respond to decreasing RPV water level by demanding increased feedwater flow, and there is no FWLB sensor in the design, this maximum feedwater flow was conservatively assumed to continue for 120 seconds, as shown in Figure 6.2-3. This is very conservative because: 1) all feedwater system flow is assumed to go directly to the drywell, 2) flashing in the broken feedwater line was ignored, 3) initial feedwater flow was assumed to be 105% NBR, and 4) the feedwater pump discharge flow will coastdown as the feedwater system pumps trip due to low suction pressure. During the inventory depletion period, the flow rate is less than 164% because of the highly subcooled blowdown. A feedwater line length of 100 M was assumed on the feedwater system side.

The enthalpy of the feedwater flow is 120% of a typical BWR/5 feedwater system inventory enthalpy. The specific enthalpy time history,

Influent and effluent lines of this group are isolated by automatic or remote-manual isolation valves located as close as possible to the containment boundary.

6.2.4.3.2.4 Evaluation Against Regulatory Guide 1.11

Instrument lines that connect to the RCPB and penetrated the containment have 1/4-inch orifices and manual isolation valves, in compliance with Regulatory Guide 1.11 requirements.

6.2.4.3.3 Evaluation of Single Failure

A single failure can be defined as a failure of a component (e.g., a pump, valve, or a utility such as offsite power) to perform its intended safety functions as a part of a safety system. The purpose of the evaluation is to demonstrate that the safety function of the system will be completed even with that single failure. Appendix A to 10CFR50 requires that electrical systems be designed specifically against a single passive or active failure. Section 3.1 describes the implementation of these standards as well as General Design Criteria 17, 21, 35, 38, 41, 44, 54, 55 and 56.

Electrical as well as mechanical systems are designed to meet the single-failure criterion, regardless of whether the component is required to perform a safety action. Even though a component, such as an electrically-operated valve, is not designed to receive a signal to change state (open or closed) in a safety scheme, it is assumed as a single failure if the system component changes state or fails. Electrically-operated valves include valves that are electrically piloted but air operated, as well as valves that are directly operated by an electrical device. In addition, all electrically-operated valves that are automatically actuated can also be manually actuated from the main control room. Therefore, a single failure in any electrical system is analyzed, regardless of whether the loss of a safety function is caused by a component failing to perform a requisite mechanical motion or a component performing an unnecessary mechanical motion.

6.2.4.4 Test and Inspections

The containment isolation system is scheduled to undergo periodic testing during reactor operation. The functional capabilities of power-operated isolation valves are tested remote-manually from the control room. By observing position indicators and changes in the affected system operation, the closing ability of a particular isolation valve is demonstrated.

Air-testable check valves are provided on influent emergency core cooling lines of the HPCF and RHR systems whose operability is relied upon to perform a safety function.

A discussion of testing and inspection of isolation valves is provided in Subsection 6.2.1.6. Instruments are periodically tested and inspected. Test and/or calibration points are supplied with each instrument. Leakage integrity tests shall be performed on the containment isolation valves with resilient material seals at least once every 3 months.

6.2.5 Combustible Gas Control in Containment

The atmospheric control system (ACS-T31) is provided to establish and maintain an inert atmosphere within the primary containment during all plant operating modes except during shutdown for refueling or equipment maintenance and during limited periods of time to permit access for inspection at low reactor power. The flammability control system (FCS-T49) is provided to control the potential buildup of oxygen from design-basis radiolysis of water. The objective of these systems is to preclude combustion of hydrogen and damage to essential equipment and structures.

6.2.5.1 Design Bases

Following are criteria that serve as the bases for design:

(1) Since there is no design requirement for the ACS or FCS in the absence of a LOCA and there is no design-basis accident in the ABWR that results in core uncovery or fuel failures, the following requirements mechanistically assume that a LOCA

producing the design-basis hydrogen and oxygen has occurred.

(2) The hydrogen generation from metal-water reaction is defined in Regulatory Guide 1.7.

(3) The hydrogen and oxygen generation from radiolysis is defined in Regulatory Guide 1.7.

(4) The ACS establishes an inert atmosphere throughout the primary containment following an outage or other occasions when the containment has been purged with air to an oxygen concentration greater than 3.5 percent.

(5) The ACS maintains the primary containment oxygen concentration below the maximum permissible limit per Regulatory Guide 1.7 during normal, abnormal, and accident conditions in order to assure an inert atmosphere.

(6) The ACS also maintains a slightly positive pressure in the primary containment during normal, abnormal and accident conditions to prevent air (oxygen) leakage into the inerted volumes from the secondary containment, and provides non-essential monitoring of the oxygen concentration in the primary containment to assure a breathable mixture for safe personnel access or an inert atmosphere, as required. Essential monitoring is provided by the containment atmospheric monitoring system (CAMS) as described in Chapter 7.

(7) The drywell and the suppression chamber will be mixed uniformly after the design-basis LOCA due to natural convection and molecular diffusion. Mixing will be further promoted by operation of the containment sprays.

(8) The system is capable of controlling combustible gas concentrations in the containment atmosphere for the design bases LOCA without relying on purging and without releasing radioactive material to the environment.

(9) The system is designed to maintain an inert primary containment after the design-bases LOCA assuming a single-active failure. The backup purge function need not meet this criterion.

(10) Components of the AC system inside the reactor building are protected from postulated missiles and from pipe whip, as required to assure proper action as well as other dynamic effects such as tornado missiles and flooding. 430.209a

(11) The AC system isolation function has the capability to withstand the dynamic effects associated with the safe shutdown earthquake without loss of function. 430.209a

(12) The system is designed so that all components subjected to the primary containment atmosphere (i.e., inboard isolation valves) are capable of withstanding the temperature and pressure transients resulting from a LOCA. These components will withstand the humidity and radiation conditions in the wetwell or drywell following a LOCA.

(13) The ACS is nonsafety class except as necessary to assure primary containment integrity (penetrations, isolation valves). The ACS and FCS are designed and built to the requirements specified in Section 3.2.

(14) The ACS includes the valves and piping carrying nitrogen to the containment, valves and piping from the containment to the SGTS and HVAC (U41) exhaust line, non-safety oxygen monitoring, and all related instruments and controls. The ACS does not include any structures housing or supporting the aforementioned equipment or any ducting in the primary containment.

(15) The system is designed to facilitate periodic inspections and tests. The ACS can be inspected or tested during normal plant conditions.

differentials and flow rates, are documented during the preoperational tests and are used as base points for measurements in subsequent operational tests.

During plant operation, the ACS, its valves, instrumentation, wiring and other components outside the containment can be inspected visually at any time. Testing frequencies of the ACS components are generally correlated with testing frequencies of the associated controls and instrumentation. When a valve control is tested, the operability of that valve and its associated instrumentation are generally tested by the same action. In addition, inservice inspection of all ASME, Section III, Class 3 components is done in accordance with Subsection 6.6.5.

Preoperational tests of the combustible gas control system are conducted during the final stages of plant construction prior to initial startup.

The overpressure protection concept was designed to minimize any adverse impact on normal operation or maintenance. Initially, several rupture disks from a batch of rupture disks could be tested to verify the opening characteristics and setpoint. The disks would be replaced every five years according to normal industry practice. The installation of the disks would not impact containment leakage tests, since disk integrity is expected to be essentially perfect.

The overpressure protection valves would be tested during preoperational testing and periodically during surveillance testing to verify their normally open position and their ability to close using AC power and pneumatic air.

6.2.5.5 Instrumentation Requirements

Separate inerting flow indication to both the drywell and wetwell are provided. Drywell pressure and makeup flow are monitored and recorded in the main control room. Additional drywell pressure instrumentation, with a lower setpoint, provided in addition to the redundant, safety-grade drywell pressure instrumentation of the nuclear boiler system. If drywell pressure exceeds a given setpoint, the makeup and inerting valves are closed. The temperature of the makeup and inerting vaporizers nitrogen outlet are monitored. Low makeup vaporizer nitrogen outlet temperature alarms (only) in the main control room. Auxiliary steam feeding the main inerting vaporizer(s) is controlled to regulate the inerting vaporizer nitrogen outlet temperature. Low inerting vaporizer nitrogen outlet temperature sounds a local alarm and low-low temperature isolates the main inerting line. It is intended that the local panel be attended full-time during all main inerting operations. All locally-mounted instruments are easily read from the local ACS panel. Keylocked switches in the main control room are provided to override the containment isolation signal to the valves providing nitrogen makeup to the drywell and wetwell and the small 50 mm drywell vent line. Position indication in the main control room is provided for all remotely-operated valves.

Backup purge and the addition of makeup nitrogen is initiated at the operator's discretion.

Design details and logic of the instrumentation is discussed in Chapter 7.

As discussed in Section 6.2.5.2, safety-grade oxygen monitoring is provided in the wetwell and drywell by the CAMS. This monitoring function is not used for normal operation. Separate oxygen monitoring is included in the ACS for use during non-accident plant operation to determine when the primary containment is inert and nitrogen purging may be terminated and when the primary containment is de-inerted and personnel re-entry procedures may be initiated.

The ACS oxygen monitors for assuring safe personnel entry and an inert condition during startup, normal, and abnormal operating conditions have a range from 0 to 25 percent at 100 percent relative humidity. The maximum and minimum inlet temperature to the oxygen monitor will be 10 and 65°C, respectively. Two sample points are provided in both the drywell and wetwell, high and low in their respective compartments and in opposing quadrants. Each airlock is also sampled.

The sample lines are sized and sloped to assure draining condensation to the containment.

There are no loops in the sample lines which could collect water and block flow. The oxygen monitors provide indication outside of the primary containment where necessary (for example, at and in each airlock) to assure safe operator access into first the airlock and then the containment.

The ACS oxygen analyzing system is provided to indicate the concentration of oxygen inside the containment during reactor operation, and to aid in maintaining the oxygen concentration below a safety limit prescribed in the plant technical specifications. The oxygen analyzing system readings are not used as a basis for determining when drywell entry criteria are not

(9) peak cladding temperature as a function of time.

A conservative assumption made in the analysis is that all offsite AC power is lost simultaneously with the initiation of the LOCA. As a further conservatism, all reactor internal pumps were assumed to trip at the start of LOCA event even though this in itself is considered to be an accident (See Subsection 15.3.1). The resulting rapid core flow coastdown produces a calculated departure from nucleate boiling in the hot bundles within the first few seconds of the transient.

LOCA analyses using break areas less than the maximum values were also considered for the steamline, feedwater line, and RHR shutdown suction line locations. The cases analyzed are indicated on the break spectrum plot (refer to Figure 6.3-10). In general, the largest break at each location is the worst in terms of minimum transient water level in the downcomer.

6.3.3.7.5 Intermediate Line Breaks Inside Containment

For this case the maximum RHR/LPFL injection line break (0.221 ft^2) was analyzed. Since the bottom head drain line ties into the RHR shutdown suction line, the total break flow for the maximum RHR shutdown suction line break includes flow from the vessel through RHR shutdown suction vessel nozzle as well as through the bottom head drain line. Important variables from this analysis are shown in Figures 6.3-37 through 6.3-43.

6.3.3.7.6 Small Line Breaks Inside Containment

For these cases the maximum high presure core flooder line break (0.099 ft^2) and the maximum bottom head drain line break (0.0218 ft^2), based on a 2 inch penetration in the vessel bottom head were analyzed. Since the bottom head drain line ties into the RHR shutdown suction line, the total break flow for the maximum bottom head drain line break includes flow from the vessel through the bottom head drain line penetration as well as through the RHR shutdown suction line. Important variables from these analyses are shown in Figures 6.3-44 through 6.3-59. A break in a reactor internal pump would involve either the welds or the casing. If the weld from the pump casing to the PRV stub tube breaks, the stretch tube will prevent the pump casing from moving. The stretch tube clamps the diffuser to the pump casing, where its nut seats. The land is located below the casing attachment weld and therefore the stretch tube forms a redundant parallel strength path to the pump casing restraint which is designed to provide support in the event of weld failure. In case the pump casing and the stretch tube break, the pump and motor will move downward until stopped by the casing restraints. The pump is part of the stretch tube. In either case the break flow would be much less than the drainline break case. Therefore, the drainline break analysis is also bounding for any credible break within the reactor internal pump recirculation system and its associated motor housing and cover.

As expected, the core flooder line break is the worst break location in terms of minimum transient water level in the downcomer. In elevation it is the lowest break on the vessel except for the drainline break. Furthermore, the worst break/failure combination leaves the fewest number of ECC systems remaining and no high pressure core flooder systems. LOCA analyses using break areas less than the maximum values were also considered. The cases analyzed are indicated on the break spectrum plot (refer to Figure 6.3-10). From these results it is clear that the overall most limiting break in terms of minimum transient water level in the downcomer, is the maximum core flooder line break case.

6.3.3.7.7 Line Breaks Outside Containment

This group of breaks is characterized by a rapid isolation of the break. Since a maximum steam line break outside the containment produces more vessel inventory loss before isolation than other breaks in this category, the results of this case are bounding for all breaks in this group. Important variables from these analyses are shown in Figure 6.3-60 through 6.3-66.

As discussed in Subsection 6.3.3.7.4, the trip of all reactor internal pumps at the start of the LOCA produces a calculated departure from

nucleate boiling for all LOCA events. Furthermore, the high void content in the bundles following a large steamline break produces the earliest times of loss of nucleate boiling for any LOCA event. Thus, the summary of results in Table 6.3-4 show that, though the PCTs for all break locations are similar, the steamline breaks result in higher calculated PCTs and the outside steamline break is the overall most limiting case in terms of the highest calculated PCT. Results of the analysis of this break will be provided for each bundle design for information by the utility referncing the ABWR design.

6.3.3.7.8 Bounding Peak Cladding Temperature Calculations

Consistent with the SAFER application methodology in Reference 2, the Appendix K peak cladding temperatures calculated in the previous sections must be compared to a statistically calculated 95% probability value. Table 6.3-6 presents the significant plant variables which were considered in the edetermination of the 95% probability PCT or the sensitivity study. Again, since the ABWR LOCA results have a large margin to the acceptance criteria, a conservative PCT calculation was performed which bounds the 95% probability PCT. This bounding PCT was calculated by varying all plant variables in the

low water level and drywell high pressure plus indication that at least one RHR or HPCF pump is operating. The HPCF, RCIC, and RHR automatically return from system flow test modes to the emergency core cooling mode of operation following receipt of an automatic initation signal. The RHR LPFL mode injection into the RPV begins when reactor pressure decreases to the RHR's pump discharge shutoff pressure.

HPCF injection begins as soon as the HPCF pump is up to speed and the injection valve is open, since the HPCF is capable of injection water into the RPV over a pressure range from 1177 psid to 100 psid or pressure difference between the vessel and drywell.

6.3.6 Interfaces

6.3.6.1 ECCS Performance Results

The exposure dependent MAPLHGR, peak cladding temperature, and oxidation fraction for each fuel bundle design based on the limiting break size will be provided by the utility to the USNRC for information.

6.3.7 Reference

1. *General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10CFR50, Appendix K*, (NEDE-20566-P-A), September 1986.

Table 6.3-1

SIGNIFICANT INPUT VARIABLES USED IN THE LOSS-OF-COOLANT ACCIDENT ANALYSIS (Continued)

Variable	Units	Value
Initiating Signals		
Low Water Level	ft above TAF	≤0.6
and		
High Drywell Pressure or	psig	≥2.0
High Drywell Pressure Bypass Timer Timed Out	sec	≤480
Delay Time from All Initiating Signals Completed to the Time Valves are Open	sec	≤29

C. FUEL PARAMETERS*

Variable	Units	Value
Fuel Type	-------	Initial Core
Fuel Bundle Geometry	-------	8x8
Lattice	-------	C
Number of Fueled Rods per Bundle	-------	62
Peak Technical Specification Linear Heat Generation Rate	kw/ft	13.4
Initial Minimum Critical Power Ratio	-------	1.13
Design Axial Peaking Factor	-------	1.40

* The system response analysis is based upon the core loading in Figure 4.3-1. The sensitivities demonstrated are valid for other core loadings.

Table 6.3-2

OPERATIONAL SEQUENCE OF EMERGENCY CORE COOLING SYSTEM MAXIMUM CORE FLOODER LINE BREAK

Time (sec)	Events
0	Design basis LOCA assumed to start; normal auxiliary power assumed to be lost (1)
~5	Reactor low water level 3 is reached. Reactor cram occurs. (2)
~10	Drywell high pressure is reached. All diesel-generators, RCIC, HPCF, RHR/LPFL signaled to start. (3)
~18	Reactor low water level 2 is reached. RCIC receives second signal to start.
~48	RCIC injection valve open and pump at design flow which completes RCIC startup.
~65	Reactor low water level 1.5 is reached. All diesel-generators and HPCF receive second signal to start. Main steam isolation valves signaled to close.
~78	All diesel-generators ready to load; RHR/LPFL and HPCF loading sequence beings.
~102	HPCF injection valves open and pumps at design flow, which completes HPCF startup
~118	Reactor low water level 1 is reached. RHR/LPFL receives second signal to start. ADS delay timer initiated.
~148	ADS delay timer timed out. ADS valves actuated.
~344	Vessel pressure decreases below shut-off head of RHR/LPFL. RHR/LPFL injection valves open and flow into vessel beings.
*	Core effectively reflooded assuming worst single failure; heatup terminated

NOTES:

(1) For the purpose of all but the next to last entry on this table, all ECCS equipment is assumed to function as designed. Performance analysis calculations consider the effects of single equipment failures (Subsection 6.3.3.3).

(2) For the LOCA analysis, the water level is initiated at the scram water level.

(3) For the LOCA analysis, the ECCS initiation on high drywell pressure is not considered.

* See Figure 6.3-46

6.5 FISSION PRODUCTS REMOVAL AND CONTROL SYSTEMS

6.5.1 Engineered Safety Features Filter Systems

The filter systems required to perform safety-related functions following a design basis accident are:

(1) Standby gas treatment system (T22-SGTS).

(2) Control room portion of the HVAC system. (U41-HVAC)

The control room portion of the HVAC system is discussed in Section 6.4 and Subsection 9.4.1. The SGTS is discussed in this Subsection (6.5.1).

6.5.1.1 Design Basis

6.5.1.1.1 Power Generation Design Basis

The SGTS has the capability to filter the gaseous effluent from the primary containment or from the secondary containment when required to limit the discharge of radioactivity to the environment to meet 10CFR100 requirements.

6.5.1.1.2 Safety Design Basis

The SGTS is designed to accomplish the following:

(1) Maintain a negative pressure in the secondary containment, relative to the outdoor atmosphere, to control the release of fission products to the environment.

(2) Filter airborne radioactivity (halogen and air particulates) in the effluent to reduce offsite doses to within the limits specified in 10CFR100.

(3) Ensure that failure of any active component, assuming loss of offsite power, cannot impair the ability of the system to perform its safety function.

(4) Remain intact and functional in the event of a safe shutdown earthquake (SSE).

(5) Meet environmental qualification requirements established for system operation.

6.5.1.2 System Design

6.5.1.2.1. General

The SGTS P&ID is provided as Figure 6.5-1.

6.5.1.2.2 Component Description

Table 6.5-1 provides a summary of the major SGTS components. The SGTS consists of two parallel and redundant trains of active equipment which share a single filter train. Suction is taken from above the refueling area or from the primary containment via the atmospheric control system (T31-ACS). The discharge goes to the main plant stack.

The SGTS consists of the following principal components:

(1) Two independent dryer trains consisting of a moisture separator and an electric process heater.

(2) Two independent process fans located upstream of the filter train.

(3) A filter train consisting of a prefilter, a high efficiency particulate air (HEPA) filter, a charcoal adsorber, a second HEPA filter, and space heaters.

6.5.1.2.3 SGTS Operation

6.5.1.2.3.1 Automatic

Upon the receipt of a high drywell pressure signal or a low reactor water level signal, or when high radioactivity is detected in the secondary containment or refueling floor

ventilation exhaust, the SGTS is automatically actuated. If system operation is not confirmed, the redundant process fan and dryer train are automatically placed into service. In the event a malfunction disables an operating process fan or dryer train, the standby process fan and dryer train are manually initiated.

6.5.1.2.3.2 Manual

The SGTS is on standby during normal plant operation and may be manually initiated before or during primary containment purging (de-inerting) when required to limit the discharge of contaminants to the environment. It may be manually initiated whenever its use may be needed to avoid exceeding radiation monitor setpoints.

6.5.1.2.3.3 Decay Heat Removal

Cooling of the SGTS filters may be required to prevent the gradual accumulation of decay heat in the charcoal. This heat is generated by the decay of radioactive iodine adsorbed on the SGTS charcoal. The charcoal is typically cooled by the air from the process fan.

A water deluge capability is also provided, but primarily for fire protection since redundant process fans are provided for air cooling. Since the deluge is available, it may also be used to remove decay heat for sequences outside the normal design basis. Temperature instrumentation is provided for control of the SGTS process and space electric heaters. This instrumentation may also be used by the operator to [re-]establish a cooling air flow post-accident, if required.

Water is supplied from the fire protection system and is connected to the SGTS via a spool piece.

6.5.1.3 Design Evaluation

6.5.1.3.1 General

(1) A slight negative pressure is normally maintained in the secondary containment by the reactor building HVAC system (Subsection 9.4.5). On SGTS initiation per Subsection 6.5.1.2.3.1, the secondary containment is automatically isolated from the HVAC system.

(2) The SGTS filter particulate and charcoal efficiencies are outlined in Table 6.5-1. Dose analyses of events requiring SGTS operation, described in Subsections 15.6.5 and 15.7.4, indicate that offsite doses are within the limits established by 10 CFR 100.

(3) The SGTS is designated as an engineered safety feature since it mitigates the consequences of a postulated accident by controlling and reducing the release of radioactivity to the environment. The SGTS, except for the deluge, is designed and built to the requirements for Safety Class 3 equipment as defined in Section 3.2, and 10 CFR 50, Appendix P.

The SGTS has independent, redundant active components. Should any active component fail, SGTS functions can be performed by the redundant component. The electrical devices of independent components are powered from separate Class 1E electrical buses.

(4) The SGTS is designed to Seismic Category I requirements as specified in Section 3.2. The SGTS is housed in a Category I structure. All surrounding equipment, components, and supports are designed to appropriate safety class and seismic requirements.

(5) The SGTS design is based on the maximum pressure and differential pressure, maximum integrated dose rate, maximum relative humidity, and maximum temperature expected in secondary containment for the LOCA event.

6.5.1.3.2 Sizing Basis

Figure 6.5-2 provides an assessment of the secondary containment pressure after the design-basis LOCA assuming an SGTS fan capacity of 4000 scfm (70°F, 1 atmosphere) per fan and the leakage rates shown in Table 6.5-2. Credit for secondary containment as a fission product control system is only taken if the secondary containment is actually at a negative pressure by considering the potential effect of wind on the ambient pressure in the vicinity of the reactor building. For the ABWR dose analysis, direct transport of containment leakage to the environment was assumed for the first 20 minutes after LOCA event initiation (in addition to the leakage through the MSIVs to the main turbine condenser). Each SGTS fan was sized to

At locations where leakages are normally expected and leakage collection systems are located, (e.g., valve stems and pump seals), the visual, VT-2, examination shall verify that the leakage collection system is operative.

Piping runs shall be clearly identified and laid out such that insulation damage, leaks and structural distress will be evident to a trained visual examiner.

6.6.3.2.2 Surface Examination

Magnetic Particle and Liquid Penetrant examination techniques shall be performed in accordance with ASME Section XI, IWA-2221 and IWA-2222, respectively. For direct examination access for magnetic particle (MT) and penetrant (PT) examination, a clearance (of at least 24 inches of clear space) is provided where feasible for the head and shoulders of a man within a working arm's length (20 inches) of the surface to be examined. In addition, access shall be provided as necessary to enable physical contact with the item as necessary to perform the examination. Remote MT and PT generally are not appropriate as a standard examination process, however, borescopes and mirrors can be used at close range to improve the angle of vision. As a minimum, insulation removal shall expose the area of each weld plus at least six inches from the toe of the weld on each side. Insulation will generally be removed 16 inches on each side of the weld.

6.6.3.2.3 Volumetric Ultrasonic Direct Examination

Volumetric ultrasonic direct examination shall be performed in accordance with ASME Section XI, IWA-2232. In order to perform the examination, visual access to place the head and shoulder within 20 inches of the area of interest shall be provided where feasible. Nine inches between adjacent pipes is sufficient spacing if there is free access on each side of the pipes. The transducer dimension has been considered: a 1 1/2 inch diameter cylinder, 3 inches long placed with the access at a right angle to the surface to be examined. The ultrasonic examination instrument has been considered as a rectangular box 12 x 12 x 20 inches located within 40 feet from the transducer. Space for a second examiner to monitor the instrument shall be provided if necessary.

Insulation removal for inspection is to allow sufficient room for the ultrasonic transducer to scan the examination area. A distance of 2T plus 6 inches, where T is the pipe thickness, is the minimum required on each side of the examination area. The insulation design generally leaves 16 inches on each side of the weld, which exceeds minimum requirements.

6.6.3.2.4 Alternative Examination Techniques

As provided by ASME Section XI, IWA-2240, alternative examination methods, a combination of methods, or newly developed techniques may be substituted for the methods specified for a given item in this section, provided that they are demonstrated to be equivalent or superior to the specified method. This provision allows for the use of newly developed examination methods, techniques, etc., which may result in improvements in examination reliability and reductions in personnel exposure.

6.6.3.2.5 Data Recording

Manual data recording will be performed where manual ultrasonic examinations are performed. If automated systems are used, electronic data recording and comparison analysis are to be employed with automated ultrasonic examination equipment. Signals from each ultrasonic transducer would be fed into a data acquisition system in which the key parameters of any reflectors will be recorded. The data to be recorded for manual and automated methods are:

(1) location;
(2) position;
(3) depth below the scanning surface;
(4) length of the reflector;
(5) transducer data including angle and frequency; and
(6) calibration data.

The data so recorded shall be compared with the results of subsequent examinations to determine the behavior of the reflector.

6.6.4 Inspection Intervals

6.6.4.1 Class 2 Systems

The inservice inspection intervals for Class 2 systems will conform to Inspection Program B as

described in Section XI, IWC-2412. Except where deferral is permitted by Table IWC-2500-1, the percentages of examinations completed within each period of the interval shall correspond to Table IWC-2412-1. Items selected to be examined within the 10 year intervals are selected in accordance with the requirements of Table IWC-2500-1 from those listed in Table 6.6-1.

6.6.4.2 Class 3 Systems

The inservice inspection intervals for Class 3 systems will conform to Inspection Program B as described in Section XI, IWD-2412. Except where deferral is permitted by Table IWD-2500-1, the percentages of examinations completed within each period of the interval shall correspond to Table IWD-2412-1. Items selected to be examined within the 10 year intervals are selected in accordance with the requirements of Table IWD-2500-1 from those listed in Table 6.6-1.

6.6.5 Evaluation of Examination Results

Examination results will be evaluated in accordance with ASME Section XI, IWC-3000 for Class 2 components, with repairs based on the requirements of IWA-4000 and IWC-4000. Examination results will be evaluated in accordance with ASME Section XI, IWD-3000 for Class 3 components, with repairs based on the requirements of IWA-4000 and IWD-4000.

6.6.6 System Pressure Tests

6.6.6.1 System Inservice Test

As required by Section XI, IWC-2500 for category C-H and by IWD-2500 for categories D-A, D-B and D-C, a system inservice test shall be performed in accordance with IWC-5221 on Class 2 systems, and IWD-5221 on Class 3 systems, which are required to operate during normal operation. The system inservice test shall include all Class 2 or 3 components and piping within the pressure retaining boundary and shall be performed once during each inspection period as defined in Tables IWC-2412-1 and IWD-2412-1 for Program B. For the purposes of the system inservice test of Class 2 systems, the pressure retaining boundary is defined in Table IWC-2500-1, Category C-H, Note 7. For the purposes of the system inservice test for Class 3 systems, the system boundary is defined in Note 1 of Table IWD-2500-1, for categories D-A, D-B and D-C. The system inservice test shall include a VT-2 examination in accordance with IWA-5240, except that, where portions of a system are subject to system pressure tests associated with two different functions, the VT-2 examination shall only be performed during the test conducted at the higher of the test pressures. The system inservice test will be conducted at approximately the maximum operating pressure and temperature indicated in the applicable process flow diagram for the system as indicated in Table 1.7-1. The system hydrostatic test (Subsection 5.2.4.6.2), when performed is acceptable in lieu of the system inservice test.

6.6.6.2 System Functional Test

As required by Section XI, IWC-2500 for category C-H and by IWD-2500 for categories D-A, D-B and D-C, a system functional test shall be performed in accordance with IWC-5221 on Class 2 systems, and IWD-5221 on Class 3 systems, which are not required to operate during normal operation but for which a periodic system functional test is performed. The system functional test shall include all Class 2 or 3 components and piping within the pressure retaining boundary and shall be performed once during each inspection period as defined in Tables IWC-2412-1 and IWD-2412-1 for Program B. For the purposes of the system functional test of Class 2 systems, the pressure retaining boundary is defined in Table IWC-2500-1, Category C-H, Note 7. For the purposes of the system functional test for Class 3 systems, the system boundary is defined in Note 1 of Table IWD-2500-1, categories D-A, D-B and D-C. The system inservice test shall include a VT-2 examination in accordance with IWA-5240, except that, where portions of a system are subject to system pressure tests associated with two different functions, the VT-2 examination shall only be performed during the test conducted at the higher of the test pressures. The system functional test will be conducted at the nominal operating pressure and temperature indicated in the applicable process flow diagram for the functional test for each system as indicated in Table 1.7-1. The system hydrostatic test (Subsection 5.2.4.6.2), when performed is acceptable in lieu of the system inservice test.

6.6.6.3 Hydrostatic Pressure Tests

As required by Section XI, IWC-2500 for Category B-P, the hydrostatic pressure test shall be

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(6) The sixth test is an integrated self-test provision built into the microprocessors within the safety system logic and control (SSLC). It consists of an on-line, continuously operating, self-diagnostic monitoring network; and an off-line semi-automatic (operator initiated, but automatic to completion), end-to-end surveillance program. Both on-line and off-line functions operate independently within each of the four divisions. There are no multi-divisional interconnections associated with self-testing.

The primary purpose of the self test is to improve the availabilty of the SSLC by optimizing the time to detect and determine the location of a failure in the functional system. It is not intended that self-test eliminate the need for the other five manual tests. However, most faults are detected more quickly than with manual testing alone.

The self-test function is classified as safety related. Its hardware and software are an integral part of the SSLC and, as such, are qualified to Class 1E standards.

The hierarchy of test capability is provided to ensure maximum coverage of all EMS/SSLC functions, including logic functions and data communications links. Testing shall include:

(a) On-line Continuous Testing.

A self-diagnostic program monitors each signal processing module from input to output. Testing is automatic and is performed periodically during normal operation. Tests will verify the basic integrity of each card or module on the mocroprocessor bus. All operations are part of normal data processing intervals and will not affect system response to incoming trip or initiation signals. Automatic initiation signals from plant sensors will override an automatic test sequence and perform the required safety function. Process or logic signals are not changed as a result of self-test functions.

Self-diagnosis includes monitoring of overall program flow, reasonableness of process variables, RAM and PROM condition, and verification of 2/4 coincidence logic and device interlock logic. Testing includes continuous error checking of all transmitted and received data on the serial data links of each SSLC controller; for example, error checking by parity check, checksum, or cyclic redundancy checking (CRC) techniques.

A fault is considered the discrepancy between an expected output of a permissive circuit and the existing present state.

Actuation of the trip function is not performed during this test. The self-test function is capable of detecting and logging intermittent failures without stopping system operation. Normal surveillance by plant personnel will identify these failures, via a diagnostic display, for preventive maintemance.

Self-test failures (except intermittent failures) are annunciated to the operator at the main control room console and logged by the process computer. Faults are indentified to the replacement board or module level and positively indicated at the failed unit.

The continuous surveillance monitoring also includes power supply voltage levels, card-out-of-file interlocks, and battery voltage levels on battery-backed memory cards (if used). Out-of-tolerance conditions will result in an inopertive (out-of-service) condition for that particular system function.

Automatic system self-testing occurs during a portion of every periodic transmission period of the data communication network. Since exhaustive tests cannot be performed during any one transmission interval, the test software is written so that sufficient overlap coverage is provided to prove system performance during tests of portions of the circuitry, as allowed in IEEE 338.

The essential multiplexing system (EMS) is included in the continuous, automatic self-test function. Faults at the Remote Multiplexing Units (RMUs) are alarmed in the main control room. Since EMS is dual in each division, self-test supports automatic reconfiguration or bypass of portions of EMS after a detected fault, such that the least effect on system availability occurs.

(b) Off-line Semi-automatic End-to-End (sensor input to trip actuator) Testing

The more complete, manually-initiated, internal self-test is available when a unit is off-line for surveillance or maintenance testing. This test exercises the trip outputs of the SSLC logic processors. The channel containing the processors will be bypassed during testing.

A fault is considered the inability to open or close any control circuit.

Self-test failures are displayed on a front panel readout device or other diagnostic unit.

To reduce operator burden and decrease outage time, a surveillance test controller (STC) is provided as a dedicated instrument in each division of SSLC. The STC performs semi-automatic (operator-initiated) testing of SSLC fonctional logic, including trip, initiation, and interlock logic. Test coverage includes verification of correct operation of the following capabilities, as defined in each system IBD.

(i) Each 2/4 coincident logic function.

(ii) Serial and parallel I/O, including manual control switches, limit switches, and other contact closures.

(iii) The 1/N trip selection function.

(iv) Interlock logic for each valve or pump.

A separate test sequence for each safety system is operator-selectable; testing will proceed automatically to conclusion after initiation by the operator. Surveillance testing is performed in one division at a time. The surveillance test frequency is given in the ABWR Technical Specification (Chapter 16).

The STC injects test patterns through the essential multiplexing system (EMS) communications links to the RMUs. It then tests the RMUs ability to format and transmit sensor data through and across the EMS/SSLC interface, in the prescribed time, to the load drivers. Under the proper bypass conditions, or with the reactor shut down, the load drivers themselves may be actuated.

All testing features adhere to the single failure criterion, as follows: 1) No single failure in the test circuitry shall incapacitate an SSLC safety function. 2) No single failure in the test circuitry shall cause an inadvertent scram, MSIV isolation, or actuation of any safety systems served by the SSLC.

conjunction with the pressure relief system is adequate to preclude over-pressurizing the nuclear system, the turbine control valve fast-closure scram provides additional margin to the nuclear system pressure limit. The turbine control valve fast-closure scram setting is selected to provide timely indication of control valve fast closure.

(6) Main Steamline Isolation

The main steamline isolation valve closure can result in a significant addition of positive reactivity to the core as nuclear system pressure rises. The main steamline isolation scram setting is selected to give the earliest positive indication of main steamline isolation without inducing spurious scrams.

(7) Low Charging Pressure to Rod hydraulic Control Unit Accumulators

The RC hydraulic system normally supplies charging water at sufficient pressure to charge all scram accumulators of the individual rod hydraulic control units (RHCUs) to pressure values that will assure adequate control rod scram insertion rates during a full reactor trip or scram. A low charging water pressure is indicative of the potential inability to maintain the scram accumulators pressurized. A reactor trip is initiated after a specified time delay, before the charging water pressure drops to a value that could eventually result in slower than normal scram speed control rod insertion.

(8) Drywell High Pressure

High pressure inside the drywell may indicate a break in the reactor coolant pressure boundary. It is prudent to scram the reactor in such a situation to minimize the possibility of fuel damage and to reduce energy transfer from the core to the coolant. The drywell high pressure scram setting is selected to be as low as possible without inducing spurious scrams.

(9) Main Steamline High Radiation

High radiation in the vicinity of the main steamlines may indicate a gross fuel failure in the core. When high radiation is detected near the steamlines, a scram is initiated to limit release of fission products from the fuel. The high radiation trip setting is selected high enough above background radiation levels to avoid spurious scrams yet low enough to promptly detect a gross release of fission products from the fuel. More information on the trip setting is available in Section 7.3.

(10) High Seismic Activity

Since high seismic activity is a source of potential damage to the plant, a reactor trip is initiated upon indication of such high seismic activity. There is one trip signal associated with each of the four RPS instrument channels.

7.2.1.1.8 Containment Electrical Penetration Assignment

Electrical containment penetrations are assigned to the protection systems on a four- division basis (Subsections 7.2.1.1.4.1 and 4.6).

Each penetration is provided with a NEMA-4 enclosure box on each end providing continuation of the metal wire ways (Subsection 7.2.1.1.4.6).

7.2.1.1.9 Cable Spreading Area Description

The cable spreading areas adjacent to the control room are termed cable rooms and electrical equipment rooms. A description of the separation criteria used in these rooms is in Section 8.3. Cable routing through the cable rooms is shown on raceway plans by reference in Section 6.7.

7.2.1.1.10 Main Control Room Area

Virtually all hardware within the RPS design scope is located within the four separate and redundant safety system logic and control (SSLC) cabinets in the main control room except the instrumentation for monitoring turbine stop valve closure and turbine control valve fast closure, and turbine first stage pressure. The panels are mounted on four separate control complex system steel floor sections which, in turn, are installed in the main control room. The major control switches are located on the principal console.

7.2.1.1.11 Control Room Cabinets and Their Contents

The SSLC logic cabinets, which contain the RPS, for Divisions 1, 11, 111, and IV include a vertical board for each division. The vertical boards contain digital and solid state discrete and integrated circuits used to condition signals transferred to the SSLC from the essential multiplex system (EMS). They also contain combinational and sequential logic circuits for the initiation of safety actions and/or alarm annunciation, isolators for electrical and physical separation of circuits used to transmit signals between redundant safety systems or between safety and nonsafety systems, and system support circuits such as power supplies, automatic testing circuits, etc. Load drivers with solid-state switching outputs for actuation solenoids, motor control centers, or switchgear may be located in the control room or throughout the plant.

The principal console contains the reactor mode switch, the RPS manual scram push button switches, the RPS scram reset switches and the bypass switches for the low RCS accumulator charging pressure.

7.2.1.1.12 Test Methods that Enhance RPS Reliability

Surveillance testing is performed periodically on the RPS during operation. This testing includes sensor calibration, response-time testing, trip channel actuation, and trip time measurement with simulated inputs to individual trip modules and sensors. The sensor channels can be checked during operation by comparison of the associated control room displays on other channels of the same variable. Fault-detection diagnostic testing is not being used to satisfy tech spec requirements for surveillance.

7.2.1.1.13 Interlock Circuits to Inhibit Rod Motion

Interlocks between the RPS and RC&IS inhibit rod withdrawal when the CRD charging pressure trip bypass switch is in the "BYPASS" position. These interlocks assure that no rods can be withdrawn when conditions are such that the RPS cannot re-insert rods if necessary.

7.2.1.1.14 Support Cooling System and HVAC Systems Descriptions

The cooling (ventilating) systems important for proper operation of RPS equipment are described in Section 9.4.

7.2.1.2 Design Bases

Design bases information requested by IEEE 279 is discussed in the following paragraphs. These IEEE 279 design bases aspects are considered separately from those more broad and detailed design bases for this system cited in Subsection 7.1.2.2.

(1) Conditions

Generating station conditions requiring RPS protective actions are defined in the Technical Specifications, Chapter 16.

(2) Variables

The generating station variables which are monitored cover the protective action conditions that are identified in Subsection 7.2.1.2.1.

(3) Sensors

A minimum number of LPRMS per APRM are required to provide adequate protective action. This is the only variable that has spatial dependence (IEEE 279, Paragraph 3.3).

(4) Operational Limits

Operational limits for each safety-related variable trip setting are selected with sufficient margin to avoid a spurious scram. It is then verified by analysis that the release of radioactive material following postulated gross failure of the fuel or the reactor coolant pressure boundary is kept within acceptable bounds. Design basis operational limits in chapter 16 are based on operating experience and constrained by the safety design basis and the safety analyses.

(5) Margin Between Operational Limits

The margin between operational limits and the limiting conditions of operation (scram) for the reactor protection system are in Chapter 16, Technical Specifications. The margin includes the maximum allowable accuracy error, sensor response times, and sensor setpoint drift.

ways; they will relieve pressure by actuation with electrical power or by mechanical actuation without power. The suppression pool provides a heat sink for steam relieved by these valves. Relief valve operation may be controlled manually from the control room to hold the desired reactor pressure. Eight of the eighteen SRVs are designated as automatic depressurization system (ADS) valves and are capable of operating from either ADS logic or safety/ relief logic signals. The safety/relief logic is discussed in Paragraph (4). Automatic depressurization by the ADS is provided to reduce the pressure during a loss-of-coolant accident in which the HPCF and/or RCIC are unable to restore vessel water level. This allows makeup of core cooling water by the low pressure makeup system (RHR/LP flooding mode).

(2) Supporting System (Power Supplies)

Supporting systems for the ADS C&I include the instrumentation, logic, control and motive power sources. The instrumentation and logic power is obtained from the SSLC Division I and II, 120-VAC buses F1 and G1. The control power is from the Division I and II, 125-VDC battery buses F and G (see Figure 8.3-1). The motive power for the electrically operated gas pilot solenoid valves is from local accumulators supplied by the high pressure nitrogen gas supply systems (Divisions I and II) (see Chapter 6).

(3) Equipment Design

The automatic depressurization system (ADS) consists of redundant trip channels arranged in two separated logics that control two separate solenoid-operated gas pilots on each ADS valve. Either pilot valve can operate its associated ADS valve. These pilot valves control the pneumatic pressure applied by accumulators and the high pressure nitrogen gas supply system. The operator can also control the SRV's manually. Separate accumulators are included with the control equipment to store pneumatic energy for relief valve operation.

The ADS accumulators are sized to operate the safety relief valve two times with the drywell at 70% of design gage pressure following failure of the pneumatic supply to the accumulator. Sensors provide inputs to local multiplexer units which perform signal conditioning and analog-to-digital conversion. The formatted, digitized sensor inputs are multiplexed with other sensor signals over an optical data link to the logic processing units in the main control room. All four transmitter signals are fed into the two-out-of-four logic for each of two divisions either of which can actuate the ADS. Station batteries and SSLC power supplies energize the electrical control circuitry. The power supplies for the redundant divisions are separated to limit the effects of electrical failures. Electrical elements in the control system energize to cause the relief valves to open.

(a) ADS Initiating Circuits

Two ADS subsystems for relief valve actuation, ADS 1 and ADS 2 are provided (see Figure 7.3-2). Sensors from all four divisions and division I control logic for low reactor water level and high drywell pressure initiate ADS 1, and sensors from all four divisions and division II control logic initiate ADS 2. The division I logic is mounted in a different cabinet than the division II logic.

The reactor vessel low water level initiation setting for the ADS is selected to depressurize the reactor vessel in time to allow adequate cooling of the fuel by the RHR (LP flooding mode) system following a loss-of-coolant accident in which the HPCF and/or RCIC fail to perform their functions adequately. Timely depressurization of the reactor vessel is provided if the reactor water level drops below acceptable limits together with an indication that high drywell pressure has occurred, which signifies there is a loss of coolant into the containment with insufficient high pressure makeup to maintain reactor water level. For breaks outside the containment, timely depressurization of the reactor vessel

is provided if the reactor vessel water level drops below acceptable limits for a time period sufficient for the ADS high drywell pressure bypass timer and the ADS timer to time-out. Reactor isolation occurs on loss of coolant outside the containment.

The HPCF and RHR-LPFL discharge pressure settings are used as a permissive for depressurization and are selected to assure that at least one of the three RHR pumps, or one of the two HPCF pumps, has received electrical power, started, and is capable of delivering water into the vessel. The pressure setting is high enough to assure that the pump will deliver at or near rated flow without being so high as to fail to show that the pump is actually running.

The level transmitters used to initiate one ADS logic are separated from those used to initiate the other ADS logic. Reactor vessel low water level is detected by eight transmitters that measure differential pressure. Drywell high pressure is detected by four pressure transmitters. All the vessel level and drywell high pressure transmitters are located in the reactor building outside the drywell. The drywell high pressure signals are arranged to "seal-in" the control circuitry. They must be manually reset to clear.

Time delay logic is used in each ADS control division. The time delay setting before actuation of the ADS is long enough that the HPCF and/or RCIC have time to restore water level, if capable, yet not so long that the RHR (LPFL-mode) is unable to adequately cool the fuel if the HPCF fails to prevent low water level. An annunciator in the control room is actuated when either of the timers is timing. Resetting the ADS initiating signals has no effect on the timers if the initiating signals are still present.

If the reactor level is restored sufficiently to reset the previous actuation setpoints before the timer times out, the timer automatically resets and auto-depressurization is aborted. Should additional level dips occur across the setpoints, the timer resets with each one.

(b) Logic and Sequencing

Two parameters of initiation signals are used for the ADS: drywell high pressure, and rector vessel low-low water level (Level 1). Two-out-of-four of each set of signals must be present throughout the timing sequence to cause the safety/relief valves to open. Each parameter separately seals itself in and annunciates following the two-out-of-four logic confirmation. Low water level 1 is the final sensor to initiate the ADS.

A permissive signal of RHR (LP flooder mode) or HPCF pump discharge pressure is also used. Discharge pressure on any one of the three RHR pumps or one of the two HPCF pumps is sufficient to give the permissive signal which permits automatic depressurization when the RHR or HPCF systems are operable.

After receipt of the initiation signals and after a delay provided by time delay elements, each of the two solenoid pilot gas valves is energized. This allows pneumatic pressure from the accumulator to act on the gas cylinder operator. The gas cylinder operator opens and holds the relief valve open. Lights in the main control room indicate when the solenoid-operated pilot valves are energized to open a safety/relief valve. Linear variable differential transformers (LVDT's) mounted on the valve operators verify each valve position to the performance monitoring and control system (PMCS), and the annunciators.

The ADS Division I control logic actuates a solenoid pilot valve on each ADS valve. Similarly, the ADS Division II control logic actuates a second separate solenoid pilot valve on each ADS valve. Actuation of either solenoid-pilot valve causes the ADS valve to open to provide depressurization.

Manual reset circuits are provided for

the ADS initiation signal and the two parameter sensor input logic signals. An attempted reset has no effect if the two-out-of-four initiation signals are still present from each parameter (high drywell pressure and low-low reactor water level). However, a keylocked inhibit switch is provided for each division which can be used to take one ADS division out of service for testing or maintenance during plant operation. This switch is ineffective once the ADS timers have timed out and thus cannot be used to abort and reclose the valves once they are signalled to open. The inhibit mode is continuously annunciated in the main control room.

Manual actuation pushbuttons are provided to allow the operator to initiate ADS immediately (no time delay) if required. Such initiation is performed by first rotating the collars surrounding the pushbuttons for each of two channels within one of the two divisions. An annunciator will sound to warn the operator that ADS is armed for that division. If the two pushbuttons are then depressed, the ADS valves will open, provided the ECCS pump(s) running permissives are present. Though such manual action is immediate, the rotating collar permissives and duality of button sets combined with annunciators assure manual initiation of ADS to be a deliberate act.

A control switch is available in the main control room for each safety/relief valve including the ones associated with the ADS. Each switch is associated with one of the four electrical divisions and maintains electrical separation consistent with the required operability though its function is not required for safety. The switches are three-position keylock-type, OFF-AUTO-OPEN, located on the main control board. The OPEN position is for manual safety/relief valve operation. Manual opening of the relief valves provides a controlled nuclear system cooldown under conditions where the normal heat sink is not available.

(c) Bypasses and Interlocks

There is one manual ADS inhibit switch in the control room for each ADS logic and control division which will inhibit ADS initiation, if ADS has not initiated. The primary purpose of the inhibit switch is to remove one of the two ADS logic and control divisions from service for testing and maintenance during plant operation. Automatic ADS is interlocked with the HPCF and RHR by means of pressure sensors located on the discharge of these pumps. Manual ADS bypasses these interlocks and the timers and immediately opens the ADS valves. The rotating collar permissives and duality of button sets combined with annunciators assure manual initiation of ADS to be a deliberate act.

(d) Redundancy and Diversity

The ADS is initiated by high drywell pressure and/or low reactor vessel water level. The initiating circuits for each of these parameters are redundant as described by the circuit description of this section. Diversity is provided by HPCF.

(e) Actuated Devices

Safety/relief valves are actuated by any one of four methods.

(1) ADS Action

Automatic action after high drywell pressure followed by 29 seconds at low water level (L1) or low water level (L1) for 8 minutes (ADS high drywell pressure bypass timer) and 29 seconds (ADS timer), plus makeup pumps running, resulting from the logic chains in either Division I or Division II control logic actuating;

(2) Manual

Manual action by the operator (either by ADS system level actuation, or by individual SRV operating switches);

(3) Pressure Relief Action

Pressure transmitter signals above setpoints as a result of high reactor pressure (see Paragraph (4)); or

(4) Safety/Relief Action

Mechanical actuation as a result of high reactor pressure (higher than pressure in item 3).

(f) Separation

Separation of the ADS is in accordance with criteria stated in Section 7.1. ADS is Division I (ADS 1) and Division II (ADS 2) system except that only one set of relief valves in supplied. Each ADS relief valve can be actuated by any one of three solenoid pilot valves supplying nitrogen gas to the relief valve gas piston operators. One of the ADS solenoid pilot valves is operated by Division I logic and the other by Division II logic. The third solenoid pilot is used for nonsafety relief valve operation. Control logic manual controls and instrumentation are mounted so that Division I and Division II separation is maintained. Separation from Divisions III and IV is likewise maintained.

(g) Testability

ADS has two complete control logics, one in Division I and one in Division II. Each control logic has two circuits, both of which must operate to initiate ADS. One circuit contains time delay logic to give HPCF an opportunity to start. The ADS instrument channels signals are verified by cross comparison between the channels which bear a known relationship to each other. Indication for each instrument channel is available on displays associated with the SSLC. The logic is tested continuously by automatic self-test circuits. The STS, the sixth test, discussed in RPS testability (7.1.2.1.6) is also applicable here for ADS. The instrument channels are automatically verified every ten minutes as explained in that section. Testing of ADS does not interfere with automatic operation if required by an initiation signal. The pilot solenoid valves can be tested when the reactor is not pressurized.

(h) Environmental Considerations

The signal cables, solenoid valves, safety/relief valve operators and accumulators, and RV low-water level instrument lines are the only essential control and instrumentation equipment for the ADS located inside the drywell. These items will operate in the most severe environment resulting from a design basis loss-of-coolant accident (Section 3.11). Gamma and neutron radiation is also considered in the selection of these items. Equipment located outside the drywell (viz., the RV level and DW pressure transmitters and multiplex interfaces) will also operate in their normal and accident environments.

(i) Operational Considerations

The instrumentation and controls of the ADS are not required for normal plant operations. When automatic depressurization is required, it will be initiated automatically by the circuits described in this section. No operator action is required for at least 30 minutes following initiation of the system. 4/30/90

A temperature element is installed on the safety/relief valve discharge piping several feet from the valve body. The temperature element provides input to a multipoint recorder and interfaces with the PMCS computer in the control room to

FIGURE 7.3-1 HIGH PRESSURE CORE FLOODER IBD

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Figure 7.3-2a NUCLEAR BOILER SYSTEM IBD, SHEET 1

FIGURE 7.3-3 REACTOR CORE ISOLATION COOLING SYSTEM IBD

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FIGURE 7.3-5 LEAK DETECTION & ISOLATION SYSTEM IBD

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7.3-104.29	20	7.3-104.68	20
7.3-104.30	20	7.3-104.69	20
7.3-104.31	20	7.3-104.70	20
7.3-104.32	20	7.3-104.71	20
7.3-104.33	20	7.3-104.72	20
7.3-104.34	20	7.3-104.73	20
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FIGURE 7.3-6 STANDBY GAS TREATMENT SYSTEM

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FIGURE 7.3-7 REACTOR BUILDING COOLING WATER SYSTEM

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Figure 7.3-8 ESSENTIAL HVAC SYSTEM IBD

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FIGURE 7.3-9 HVAC EMERGENCY COOLING WATER

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FIGURE 7.3-10 HIGH PRESSURE NITROGEN GAS

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7.4 SYSTEMS REQUIRED FOR SAFE SHUTDOWN

7.4.1 Description

This section examines and discusses the instrumentation and control aspects of the following plant systems and functions designed to assure safe and orderly shutdown of the ABWR:

(1) Alternate rod insertion function (ARI)

(2) Standby liquid control system (SLCS)

(3) Reactor shutdown cooling mode (RHR)

(4) Remote shutdown system (RSS)

See Subsection 7.1.2.4 which addresses the design basis information required by Section 3 of IEEE 279.

7.4.1.1 Alternate Rod Insertion Function-Instrumentation and Controls

The alternate rod insertion (ARI) function is accomplished independently and diversely from the reactor protection system (RPS). Independent sensors (i.e., ECCS sensors) provide reactor trip signals, via the recirculation flow control system (RFCS), both to ARI valves (part of the control rod drive system) and to the rod control and information system (RC&IS). The ARI valves, (seperate from the scram valves), cause reactor shut-down by hydraulic scram of the control rods. The RC&IS, acting upon the same ARI signals that are provided to ARI valves, cuases reactor shut-down by electromechanical (i.e., through the usage of FMCRD motors) insertion of control rods. 420.15

The RC&IS, including the active run-in function of the FMCRD motors and the ARI valves are not required for safety, nor are these components qualified in accordance with safety criteria. However, the FMCRD components associated with hydraulic scram are qualified in accordance with safety criteria.

The inherent diversity of ARI provides mitigation of the consequences of ATWS (anticipated transient without scram) events.

7.4.1.2 Standby Liquid Control System-Instrumentation and Controls

(1) Function

The instrumentation and controls for the SLCS are designed to initiate and continue injection of a liquid neutron absorber into the reactor when manually or automatically called upon to do so. This equipment also provides the necessary controls to maintain this liquid chemical solution well above saturation temperature in readiness for injection. The system P&ID is shown in Figure 9.3-1. The interlock block diagram (IBD) is shown in Figure 7.4-1.

(2) Classification

The SLCS is a backup method to automatically shut down the reactor to cold subcritical conditions by independent means other than the normal method by the control rod system. Thus, the system is considered a safe shutdown system. The standby liquid control process equipment, instrumentation, and controls essential for injection of the neutron absorber solution into the reactor are designed to withstand Seismic Category I earthquake loads. Any nondirect process equipment, instrumentation, and controls of the system are not required to meet Seismic Category I requirements; however, the local and control room mounted equipment is located in seismically qualified panels.

(3) Power Sources

The power supply to one motor-operated injection valve, storage tank discharge valve, and injection pump is powered from Division I, 480VAC. The power supply to the other motor-operated injection valve, storage tank outlet valve, and injection pump is powered from Division II, 480VAC. The power supply to the tank heaters and heater controls is connectable to a standby AC power source. The standby power source is Class 1E from an onsite source and is independent of the offsite power. The power supply to the main control room benchboard indicator lights and the level and pressure sensors is powered from a Class 1E instrument bus.

(4) Equipment

The SLCS is a special plant-capability event system. No single active component failure of any plant system or component would necessitate the need for the operational

function of the SLCS. It is included for a number of special consideration events:

(a) Plant capability to shut down the reactor without control rods from normal operation (Chapter 15).

(b) Plant capability to shut down the reactor without control rods from a transient incident (Chapter 15).

Although this system has been designed to a high degree of reliability with many safety system features, it is not required to meet the safety design basis requirements of the safety-related systems.

(5) Initiating Circuits

The standby liquid control system is automatically initiated upon receiving an anticipated transient without scram (ATWS) signal.

The standby liquid control system is initiated manually in the main control room by turning a keylocking switch for system A or a different keylocking switch for system B to the RUN position.

(6) Logic and Sequencing

When one division of SLCS is initiated, one injection valve and one tank discharge valve start to open immediately. The pump that has been selected for injection will not start until its associated tank discharge valve is at the fully open position. In order to provide maximum MOV availability when the SLCS is in normal standby readiness, the overloads for the storage tank outlet valves are bypassed by a contact from a test switch in its NORMAL position. When the TEST position is selected, the overload short is removed, thus allowing motor protection during test operation of the valves.

(7) Bypasses and Interlocks

Pumps are interlocked so that either the storage tank discharge valve or the test tank discharge valve must be fully open for the pump to run. When the SLCS is initiated to inject the neutron absorber into the reactor, the outboard isolation valve of the reactor water cleanup system is automatically closed from the Division I logic and the inboard valve from Division II logic.

(8) Redundancy and Diversity

Under special shutdown conditions, the SLCS is functionally redundant to the control rod drive system in achieving and maintaining the reactor subcritical. Therefore, the SLCS as a system by itself is not required to be redundant, although the active components and control channels are redundant for serviceability.

The SLCS provides a diverse means for shutting down the reactor using a liquid neutron absorber in the event of a control rod drive system failure.

The method of identifying redundant power cables, signal cables, and cable trays and the method of identifying non-safety-related cables as associated circuits are discussed in Subsection 8.3.1.3.

(9) Actuated Devices

When the SLCS is automatically initiated to inject a liquid neutron absorber into the reactor, the following devices are actuated:

(a) the two injection valves are opened;

(b) the two storage tank discharge valves are opened;

(c) the two injection pumps are started; and

(d) the reactor water cleanup isolation valves are closed.

When the SLCS is initiated to inject a liquid neutron absorber into the reactor, the following devices are actuated:

(a) one of the two injection valves is opened;

(b) one of the two storage tank discharge valves is opened;

(c) one of the two injection pumps is started; and

(d) one of the reactor water cleanup isolation valves is closed.

Additionally, the pressure and tank level sensing equipment indicates that the SLCS is pumping liquid into the reactor.

(10) Separation

The SLCS is separated both physically and electrically from the control rod drive system. The SLCS electrical control channels are separated in accordance with the requirements of Subsection 8.3.1.4.

(11) Testability

The SLCS is capable of being tested by manual initiation of actuated devices during normal operation. In the test mode, demineralized water is circulated in the SLCS loops rather than sodium pentaborate. During reactor shutdown, demineralized water may be injected into the reactor vessel for the injection test mode.

(12) Environmental Considerations

The environmental considerations for the instrument and control portions of the SLCS are the same as for the active mechanical components of the system (Section 3.11). The instrument and control portions of the SLCS are seismically qualified not to fail during, and to remain functional following, a safe shutdown earthquake (SSE) (see Section 3.10 for seismic qualification aspects).

(13) Operational Considerations

The control scheme for the SLCS can be found in the interlock block diagram (Figure 7.4-1). The SLCS is automatically initiated upon receiving an ATWS signal or can be manually initiated in the control room by inserting the key in the A or B keylocking switch and turning it to the "pump run" position. It will take between 50 and 150 minutes to complete the injection and for the storage tank level sensors to indicate that the storage tank is dry. When the injection is completed, the system automatically shutdown on low tank level or may be manually turned off by turning the keylocking switch counterclockwise to the STOP position.

(14) Reactor Operator Information

(a) The following items are located in the control room for operation information:

(1) Analog Indication

(i) Storage tank level and temperature;

(ii) System pressures;

(2) Status Lights

(i) Pump or storage tank outlet valve overload trip or power loss;

(ii) Position of injection line manual service valve;

(iii) Position of storage tank outlet valve and in-test status;

(iv) Position of test tank discharge manual service valve;

(v) SLCS manually out of service;

(vi) Pump auto trip.

(3) Annunciators

The SLCS annunciators indicate:

(i) Manual or automatic out-of-service condition of SLCS A and/or B due to:

- operation of manual out-of-service switch;

- storage tank outlet valve in test status; or

- overload trip or power loss in pump or storage tank outlet valve controls,

(ii) Standby liquid storage tank high or low temperature;

(iii) Standby liquid tank high or low level;

(iv) Standby liquid pump A (B) auto trip.

(b) The following items are located locally at the equipment for operator utilization:

(1) Analog Indication

(i) Storage tank level;

(ii) System pressures;

(iii) Storage tank temperature.

(2) Indicating lamps

(i) Pump status;

(ii) Storage tank operating heater status;

(iii) Storage tank mixing heater status.

(15) Setpoints

The SLCS has setpoints for the various instruments as follows:

(a) The high and low standby liquid temperature switch is set to activate the annunciator at temperatures outside the range allowed for correct chemical balance of the boron concentration.

(b) The high and low standby liquid storage tank level switch is set to activate the annunciator when the level is outside its allowable limits.

(c) The thermostatic controller and operating heater assure the temperature of the liquid is maintained within the range allowed for correct chemical balance of the boron concentration.

The technical specifications for the SLCS are in Chapter 16.

7.4.1.3 Reactor Shutdown Cooling Mode - Instrumentation and Controls

(1) Function

The shutdown cooling mode of the RHR system is used during the normal or emergency reactor shutdown and cooldown. The RHR system P&ID is Figure 5.4-10 and the RHR system IBD is Figure 7.3-4.

The initial phase of the shutdown cooling mode is accomplished following insertion of the control rods and steam blowdown to the main condenser which serves as the heat sink.

Reactor shutdown cooling has three independent loops. Each loop consists of pump, valves, heat exchanger, and instrumentation designed to provide decay heat removal capability for the core. This mode specifically accomplishes the following:

(a) Reactor Shutdown - removes enough residual heat (decay and sensible) from the reactor vessel water to cool it to 125°F within 20 hours after the control rods are inserted, then maintains or reduces this temperature so that the reactor can be refueled and serviced. This mode is manually activated with the reactor pressure below 135 psig, with all three shutdown cooling loops available.

(b) Safe Shutdown (Emergency Shutdown) - brings the reactor to a cold shutdown condition (< 212°F) within 36 hours after control rod insertion. This mode is manually activated with the reactor pressure below 135 psig, with two-out-of-three shutdown cooling loops available.

The RHRS mode can accomplish its design objective by a preferred means by directly extracting reactor vessel water from the vessel shutdown nozzle and routing it to a heat exchanger and back to the vessel. Cooling water is returned to the vessel via the feedwater line (Loop A) and via the core cooling injection nozzles (Loops B and C).

(2) Classification

Electrical components for the reactor shutdown cooling mode of the residual heat removal system are safety-related and are classified as Class 1E.

(3) Power Sources

This system utilizes normal plant power

leaving the main control room. If this was not possible, the capability of opening the RPS logic input power breakers from outside the main control room can be used as a backup means to achieve initial reactor reactivity shutdown.

(7) The main turbine pressure regulators may be controlling reactor pressure via the bypass valves. However, in the interest of demonstrating that the plant can accommodate even loss of the turbine controls, it is assumed that this turbine generator control panel function is also lost. Therefore, main steamline isolation is assumed to occur at a specified low turbine inlet pressure and reactor pressure is relieved through the relief valves to the suppression pool.

(8) The reactor feedwater system which is normally available is also assumed to be inoperable. Reactor water is made up by the HPCF system.

(9) It shall be assumed that the event causing the evacuation will not cause any failure of the DC or AC control power supplies to the remote shutdown panels or any failure of the DC or AC power feeds to the equipment whose functions are being controlled from the remote shutdown panels.

The above initial conditions and associated assumptions are very severe and conservatively bound any similar postulated situation.

7.4.1.4.3 Remote Shutdown Capability Description

(1) The capability described provides remote control for reactor systems needed to carry out the shutdown function from outside the main control room and bring the reactor to cold condition in an orderly fashion.

(2) It provides a variation to the normal system used in the main control room permitting the shutdown of the reactor when feedwater is unavailable and the normal heat sinks (turbine and condenser) are lost.

(3) Reactor pressure will be controlled and core decay and sensible heat rejected to the suppression pool by relieving steam pressure through the automatic activation of relief valves. Reactor water inventory will be maintained by the HPCF system. During this phase of shutdown, the suppression pool will be cooled by operating the residual heat removal (RHR) system in the suppression pool cooling mode.

(4) Manual operation of the relief valves will cool the reactor and reduce its pressure at a controlled rate until reactor pressure becomes so low that HPCF system operation is discontinued.

(5) The RHR system will then be operated in the shutdown cooling mode using the RHR system heat exchanger in the reactor water circuit to bring the reactor to the cold low pressure condition.

7.4.1.4.4 Remote Shutdown Capability Controls and Instrumentation - Equipment, Panels, and Displays

(1) Main Control Room - Remote Shutdown Capability Interconnection Design Considerations

Some of the existing systems used for normal reactor shutdown operations are also utilized in the remote shutdown capability to shut down the reactor from outside the main control room. The functions needed for remote shutdown control are provided with manual transfer devices which override controls from the main control room and transfer the controls to the remote shutdown control. Control and process sensor signals are interrupted by the transfer devices at the hardwired, analog loop. Sensor signals which interface with the remote shutdown system are routed from the sensor, through the transfer devices on the remote shutdown panels, and then to the multiplexing system remote multiplexing units (RMUs) for transmission to the main control room. Similary, control signals from the main control room are routed from the RMUs, through the remote shutdown transfer devices, and then to the interfacing system equipment. Actuation of the transfer devices interrupts the connection to the RMUs and transfers control

to the remote shutdown system. All necessary power supply circuits are also transferred to other sources. Remote shutdown control is not possible without actuation of the transfer devices. Operation of the transfer devices causes an alarm in the main control room. The remote shutdown control panels are located outside the main control room. Access to this point is administratively and procedurally controlled.

Instrumentation and controls located on the remote shutdown control panels are shown in instrument and electrical diagram Figure 7.4-2.

(2) High Pressure Core Flooder (HPCF)

(a) The following HPCF system loop B equipment functions have transfer and control switches located on the Division II remote shutdown control panel:

(i) valve (pump suction from condensate storage)

(ii) valve (HPCF injection/shutoff)

(iii) valve (minimum flow to suppression pool)

(iv) valve (test line isolation)

(v) valve (pump suction from suppression pool)

(vi) HPCF Pump (B) start/stop

(See HPCF P&ID in Section 6.3.)

(b) The following HPCF System instrumentation is provided on the Division II remote shutdown control panel:

(i) HPCF flow indication

(ii) Indicating lights for all valve (with RSS interface) positions and for the HPCF pump B stop/run

(3) Residual Heat Removal (RHR) System

(a) The following RHR system equipment functions have transfer and control switches located on one or both remote shutdown panels as indicated:

(i) Residual heat removal pump A, B

(ii) valve (suppression pool suction) A, B

(iii) valve (heat exchanger bypass) A, B

(iv) valve (shutdown cooling injection) A, B

(v) valve (heat exchanger inlet) A, B

(vi) valve (suppression pool injection) A, B

(vii) valve (shutdown cooling suction - inboard isolation) A, B

(viii) valve (shutdown cooling suction - outboard isolation) A, B

(ix) valve (shutdown cooling suction) A, B

(x) valve (minimum flow) A, B

(xi) valve (liquid waste flush isolation) A, B

(xii) valve (drywell spray) B

(xiii) valve (wetwell spray) B

(xiv) valve (fuel pool cooling isolation) B

(xv) valve (fuel pool cooling isolation) B

(b) The following RHR instrumentation is located on both remote shutdown control panels as indicated:

(i) RHR flow indicator and transmitter (A,B)

(ii) RHR heat exchanger inlet temperature indication (A,B)

(iii) RHR heat exchanger inlet valve position (A,B)

(iv) RHR heat exchanger bypass valve position (A,B)

(4) Nuclear Boiler System

(a) The following functions have transfer and control switches located at the Division I remote shutdown control panel:

Three air-operated relief valves. (The valves are 125-volt DC solenoid pilot operated.)

FIGURE 7.4-1 STANDBY LIQUID CONTROL SYSTEM IBD

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Figure 7.4-2a REMOTE SHUTDOWN SYSTEM IED, SHEET 1

FIGURE 7.6-1 NEUTRON MONITORING SYSTEM IED

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FIGURE 7.6-2 NEUTRON MONITORING SYSTEM IBD

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FIGURE 7.6-5 PROCESS RADIATION MONITORING SYSTEM IED

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Figure 7.6-6a FUEL POOL COOLING AND CLEANUP SYSTEM IBD, SHEET 1

FIGURE 7.6-7 CONTAINMENT ATMOSPHERE MONITORING SYSTEM IED

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FIGURE 7.6-8 CONTAINMENT ATMOSPHERE MONITORING SYSTEM IBD

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two channel operability. The operator has the capability to invoke bypass conditions within the following system or subsystems:

(a) Synchro A or B position bypass
(b) Rod server module channel A or B bypass
(c) Uncoupled condition bypass
(d) File control module channel A or B bypass
(e) ARBM channel A or B bypass
(f) MRBM channel A or B bypass
(g) RPC channel A or B bypass
(h) RACS channel A or B bypass
(i) DAM channel A or B bypass

(11) Scram Time Test Data Recording

The logic of the RC&IS provides the capability to automatically record individual fine motion control rod drive (FMCRD) scram timing data based upon scram timing reed switches. When a particular FMCRD scram timing switch is activated, the time of actuation is recorded by DAMs for time tagging of stored scram time test data for that particular fine motion control rod drive. The time tagged data is stored in memory until the next actuation of that particular reed switch is detected again.

The RC&IS also time tags the receipt of a reactor scram condition being activated based upon the scram following function input signals from the reactor protection system which is received via the essential multiplexing system.

The resolution of this time tagging feature is less than 10 milliseconds. Contact bounce of the reed switch inputs and the DAM inputs are properly masked to support this function. The reference real time clock for time tagging is the real time clock of the RC&IS.

When RC&IS detects a reactor scram condition, the current position of all control rods in the core are recorded, time tagged, and stored in memory. RC&IS logic stores this data in memory until a request is received from the performance monitoring and control system. The transmitted data is used by the PMCS to calculate and summarize scram time performance based on the scram timing data received from the RC&IS.

(12) ATLM Algorithm Description

The ATLM is a microprocesser based subsystem of RC&IS that executes two different algorithms for enforcing fuel operating thermal limits. One algorithm enforces operating limit minimum critical power ratio (OLMCPR), and the other the operating limit minimum linear heat generation rate (OLMLHGR). For OLMCPR algorithm, the core is divided into 48 regions; each region consisting of 16 fuel bundles. For OLMLHGR algorithm, each region is further vertically divided up into four segments. During a calculation cycle of ATL (about 100 milliseconds), rod block setpoints (RBS) are calculated for OLMCP monitoring (48 values) and for OLMLHGR monitoring (48 x 4 values). Then the calculated setpoints are compared with the real time averaged LPRM readings for each region/segment. The ATLM issues a trip signal if any regionally averaged LPRM reading exceeds the calculated RBS. This trip signal causes a rod block within RC&IS and also a flow change block in the recirculation flow control system (RFCS).

Provided below is a summary description of OLMCPR and OLMLHGR RBS calculation methodology.

(a) OLMCPR RBS Calculation Methodology. The 16 fuel bundles of each region are surrounded by four LPRM strings. There are four LPRMs in each string. For regional OLMCPR monitoring, the sum of the average of each level of B, C, and D of the four LPRM strings is used. The formula for calculating the OLMCPR RBS is:

$$RBS_0 = \frac{LPRM_i * A_0 * RMCPR_1}{OLMCPR} \quad (1)$$

where:

RBS_0 : Operating limit rod block setpoint.

$LPRM_i$: Initial sum of average of four LPRMs of B, C, and D levels that surround each region.

A_o : Margin factor for operating limit rod block; a known function of rod pull distance.

$RMCPR_i$: Regional initial MCPR, i.e., the minimum CPR of the 16 bundles in the region spanned by the four LPRM strings. Known input from predictor (process computer).

OLMCPR : Operating limit MCPR in the current cycle; a known function of power.

Formula (1) is applicable to cases where there is no core flow change and when only one control rod is moved. Adjustments are made to the calculated RBS_o to account for changes in core flow and adjacent control rods movements.

(b) OLMLHGR RBS Calculation Methodology. The formula for calculating the OLMLHGR RBS is:

$$RBS_m(X) = \frac{LPRM_i(X) * B_m * M_p}{MAPRAT_i(X)} \quad (2)$$

where:

$RBS_m(X)$: Calculated operating limit maximum average planar linear heat generation rate (OLMAPLHGR) RBS at LPRM level X.

$LPRM_i(X)$: Initial average of the four LPRMs (level X) at the four corners of each 16 bundle fuel region. The region monitored by the level LPRM is the region covered up to 1.5 feet above and below the LPRM (3 feet total).

B(X) : Margin factor for MAPLHGR operating limit rod block for X level LPRMs. A known function of power and rod position.

M_p : Off-rated power factor to consider overpower condition during worst transient at off-rated condition. A known function of power.

$MAPRAT_i(X)$: Regional initial maximum MAPRAT for level X, i.e., the maximum MAPRAT of the 16 bundles within the 3 feet section covered by the X level LPRMs. A known input from 3D monitor.

In formulas (1) and (2) above, "initial" refers to values that are downloaded from the "3D Predictor Monitor" subsystem of the plant performance monitoring & control system (PMCS). A download is requested by ATLM whenever changes in reactor power and/or core flow exceed a preset limit. A download can also be manually requested by the operator.

7.7.1.2.2 Other Systems Interfaces

(1) Alternate Rod Insertion (ATWS) (Anticipated Transient Without Scram)

The RC&IS logic, during an anticipated transient without scram (on receipt of signals as a result of high reactor dome pressure or low reactor water level) initiates ARI signals which controls the fine motion control rod drive motors such that all control rods are driven to their full-in position automatically. The four divisions of the nuclear boiler system provide each of the two channels of the RC&IS logic with the reactor high dome pressure and reactor low water level

signals for generation of the ARI signal based on two-out-of-four logic.

The operator at the RC&IS dedicated display can take action and initiate the ARI function. Two manual actions are required to manually initiate ARI. The RC&IS logic has been designed to complete the ARI functions in the worst case non-accident environment, completely independent of reactor pressure transient conditions. This capability is accomplished with control logic for insertion of all control rods by an alternate and diverse method, based on receiving reactor high dome pressure and low water level (Level 2) signals for generating its own ATWS (anticipated transient without scram) signal. The logic of the RC&IS has been designed such that no single failure results in failure to insert more than one operable control rod when the ARI function is activated.

(2) Recirculation Flow Control System

The recirculation flow control system (RFCS) provides each of the two channels of the RC&IS with two separate isolated trip signals indicating the need for automatic selected control rod run-in (SCRRI). The signals are treated as nonsafety-related signals within the logic of the RC&IS. 440.3

The RFCS provides signals to both channels

of the RC&IS that represent validated total core flow. These signals are used for part of the validity checks when performing an ARBM operating limit setpoint update. The RC&IS can obtain these signals from the RFCS via the multiplexing system of direct communication links to the RC&IS channels. These signals are also completely independent from the process computer system.

The RFCS receives reference power level signals from the neutron monitoring system and compares the reference power level signals with the nominal power level setpoint.

Selected control rod run in (SCRRI) is automatically initiated when a trip of two or more reactor internal pumps (RIPs) occur. This function is part of the stability control and protection logic.

When two or more RIPs are tripped, the trip signal is "ANDED" with the power level "AND" flow rate signals and RFCS automatically sends a request for control rod blocks to the RC&IS. When the power level signal with two or more RIPs tripped is "ANDED" with the flow rate the RFCS automatically sends four signals to the RC&IS to initiate the SCRRI function. 440.3

The SCRRI function is bypassed when power level is below the specified setpoint, or when the core flow is above the specified setpoint.

The SCRRI function if designed as a non-safety related system. The function is designed to meet the reliability requirement that no single failure shall cause a loss of the function.

The RFCS automatic initiation signal for the SCRRI function is sent as two independent sets of signals, one set to each channel of the RC&IS, each channel of the RC&IS uses the input in two-out-of-two logic to control the fine motion control rod drive (FMCRD) motors of preselected control rods. The preselected control rods are driven to their full-in position on receipt of the automatic initiation signals. Either channel of an RC&IS is capable of initiating the SCRRI function on receipt of the automatic signal from the RFCS.

The preselected control rods for an SCRRI function are selected at the RC&IS dedicated operators control panel and the CRT displays of the performance monitor control system in the main control room. The preselected SCRRI rod data is stored in memory in the rod action and position information subsystem of the RC&IS. The total control rod worth for the preselected control rods is designed to bring down the reactor power level from the 100% rod line to the 80% line.

The RC&IS dedicated operators control panel also provides control switches that requires two manual operator actions for the operator to manually initiate the SCRRI function. 440.3

For the manual or the automatic initiation of the SCRRI function the RC&IS dedicated operators panel provides status indications and alarm annunciators in the control room.

The RC&IS provides the capability for manual or automatic initiation of the SCRRI function and the total delay time to start of control rod motion for the preselected control rods is less than 350 milliseconds.

(3) Feedwater Control System

The feedwater control system provides signals to both channels of the logic of the RC&IS that represents validated total feedwater flow to the vessel, validated narrow range vessel dome pressure, and validated feedwater temperature. These signals are used as part of the validity checks when performing an ARBM operating limit setpoint update.

The RC&IS can obtain these signals from the feedwater control system via the multiplexing system of direct communication links to the RC&IS channels. These signals are also completely independent from the process computer system.

(4) Neutron Monitoring System

Each of the four divisions of the neutron monitoring system provides independent signals to both channels of the RC&IS that indicate when the following conditions are active:

FIGURE 7.7-4 CONTROL ROD DRIVE SYSTEM IBD

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Figure 7.7-5a RECIRCULATION FLOW CONTROL SYSTEM IED, SHEET 1

SECTION 7.8

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7.8 INTERFACES

7.8.1 Effects of Station Blackout on the HVAC

A temperature heat rise analysis shall be performed for the station blackout scenario applied to the control room on consideration of the environmental temperatures unique to the plant location. [See Chapter 20, NRC Question 420.14 and Subsection 7.1.2.3.9]

7.8.2 Deleted

7.8.3 Localized High Heat Spots in Semiconductor Materials for Computing Devices

The response to NRC Question 420.92 provides recommendations for limiting high cuurent densities whigh could result in localized heat spots in semiconductor materials used in computing devices. The applicant shall provide assurance that these recommendation are followed, or an acceptable alternative is presented, by the selected equipment vendor(s). To ensure that adequate compensation for heat rise is incorporated into the design, a thermal analysis shall be performed at the circuit board, instrument and panel design stages. [See Chapter 20, NRC Question 420.92]

7.8.4 Safety-Related C&I Interfaces

Each of the systems addressed in Chapter 7 were reviewd for safety-related C&I (signal) interfaces which extend outside the scope of the ABWR Standard Plant. Since the scope of the ABWR Standard Plant includes all of the reactor building, the turbine building and the control building, the study determined there are no safety-related electrical signal interfaces for any of these systems which extend beyond the scope definition.

7A.2 MULTIPLEXING SYSTEMS

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9.1.3 Fuel Pool Cooling and Cleanup System

9.1.3.1 Design Bases

The fuel pool cooling and cleanup (FPC) system shall be designed to remove the decay heat from the fuel pool, maintain pool water level and quality and remove radioactive materials from the pool to minimize the release of radioactivity to the environs.

The FPC system shall:

(1) minimize corrosion product buildup and shall control water clarity, so that the fuel assemblies can be efficiently handled underwater;

(2) minimize fission product concentration in the water which could be released from the pool to the reactor building environment;

(3) monitor fuel pool water level and maintain a water level above the fuel sufficient to provide shielding for normal building occupancy;

(4) maintain the pool water temperature below 125°F under normal operating conditions. The temperature limit of 125°F is set to establish an acceptable environment for personnel working in the vicinity of the fuel pool. The design basis for the FPC system is to provide cooling after closure of the fuel gates (21 days) at the normal heat load from spent fuel stored in the pool is the sum of decay heat of the most recent 35% batch plus the heat from the previous 4 fuel batches after closure of the fuel gates. The RHR system will be used to supplement the FPC system under the maximum load condition as defined in Subsection 9.1.3.2.

9.1.3.2 System Description

The FPC system (Figures 9.1-1 a and b, and 9.1-2 and Table 9.1-11) maintains the spent fuel storage pool below the desired temperature at an acceptable radiation level and at a degree of clarity necessary to transfer and service the fuel bundles.

The FPC system cools the fuel storage pool by transferring the spent fuel decay heat through two 6.55×10^6 Btu/hr heat exchangers to the reactor building closed cooling water system (RCW). Each of the two heat exchangers is designed to transfer one half the system design heat load. The system utilizes two parallel 250 m^3/hr pumps to provide a system design flow of 500 m^3/hr. Each pump is suitable for continuous duty operation. The equipment is located in the reactor building.

The system pool water temperature is maintained at or below 125°F. The decay heat released from the stored fuel is transferred to the RCW. During refueling prior to 21 days following shutdown, the reactor (shutdown cooling) and fuel pool cooling are provided jointly by the residual heat removal (RHR) and FPC systems in parallel. The reactor cavity communicates with the fuel pool since the reactor well is flooded and the fuel gates are open. RHR suction is taken from the vessel shutdown suction lines, pumped through RHR heat exchangers and discharged into the upper pools to improve water clarity for refueling. For the FPC system, fuel pool water is circulated by means of overflow through skimmers around the periphery of the pool and a scupper at the end of the transfer pool drain tanks, pumped through the FPC heat exchangers and filter-demineralizers and back to the pool through the pool diffusers.

After 21 days, the fuel gates are closed. At this point, the FPC system, solely provides, the fuel pool cooling function. However, when the reactor is defueled more than the design-basis 35% batch (maximum heat load condition), RHR can provide supplemental cooling. RHR supplemental cooling suction is taken from the skimmer surge tank, passed through RHR heat exchanger and back to the fuel pool.

Clarity and purity of the pool water are maintained by a combination of filtering and ion exchange. The filter-demineralizers maintain total corrosion product metals at 1.0 ppb or less with pH range of 5.6 to 8.6 at 25°F for compatibility with fuel storage racks and other equipments. Conductivity is maintained at less than 1.2 μS/cm at 25°C and chlorides less 281.12 281.11

than 20 ppb. Each filter unit in the filter-demineralizer subsystem has adequate capacity to maintain the desired purity level of the pools under normal operating conditions. The flow rate is designed to be approximately that required for two complete water changes per day for the fuel transfer and storage pools. The maximum system flow rate is twice that needed to maintain the specified water quality.

The FPC system is designed to remove suspended or dissolved impurities from the following sources:

(1) dust or other airborne particles;

(2) surface dirt dislodged from equipment immersed in the pool;

(3) crud and fission products emanating from the reactor or fuel bundles during refueling;

(4) debris from inspection or disposal operations; and

(5) residual cleaning chemicals or flush water.

A post-strainer in the effluent stream of the filter-demineralizer limits the migration of filter material. The filter-holding element can withstand a differential pressure greater than the developed pump head for the system.

The filter-demineralizer units are located separately in shielded cells with enough clearance to permit removing filter elements from the vessels.

Each cell contains only the filter-demineralizer and piping. All valves (inlet, outlet, recycle, vent, drain, etc.) are located on the outside of one shielding wall of the room, together with necessary piping and headers, instrument elements and controls. Penetrations through shielding walls are located so as not to compromise radiation shielding requirements.

The filter-demineralizers are controlled from a local panel. A differential pressure and conductivity instruments provided for each filter-demineralizer unit indicate when backwash is required. Suitable alarms, differential pressure indicators and flow indicators monitor the condition of the filter-demineralizers.

System instrumentation is provided for both automatic and remote-manual operations. A low-low level switch stops the circulating pumps when the fuel pool drain tank reserve capacity is reduced to the volume that can be pumped in approximately one minute with one pump at rated capacity (250 m^3/hr). A level switch is provided in the fuel pool to alarm locally and in the control room on high and low level. A temperature element is provided to display pool temperature in the main control room. In addition, leakage flow detectors in the pool drains and pool liners are provided and alarmed in the control room.

The circulating pumps are controlled from the control room and a local panel. Pump low suction pressure automatically turns off the pumps. A pump low discharge pressure alarm is indicated in the control room and on the local panel. The circulating pump motors can be powered from the diesel- generators if normal power is not available. Circulating pump motor loads are considered nonessential loads and will be operated as required under accident conditions.

The water level in the spent fuel storage pool is maintained at a height which is sufficient to provide shielding for normal building occupancy. Radioactive particulates removed from the fuel pool are collected in filter-demineralizer units which are located in shielded cells. For these reasons, the exposure of plant personnel to radiation from the FPC system is minimal. Further details of radiological considerations for this system are described in Chapter 12.

The circulation patterns within the reactor well and spent fuel storage pool are established by placing the diffusers and skimmers so that particles dislodged during refueling operations are swept away from the work area and out of the pools.

Check valves prevent the pool from siphoning in the event of a pipe rupture.

Heat from pool evaporation is handled by the building ventilation system. Makeup water is provided through a remote-operated valve.

9.1.3.3 Safety Evaluation

The maximum possible heat load for the FPC system upon closure of the fuel gates (21 days) is the decay heat of the full core load of fuel at the end of the fuel cycle plus the remaining decay heat of the spent fuel discharged at previous refuelings upon closure of the fuel gates; the maximum capacity of the spent fuel storage pool is 270% of a core. The temperature of the fuel pool water may be permitted to rise to approximately 140°F under these conditions. During cold shutdown conditions, if it appears that the fuel pool temperature will exceed 125°F, the operator can connect the FPC system to the RHR system. Combining the capacities enables the two systems to keep the

water temperature below 125°F. The RHR system will be used only to supplement the fuel pool cooling when the reactor is shut down. The reactor will not be started up whenever portions of the RHR system are needed to cool the fuel pool. The connecting piping from the fuel storage pool to the RHR system is designed Seismic Category I and can be isolated, assuming a single active failure, from the remainder of the fuel pool system.

These connections may also be utilized during emergency conditions to assure cooling of the spent fuel regardless of the availability of the fuel pool cooling system. The volume of water in the storage pool is such that there is enough heat absorption capability to allow sufficient time for switching over to the RHR system for emergency cooling.

During the initial stages of refueling, the reactor cavity communicates with the fuel pool since the reactor well is flooded and the fuel pool gates are open. Decay heat removal is provided jointly by RHR and FPC systems and the pool temperature kept below 140°F. Evaluation studies concluded that after 150 hours decay following shutdown (fuel pool gates open), the combined decay heat removal capacity of the 1-RHR and 1-FPC heat exchangers (single active failure postulated) can keep the pool temperature well below 140°F. The RHR-FPC joint decay heat removal performance evaluation is shown in Table 9.1-12.

The 140°F temperature limit is set to assure that the fuel building environment does not exceed equipment environmental limits.

The spent fuel storage pool is designed so that no single failure of structures or equipment will cause inability to: (1) maintain irradiated fuel submerged in water; (2) re-establish normal fuel pool water level; or (3) remove decay heat from the pool. In order to limit the possibility of pool leakage around pool penetrations, the pool is lined with stainless steel. In addition to providing a high degree of integrity, the lining is designed to withstand abuse that might occur when equipment is moved about. No inlets, outlets or drains are provided that might permit the pool to be drained below a safe shielding level. Lines extending below this level are equipped with siphon breakers, check valves, or other suitable devices to prevent inadvertent pool drainage. Interconnected drainage paths are provided behind the liner welds. These paths are designed to: (1) prevent pressure buildup behind the liner plate; (2) prevent the uncontrolled loss of contaminated pool water to other relatively cleaner locations within the containment or fuel-handling area; and (3) provide liner leak detection and measurement. These drainage paths are designed to permit free gravity drainage to the equipment drain tanks or sumps of sufficient capcity and/or pumped to the liquid radwaste facility.

A makeup water system and pool water level instrumentation are provided to replace evaporative and leakage losses. Makeup water during normal operation will be supplied from condensate. The suppression pool cleanup system can be used as a Seismic Category I source of makeup water in case of failure of the normal makeup water system utilizing either one of the two Seismic Category I makeup lines: (1) the makeup line to the spent fuel pool storage (2) or the makeup line to the dryer/separator storage pool. Makeup water from the dryer/separator pool is delivered to the spent fuel pool by opening pool gates to the spent fuel pool.

Both FPC and SPCU systems are Seismic Category I Quality Group C design with the exception of the filter demineralizer portion which is shared by both systems. Following an accident or seismic event, the filter demineralizers are isolated from FPCS cooling portion and the SPCU by two block valves in series at both the inlet and outlet of the common filter demineralizer portion. Seismic Category I Quality Group C bypass lines are provided on both FPC and SPCU systems to allow continued flow of cooling and makeup water to the spent fuel pool.

Connections from the RHR system to the FPC system provide a Seismic Category I, safety-related makeup capability to the spent fuel pool. The FPC system from the RHR connections to the spent fuel pool are Seismic Category I, safety-related.

Furthermore, firehoses can be used as an alternate makeup source. The fire protection standpipes in the reactor building and their

water supply (yard main, one motor driven pump and water source) are seismically designed. The motor driven pump is powered from a bus which has a safety-related diesel generator as one of its power sources. A second seismically designed pump, directly driven by a diesel engine is also provided.

The FPC components, housed in the Seismic Category I reactor building, are Seismic Category I, Quality Group C including all components except the filter demineralizer. These components are protected from the effects of natural phenomena, such as: earthquake, external flooding, wind, tornado and external missiles. Inside the reactor building the FPC safety-related components are protected from the effects of pipe whip, internal flooding, internally generated missiles, and the effects of a moderate pipe rupture within the vicinity.

From the foregoing analysis, it is concluded that the FPC system meets its design bases.

9.1.3.4 Inspection and Testing Requirements

No special tests are required because, normally, one pump, one heat exchanger and one filter-demineralizer are operating while fuel is stored in the pool. The spare unit is operated periodically to handle abnormal heat loads or to replace a unit for servicing. Routine visual inspection of the system components, instrumentation and trouble alarms is adequate to verify system operability.

9.1.3.5 Radiological Considerations

The water level in the spent fuel storage pool is maintained at a height which is sufficient to provide shielding for normal building occupancy. Radioactive particulates removed from the fuel pool are collected in filter-demineralizer units which are located in shielded cells. For these reasons, the exposure of plant personnel to radiation from the FPC system is minimal. Further details of radiological considerations for this and other systems are described in Chapters 11, 12, and 15.

9.1.4 Light Load Handling System (Related to Refueling)

9.1.4.1 Design Bases

The fuel-handling system is designed to provide a safe and effective means for transporting and handling fuel from the time it reaches the plant until it leaves the plant after post-irradiation cooling. Safe handling of fuel includes design considerations for maintaining occupational radiation exposures as low as practicable during transportation and handling.

Design criteria for major fuel-handling system equipment are provided in Table 9.1-2 through 9.1-4, which list the safety class, quality group and seismic category. Where applicable, the appropriate ASME, ANSI, Industrial and Electrical Codes are identified. Additional design criteria are shown below and expanded further in Subsection 9.1.4.2.

The transfer of new fuel assemblies between the uncrating area and the new fuel inspection stand and/or the new fuel storage vault is accomplished using 5-ton auxiliary hook on the reactor building crane equipped with a suitable grapple.

The 1,000 pound auxiliary hoist on the reactor building crane is used with an auxiliary fuel grapple to transfer new fuel from the new fuel vault to the fuel storage pool. From this point on, the fuel will either be handled by the telescoping grapples on the refueling platform or auxiliary hoist.

The refueling platform is Seismic Category I from a structural standpoint in accordance with 10CFR50, Appendix A. The refueling platform is constructed in accordance with a quality assurance program that ensures the design, construction and testing requirements are met. Allowable stress due to safe shutdown earthquake (SSE) loading is 120% of yield or 70% of ultimate, whichever is least. A dynamic analysis is performed on the structures using the response spectrum method with load contributions resulting from each of three directions acting simultaneously being combined by the RMS procedure. Working loads of the platform structure are in accordance with the AISC Manual of Steel Construction. All parts of the hoist systems are designed to have a safety factor of at least ten, based on the ultimate strength of the material. A redundant load path is incorporated in the fuel hoists so that no single component failure could result in a fuel bundle drop. Maximum deflection limitations are imposed on the main structures to maintain relative stiffness of the platform. Welding of the platforms is in accordance with AWS D14-1 or ASME Boiler and Pressure Vessel Code Section IX. Gears and bearing meet AGMA Gear Classification Manual and ANSI B3.5. Materials used in construction of load bearing members are to ASTM specifications. For personnel safety, OSHA Part 1910-179 is applied. Electrical equipment and controls meet ANSI CI, National Electric Code, and NEMA Publication No. ICS1, MG1.

The auxiliary fuel grapple and the main telescoping fuel grapple have redundant lifting features and an indicator which confirms positive grapple engagement.

The fuel grapple is used for lifting and transporting fuel bundles. It is designed as a telescoping grapple that can extend to the proper work level and, in its fully retracted state, still maintain adequate water shielding over fuel.

In addition to redundant electrical interlocks to preclude the possibility of raising radioactive material out of the water, the cables on the auxiliary hoists incorporate an adjustable, removal stop that will jam the hoist cable against some part of the platform structure to prevent hoisting when the free end of the cable is at a preset distance below water level.

Provision of a separate cask pit, capable of being isolated from the fuel storage pool, will eliminate the potential accident of dropping the cask and rupturing the fuel storage pool. Furthermore, limitation of the travel of the crane handling the cask will preclude transporting the cask over the spent fuel storage pool.

9.1.4.2 System Description

Table 9.1-5 is a listing of typical tools and servicing equipment supplied with nuclear system. The following paragraphs describe the use of some of the major tools and servicing equipment and address safety aspects of the design where applicable.

Subsection 9.1.5 provides the data that verifies the ABWR Standard Plant heavy load handling systems and satisfies the guidelines of NUREG-0612.

9.1.4.2.1 Spent Fuel Cask

Out of ABWR Standard Plant scope.

9.1.4.2.2 Overhead Bridge Cranes

9.1.4.2.2.1 Reactor Building Crane

The reactor building crane is a seismically analyzed piece of equipment. The crane consists of two crane girders and a trolley which carries two hoists. The runway track, which supports the crane girders, is supported from the reactor building walls at elevation 34,600. The trolley travels laterally on the crane girders carrying the main hoist and auxiliary hoist.

The reactor building crane is used to move all of the major components (reactor vessel head, shroud head and separator, dryer assembly and pool gates) as required by plant operations. The reactor building crane is used for handling new fuel from the reactor building entry hatch to new fuel storage, the new fuel inspection stand and the spent fuel pool. It also is used for handling spent fuel cask. The principal design criteria for the reactor building crane are described in Subsection 9.1.5.

9.1.4.2.3 Fuel Servicing Equipment

The fuel servicing equipment described below has been designed in accordance with the criteria listed in Table 9.1-2. Items not listed as Seismic Category I, such as hoists, tools and other equipment used for servicing shall either be removed during operation, moved to a location where they are not a potential hazard to safety related equipment, or seismically restrained to prevent them from becoming missiles.

9.1.4.2.3.1 Fuel Prep Machine

Two fuel preparation machines (Figure 9.1-3) are mounted on the wall of the spent fuel pool and are used for stripping reusable channels from the spent fuel and for rechanneling of the new fuel. The machines are also used with the fuel inspection fixture to provide an underwater inspection capability.

Each fuel preparation machine consists of a work platform, a frame, and a movable carriage. The frame and movable carriage are located below the normal water level in the spent fuel pool thus providing a water shield for the fuel assemblies being handled. The fuel preparation machine carriage has a permanently installed up-travel-stop to prevent raising fuel above the safe water shield level.

9.1.4.2.3.2 New Fuel Inspection Stand

The new fuel inspection stand (Figure 9.1-4) serves as a support for the new fuel bundles undergoing receiving inspection and provides a working platform for technicians engaged in performing the inspection.

The new fuel inspection stand consists of a vertical guide column, a lift unit to position the work platform at any desired level, bearing seats and upper clamps to hold the fuel bundles in position.

The new fuel inspection stand will be firmly attached to the wall so that it does not fall into or dump personnel into the spent fuel pool during an SSE. (See Subsection 9.1.6.5 for interface requirements.)

9.1.4.2.3.3 Channel Bolt Wrench

The channel bolt wrench (Figure 9.1-5) is a manually operated device approximately 3.76 meters (12 ft) in overall length. The wrench is used for removing and installing the channel fastener assembly while the fuel assembly is held in the fuel preparation machine. The channel bolt wrench has a socket which mates and captures the channel fastener capscrew.

9.1.4.2.3.4 Channel-Handling Tool

The channel-handling tool (Figure 9.1-6) is used in conjunction with the fuel preparation machine to remove, install and transport fuel channels in the spent fuel pool.

The tool is composed of a handling bail, a lock/release knob, extension shaft, angle guides and clamp arms which engage the fuel channel. The clamps are actuated (extended or retracted) by manually rotating lock/release knob.

The channel-handling tool is suspended by its bail from a spring balancer on the channel-handling boom located on the spent fuel pool periphery.

9.1.4.2.3.5 Fuel Pool Vacuum Sipper

The fuel pool vacuum sipper (Figure 9.1-7) provides a means of identifying fuel suspected of having cladding failures. The fuel pool Vacuum sipper consists of a fuel isolation container, fluid console, monitoring console with program controller and beta detector and the inter-connecting tubing and cables. The suspected fuel assembly is placed in the isolation

container. A partial Vacuum is established in the gas volume above the fuel assembly. The fission product gas leakage is sensed by the beta detector and monitoring console.

9.1.4.2.3.6 General Purpose Grapple

The general purpose grapple (Figure 9.1-8) is a handling tool used generally with the fuel. The grapple can be attached to the jib crane to handle fuel during channeling.

9.1.4.2.3.7 (Deleted)

9.1.4.2.3.8 Refueling Platform

Refer to Subsection 9.1.4.2.7 for a description of the refueling platform.

9.1.4.2.3.9 Channel Handling Boom

A channel handling boom (Figure 9.1-10) with a spring-loaded balance reel is used to assist the operator in supporting a portion of the weight of the channel as it is removed from the fuel assembly. The boom is set between the fuel preparation machines. With the channel handling tool attached to the reel, the channel may be conveniently moved between the fuel preparation machines.

9.1.4.2.4 Servicing Aids

General area underwater lights are provided with a suitable reflector for illumination. Suitable light support brackets are furnished to support the lights in the reactor vessel to allow the light to be positioned over the area being serviced independent of the platform. Local area underwater lights are small diameter lights for additional illumination. Drop lights are used for illumination where needed.

A radiation hardened portable underwater closed circuit television camera is provided. The camera may be lowered into the reactor vessel and/or spent fuel pool to assist in the inspection and/or maintenance of these areas.

A general purpose, plastic viewing aid is provided to float on the water surface to provide better visibility. The sides of the viewing aid are brightly colored to allow the operator to observe it in the event of filling with water and sinking. A portable, submersible-type, underwater vacuum cleaner is provided to assist in removing crud and miscellaneous particulate matter from the pool floors or reactor vessel. The pump and the filter unit are completely submersible for extended periods. The filter "package" is capable of being remotely changed, and the filters will fit into a standard shipping container for offsite burial. Fuel pool tool accessories are also provided to meet servicing requirements. A fuel sampler is provided. This is to be used to detect defective fuel assemblies during open vessel periods while the fuel is in the core. The fuel sampler head isolates individual fuel assemblies by sealing the top of the fuel channel and pumping water from the bottom of the fuel assembly, through the fuel channel, to a sampling station, and returning the water to the primary coolant system. After a "soaking" period, a water sample is obtained and is radiochemically analyzed to determine possible fuel bundle leakage.

9.1.4.2.5 Reactor Vessel Servicing Equipment

The essentiality and safety classifications, the quality group, and the seismic category for this equipment are listed in Table 9.1-3. Following is a description of the equipment designs in reference to that table.

9.1.4.2.5.1 Reactor Vessel Service Tools

These tools are used when the reactor is shut down and the reactor vessel head is being removed or reinstalled. Tools in this group are:

Stud Handling Tool

Stud Wrench

Nut Runner

Stud Thread Protector

Thread Protector Mandrel

Bushing Wrench

Seal Surface Protector

Stud Elongation Measuring Rod

Dial Indicator Elongation Measuring Device

Head Guide Cap

RIP Impeller /Shaft Assembly Tool

Impeller Storage Rack.

The tools are designed for a 60-year life in the specified environment. Lifting tools are designed for a safety factor of 10 or better with respect to the ultimate strength of the material used. When carbon steel is used, it is either hard chrome plated, parkerized, or coated with an approved paint per Regulatory Guide 1.54.

9.1.4.2.5.2 Steamline Plug

The steamline plugs are used during reactor refueling or servicing; they are inserted in the steam outlet nozzles from inside of the reactor vessel to prevent a flow of water from the reactor well into the main steamline during servicing of safety relief valves, main isolation valves, or other components of the main steamlines, while the reactor water level is at the refueling level. The steam line plug design provides two seals of different types. Each one is independently capable of holding full head pressure. The equipment is constructed of corrosion resistant materials. All calculated safety factors are 5 or better. The plug body is designed in accordance with the "Aluminum Construction Manual" by the Aluminum Association.

9.1.4.2.5.3 Shroud Head Stud Wrench

This is a hand-held tool for tightening and loosening the shroud head studs. It is designed for a 60-year life and is made of aluminum for easy handling and to resist corrosion. Calculations have been performed to confirm the design.

9.1.4.2.5.4 Head Holding Pedestal

Three pedestals are provided for mounting on the refueling floor for supporting the reactor vessel head and strongback/carousel during periods of reactor service. The pedestals have studs which engage three evenly spaced stud holes in the head flange. The flange surface rests on replaceable wear pad made of aluminum.

When resting on the pedestals, the head flange is approximately 3 ft above the floor to allow access to the seal surface for inspection and O-ring replacement.

The pedestal structure is a carbon steel weldment coated with an approved paint. It has a base with bolt holes for mounting it to the concrete floor.

A seismic analysis was made to determine the seismic forces imposed on the pedestals, floor anchors, using the floor response spectrum method. The structure is designed to withstand these calculated forces and meet the requirements of AISC.

9.1.4.2.5.5 Head Stud Rack

The head stud rack is used for transporting and storage of eight reactor pressure vessel studs. It is suspended from the reactor building crane hook when lifting studs from the reactor well to the operating floor.

The rack is made of aluminum to resist corrosion and is designed for a safety factor of 5 with respect to the ultimate strength of the material.

The structure is designed in accordance with the "Aluminum Construction Manual" by the Aluminum Association.

9.1.4.2.5.6 Dryer and Separator Strongback

The Dryer and Separator Strongback is a lifting device used for transporting the steam dryer or the shroud head with the steam separators between the reactor vessel and the storage pools. The strongback is a cruciform-shaped I-beam structure, which has a hook box with two hook pins in the center for engagement with the reactor building crane sister hook. The strongback has a socket with a pneumatically operated pin on the end of each arm for engaging it to the four lift eyes on the steam dryer or shroud head.

The strongback has been designed such that one hook pin and one main beam of the cruciform will be capable of carrying the total load and so that no single

9.1.6 Interfaces

9.1.6.1 New Fuel Storage Racks Criticality Analysis

The applicant referencing the ABWR design shall provide the NRC confirmatory criticality analysis as required by Subsection 9.1.1.1.1.

9.1.6.2 Dynamic and Impact Analyses of New Fuel Storage Racks

The applicant referencing the ABWR design shall provide the NRC confirmatory dynamic and impact analyses of the new fuel storage racks. See Subsection 9.1.1.1.6.

9.1.6.3 Spent Fuel Storage Racks Criticality Analysis

The applicant referencing the ABWR design shall provide the NRC confirmatory critically analysis as required by Subsection 9.1.2.3.1.

9.1.6.4 Spent Fuel Racks Load Drop Analysis

The applicant referencing the ABWR design shall provide the NRC confirmatory load drop analysis as required by Subsection 9.1.4.3.

9.1.6.5 New Fuel Inspection Stand Seismic Capability

The applicant referencing the ABWR design will install the new fuel inspection stand firmly to the wall so that it does not fall into or dump personnel into the spent fuel pool during an SSE. (See Subsection 9.1.4.2.3.2.)

9.1.7 References

1. *General Electric Standard Application for Reactor Fuel*, (NEDE-24011-P-A, latest approved revision).

(e) MUWC transfer pumps (see Table 9.2-3) (three 550 gpm at 141 psi head)

(3) Water can be sent to the CST from the following sources:

(a) MUWP pumps

(b) CRD system

(c) radwaste disposal system

(d) condensate demineralizer system effluent (main condenser high level relief)

(4) Associated receiving and distribution piping valves, instruments, and controls shall be provided.

(5) Overflow and drain from the CST shall be sent to the radwaste system for treatment.

(6) Any outdoor piping shall be protected from freezing.

(7) All surfaces coming in contact with the condensate shall be made of corrosion-resistant materials.

(8) All of the pumps mentioned in (2) above shall be located at an elevation such that adequate suction head is present at all water levels in the CST.

(9) Instrumentation shall be provided to indicate CST water level in the main control room. High water level shall be alarmed both locally and in the main control room.

(10) Potential flooding is discussed in Subsection 3.4. Potential flooding from lines within the reactor building and the control building are evaluated in Subsection 3.4.1.1.1.

9.2.9.3 Safety Evaluation

Operation of the MUWC system is not required to assure any of the following conditions:

(1) integrity of the reactor coolant pressure boundary;

(2) capability to shut down the reactor and maintain it in a safe shutdown condition; or

(3) ability to prevent or mitigate the consequences of events that could result in potential offsite exposures.

The MUWC system is not safety-related. However, the systems incorporate features that assure reliable operation over the full range of normal plant operations.

9.2.9.4 Tests and Inspections

The MUWC system is proved operable by its use during normal plant operation. Portions of the system normally closed to flow can be tested to ensure operability and the integrity of the system.

The air-operated isolation valves are capable of being tested to assure their operating integrity by manual actuation of a switch located in the control room and by observation of associated position indication lights.

Flow to the various systems is balanced by means of manual valves at the individual takeoff points. Divisional isolation valves are installed at the primary containment boundaries.

9.2.10 Makeup Water System (Purified) Distribution System

9.2.10.1 Design Bases

(1) The makeup water-purified (MUWP) distribution system shall provide makeup water purified for makeup to the reactor coolant system and plant auxiliary systems.

(2) The MUWP system shall provide purified water to the uses shown in Table 9.2-2.

(3) The MUWP system shall provide water of the quality shown in Table 9.2-2a. If these water quality requirements are not met, the water shall not be used in any safety-related system. The out-of-spec water shall be reprocessed or discharged. 410.51

(4) The MUWP system is not safety-related.

(5) All tanks, pumps, piping, and other equipment shall be made of corrosion-resistant materials.

(6) The system shall be designed to prevent any radioactive contamination of the purified water.

(7) The interfaces between the MUWP system and all safety-related systems are located in the control building or reator building which are Seismic Category I, tornado-missile resistant and flood protected structures. The interfaces with safety-related systems are safety-related valves which are part of the safety-related systems. The portions of the MUWP system, which upon their failure during a seismic event can adversly impact structures, systems, or components important to safety, shall be designed to assure their integrity under seismic loading resulting from a safe shutdown earthquake.

(8) Safey-related equipment located by portions of the MUWP system are in Seismic Category I structures and protected from all system impact.

9.2.10.2 System Description

410.50 The MUWP system P&ID is shown in Figure 9.2-5. This system includes the following:

(1) Any purified water storage tank shall be provided outdoors with adequate freeze protection and adequate diking and other means to control spill and leakage.

(2) Two MUWP forwarding pumps shall take suction from any purified water storage tanks. They shall have a capacity of 308 gpm and a discharge head of 114 psi.

(3) Distribution piping, valves, instruments and controls shall be provided.

(4) Any outdoor piping shall be protected from freezing.

(5) All surfaces coming in contact with the purified water shall be made of corrosion-resistant materials.

(6) All pumps shall be located at an elevation such that adequate suction head is present at all levels in a purified water storage tanks.

(7) Instruments shall be provided to indicate purified water storage tank level in the main control room.

(8) Continuous analyzers are located at the demineralized water makeup system and at any demineralized water storage tank. These are supplemented as needed by grab samples. Allowance is made in the water quality specifications for some pickup of carbon dioxide and air in any demineralized water storage tank. The pickup of corrision products should be minimal because the MUWP piping is stainless steel.

(9) Intrusions of radioactivity into the MUWP system from other potentially radioactive systems are prevented by one or more of the following:

(a) check valves in the MUWP lines

(b) air (or syphon) breaks in the MUWP lines

(c) the MUWP system lines are pressurized while the receiving system is at essentially atmospheric pressure.

(d) piping to the user is dead ended.

(10) There are no automatic valves in the MUWP system. During a LOCA, the safety-related systems are isolated from the MUWP system by automatic valves in the safety-related system.

9.2.10.3 Safety Evaluation

Operation of the MUWP system is not required to assure any of the following conditions:

(1) integrity of the reactor coolant pressure boundary;

(2) capability to shut down the reactor and maintain it in a safe shutdown condition; or

(3) ability to prevent or mitigate the consequences of events which could result in potential offsite exposures.

The MUWP system is not safety-related. However, the systems incorporate features that assure reliable operation over the full range of normal plant operations.

9.2.10.4 Tests and Inspections

The makeup water purified distribution system is proved operable by its use during normal plant operation. Proportions of the system normally closed to flow can be tested to ensure operability and integrity of the system.

Flow to the various systems is balanced by means of manual valves at the individual takeoff points.

9.2.11 Reactor Building Cooling Water System

9.2.11.1 Design Bases

9.2.11.1.1 Safety Design Bases

(1) The reactor building cooling water (RCW) system shall be designed to remove heat from plant auxiliaries which are required for a safe reactor shutdown, as well as those auxiliaries whose operation is desired following a LOCA, but not essential to safe shutdown.

The heat removal capacity is based on the heat removal requirement during LOCA with the maximum ultimate heat sink temperature, 95°F. As shown in Table 9.2-4, the heat removal requirement is higher during other plant operation modes, such as shutdown at 4 hours. However, the RCW system is designed to remove this larger amount of heat to meet the requirements in Subsection 5.4.7.1.1.7.

(2) The RCW system shall be designed to perform its required cooling functions following a LOCA, assuming a single active or passive failure.

(3) The safety-related portions and valves isolating the nonsafety-related portions of

9.2.1.1.3.2 Safety Evaluation of Equipment

Equipment served by the RCW system is listed in Tables 9.2-4a, b, and c. The tables contain five operating modes:

(1) normal operation;

(2) shutdown at [illegible];

(3) shutdown at 20 hr.;

(4) hot standby (No LOPP);

(5) hot standby (LOPP); and

(6) post-LOCA.

The flow rates and heat loads are given for each equipment in each operating mode.

In the event of a LOCA, most of the nonessential cooling water uses are isolated by proper isolation valves. The instrument air system, service air system, control rod drive pump oil cooler and the reactor water cleanup system pump coolers remain in service until the operator removes them from service. The nonsafety-related portion of the system is automatically isolated in the event of a rupture in the nonsafety-related subsystem. The surge tank water level is monitored. A level switch is activated by a significant leak, sending an isolation signal to close two valves. One valve on the supply line and one valve on the discharge line are used, with suitable power and controls from divisional sources to assure isolation in the event of any single active failure. Single isolation valves are used on the basis that an active failure of one isolation valve disables only that system of which it was a part.

The RCW system is designed to withstand a single active failure without losing its capability to participate in the safe shutdown of the reactor following a LOCA or DBA. Table 9.2-5 gives the result of a system failure analysis of active and passive components.

Redundant trains of the RCW system are separated and protected to the extent necessary to assure that sufficient equipment remains operating to permit shutdown of the unit in the event of any of the following (separation is applied to electrical equipment and instrumentation and controls as well as to mechanical equipment and piping):

(1) flooding, spraying, or steam release due to pipe rupture or equipment failure;

(2) pipe whip and jet forces resulting from postulated pipe rupture of nearby high energy pipes;

(3) missiles which may result from equipment failure;

(4) fire; and

(5) failures of any non-Category I equipment (pertains to Seismic Category I equipment).

Radiation monitors are provided to sample the RCW cooling water. Upon detection of radiation leakage in one of the systems, that system is isolated by operator action from the control room, and the total cooling load can be met by the other two systems. Consequently, radioactive contamination released by the RCW system to the environment does not exceed allowable limits defined by 10CFR100.

The safety-related parts of the RCW system are designed to Seismic Category I and ASME Code, Section III, Class 3, Quality Assurance B and Quality Group C requirements. The design also meets IEEE-279 and IEEE-308 requirements. Isolation valves for nonsafety-related service water systems also meet the above requirements.

The nonessential portion of the RCW system is designed to the ANSI B31.1 Power Piping Code and the requirements of Quality Group D.

The design pressure and temperature of the RCW system and piping are 14 kg/cm^{2}g (200 psig) and 70°C (158°F) maximum.

System low point drains and high point vents are provided as required.

All divisions are maintained full of water when not in service except when undergoing maintenance.

System components and piping materials are selected where required to be compatible with the available site cooling water in order to minimize corrosion. Cathodic protection of the tubing side of the heat exchanger shall be provided. Adequate corrosion safety factors are used to assure the integrity of the system during the life of the plant.

During all plant operating modes, all divisions have at least one RCW cooling water pump operating. Therefore, if a LOCA occurs, the RCW cooling water system required to shut down the plant safely is already in operation. If a loss of offsite power occurs during a LOCA, the pumps momentarily stop until transfer to standby diesel generator power is completed. The pumps are restarted automatically according to the diesel loading sequence. If a LOCA occurs, most nonsafety-related components are automatically isolated from the RCW system. Consequently, no operator action is required, following a LOCA, to start the RCW system in its LOCA operating mode.

All heat exchangers and pumps will be required during the following plant operating conditions, in addition to LOCA: shutdown at 4 hours, shutdown at 20 hours and hot standby with loss of AC power.

Loss of one RCW division will result in loss of RCW cooling to every other RIP (five total) as shown on RRS P&ID (Figure 5.4-4) and will cause those five RIPs to runback to minimum speed. The RIP M-G set in the same electrical division, which is cooled by the same RCW division which failed and powers two more RIPs, would stop by M-G set cooling water prptection. This would completely shutdown three RIPs and would have the resulting total of seven RIPs either at minimum speed or stopped. Assuming the event began at full power on the 100% Control Rod Line, the resulting temporary reactor power would be approximately 60% power. The operator would then correct the RCW problem or initiate a normal plant shutdown.

Complete failure of any RCW division will reduce drywell cooling, but, not enough to require plant shutdown or power level reduction. Failure of RCW division A would have only one drywewll cooler using RCW cooling and the normal HNCW cooling. Drywell temperatures would not increase enough to adversely affect any drywell components.

The drywell cooling system can perform its function after the loss of any RCW division. With only one RCW division and one drywell cooler operating, the drywell temperature will increase but not to a temperature that would damage equipment or require an immediate shutdown.

9.2.11.4 Testing and Inspection Requirements

The RCW system is designed to permit periodic in-service inspection of all system components to assure the integrity and capability of the system.

The RCW system is designed for periodic pressure and functional testing to assure: (1) the structural and leaktight integrity by visible inspection of the components; (2) the operability and the performance of the active components of the system; and (3) the operability of the system as a whole.

The tests shall assure, under conditions as close to design as practical, the performance of the full operational sequence that brings the system into operation for reactor shutdown and for LOCA, including operating of applicable portions of the Reactor Protection System and the transfer between normal and standby power sources. These tests shall include periodic testing of the heat removal capability of each RCW heat exchanger. Each of these heat exchangers has been designed to provide 20% margin above the heat removal capability required for LOCA in Tables 9.2-4a, b and c. The revised heat removal capacity of the heat exchangers is shown in Table 9.2-4d. This 20% margin is provided to compensate for the combined effects of fouling and tube plugging. When this margin is no longer present, the heat exchanger heat removal capacity will be increased by either cleaning or retubing.

The RCW system is supplied with a chemical addition tank to add chemicals to each division. The RCW system is initially filled with demineralized water. A corrosion inhibitor can be added if desired. These measures are adequate to protect the RCW system from the ill effects of corrosion or organic fouling.

The RCW system is designed to conform with the foregoing requirements. Initial tests shall be made as described in Subsection 14.2.12.

9.2.11.5 Instrumentation and Control Requirements

All equipment is provided with either globe or butterfly valves to give the capability for manual control. These valves are accessible downstream of the equipment for regulation of flow through the equipment or for balancing the circuits. The isolation valves to the nonessential RCW system are automatically and remote-manually operated.

Pressure taps or indicators at equipment are provided to enable the operator to adjust the differential pressure across each heat exchanger or cooler and also to allow leak checking.

Locally mounted temperature indicators or test wells are furnished on the equipment cooling water discharge lines to enable verification of specified heat removal during plant operation. The required heat removal and flow rates are shown in Tables 9.2-4a, b, and c.

The combination of pressure taps (or indicators) and temperature indicators allow correct system balancing with or without a system heat load. For purposes of system balancing, provisions for flow measurement are provided as required.

Connections to a radiation monitor are provided in each division to detect radioactive contamination resulting from a tube leak in one of the RHR exchangers, fuel pool exchangers, or other exchangers.

Isolation valves for RHR heat exchangers and nonessential cooling water subsystems are provided with remote manual switches and indication on the remote shutdown panel.

evaporator. If the temperature of the chilled water drops below a specified level, the controller automatically adjusts the position of the compressor inlet guide vanes. Flow switches prohibit the chiller from operating unless there is water flow through both evaporator and condenser.

9.2.14 Turbine Building Cooling Water System

9.2.14.1 Design Bases

9.2.14.1.1 Safety Design Bases

The turbine building cooling water (TCW) system serves no safety function and has no safety design basis.

430.208 There are no connections between the TCW system and any other safety-related systems.

9.2.14.1.2 Power Generation Design Bases

(1) The TCW system provides corrosion-inhibited, demineralized cooling water to all turbine island auxiliary equipment listed in Table 9.2-11.

(2) During power operation, the TCW system operates to provide a continuous supply of cooling water, at a maximum temperature of 105°F, to the turbine island auxiliary equipment, with a service water inlet temperature not exceeding 95°F.

(3) The TCW system is designed to permit the maintenance of any single active component without interruption of the cooling function.

(4) Makeup to the TCW system is designed to permit continuous system operation with design failure leakage and to permit expeditious post-maintenance system refill.

(5) The TCW system is designed to have an atmospheric surge tank located at the highest point in the system.

(6) The TCW system is designed to have a higher pressure than the power cycle heat sink water to ensure leakage is from the TCW system to the power cycle heat sink in the event a tube leak occurs in the TCW system heat exchanger.

9.2.14.2 System Description

9.2.14.2.1 General Description

The TCW system is illustrated on Figure 9.2-6. The system is a single loop system and consists of one surge tank, one chemical addition tank, two pumps with a capacity of 29,000 gpm each, two heat exchangers with heat removal capacity of 130 x 10^6 Btu/h each. (connected in parallel), and associated coolers, piping, valves, controls, and instrumentation. Heat is removed from the TCW system and transferred to the non-safety related turbine service water system (Subsection 9.2.16). 430.207b

A TCW system sample is periodically taken for analysis to assure that the water quality meets the chemical specifications.

9.2.14.2.2 Component Description

Codes and standards applicable to the TCW system are listed in Table 3.2-1. The system is designed in accordance with quality group D specifications.

The chemical addition tank is located in the turbine building in close proximity to the TCW system surge tank.

The TCW pumps are 100% capacity each and are constant speed electric motor driven, horizontal centrifugal pumps. The two pumps are connected in parallel with common suction and discharge lines.

The TCW heat exchangers are 100% capacity each and are designed to have the TCW water circulated on the shell side and the power cycle heat sink water circulated on the tube side. The surface area is based on normal heat load.

The surge tank, which is shared between the HNCW and TCW systems, is an atmospheric carbon steel tank located at the highest point in the TCW system. The surge tank is provided with a level control valve that controls makeup water addition.

The surge tank is located above the TCW pumps and heat exchangers in the turbine building in a

location away from any safety-related components. Failure of the surge tank will not affect any safety-related systems.

4NO-207a

Those parts of the TCW system in the turbine building are located in areas that do not contain any safety-related systems. All safety-related systems in the turbine building are located in special areas to prevent any damage from non-safety-related systems during seismic events. Those parts of the TCW system outside the turbine building are located away from any safety-related systems.

9.2.14.2.3 System Operation

During normal power operation, one of the two 100% capacity TCW system pumps circulate

9.2.17 Interfaces

9.2.17.1 Ultimate Heat Sink Capability

Interface requirements pertaining to ultimate heat sink capability are delineated in Subsection 9.2.5 as follows:

Subsection	Title
9.2.5.1	Safety Design Bases
9.2.5.2	Power Generation Design Bases
9.2.5.6	Evaluation of UHS Performance
9.2.5.7	Safety Evaluation
9.2.5.8	Conformance to Regulatory Guide 1.27
9.2.5.9	Instrumentation and Alarms
9.2.5.10	Tests and Inspections

If any spray pond piping is made from fiberglass-reinforced thermosetting resin, the applicant shall provide information to show that all applicable requirements of Regulatory Guide 1.72 are met.

9.2.17.2 Makeup Water System Capability

The raw water treatment and preparation of the demineralized water is sent to the makeup water system (purified) described in Subsection 9.2.10.

The demineralized water preparation system shall consist of at least two divisions capable producing at least 200 gpm of demineralized water each. Storage of demineralized water shall at least 200,000 gallons. If additional demineralized water is needed during peak usage ds, rented portable demineralizers shall be as required.

makeup water preparation system shall be in a building which does not contain any related structures, systems or nts. If the system is not available, ized water can be obtained from mobile The system shall be designed so that any failure in the system, including any that cause flooding, shall not result in the failure of any safety-related structure, system or component.

9.2.17.3 Potable and Sanitary Water System

The potable and sanitary water system shall be designed with no interconnections with systems having the potential for containing radioactive materials. Protection shall be provided through the use of air gaps, where necessary. (See Subsection 9.2.4).

9.2.17.4 Reactor Service Water System

The RSW pumps are described in Table 9.2-13. The applicant shall provide the following additional information which is site dependent: (See Subsection 9.2.15.2 and 9.2.15.3).

(1) temperature increase and pressure drop across the heat exchangers;

(2) the required and available net positive suction head for the RSW pumps at pump suction locations considering anticipated low water levels;

(3) the location of the RSW pump house;

(4) the design features to assure that the requirements in Subsection 9.2.15.1.1(3) are met; and

(5) an analysis of a pipeline break and a single active component failure shall show that flooding shall not affect the main control room or more than one division of the RSW system.

9.2.17.5 Turbine Service Water System

The applicant shall demonstrate that all safety-related components, systems, and structures are protected from flooding in the event of a pipeline break in the TSW system. (See Subsection 9.2.16.3)

9.2.17 Interfaces

9.2.17.1 Ultimate Heat Sink Capability

Interface requirements pertaining to ultimate heat sink capability are delineated in Subsection 9.2.5 as follows:

Subsection	Title
9.2.5.1	Safety Design Bases
9.2.5.2	Power Generation Design Bases
9.2.5.6	Evaluation of UHS Performance
9.2.5.7	Safety Evaluation
9.2.5.8	Conformance to Regulatory Guide 1.27
9.2.5.9	Instrumentation and Alarms
9.2.5.10	Tests and Inspections

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If any spray pond piping is made from fiberglass-reinforced thermosetting resin, the applicant shall provide information to show that all applicable requirements of Regulatory Guide 1.72 are met.

9.2.17.2 Makeup Water System Capability

The raw water treatment and preparation of the demineralized water is sent to the makeup water system (purified) described in Subsection 9.2.10.

The demineralized water preparation system shall consist of at least two divisions capable of producing at least 200 gpm of demineralized water each. Storage of demineralized water shall be at least 200,000 gallons. If additional demineralized water is needed during peak usage periods, rented portable demineralizers shall be used as required.

430.205

The makeup water preparation system shall be located in a building which does not contain any safety-related structures, systems or components. If the system is not available, demineralized water can be obtained from mobile equipment. The system shall be designed so that any failure in the system, including any that cause flooding, shall not result in the failure of any safety-related structure, system or component.

9.2.17.3 Potable and Sanitary Water System

The potable and sanitary water system shall be designed with no interconnections with systems having the potential for containing radioactive materials. Protection shall be provided through the use of air gaps, where necessary. (See Subsection 9.2.4).

9.2.17.4 Reactor Service Water System

The RSW pumps are described in Table 9.2-13. The applicant shall provide the following additional information which is site dependent: (See Subsection 9.2.15.2 and 9.2.15.3).

(1) temperature increase and pressure drop across the heat exchangers;

(2) the required and available net positive suction head for the RSW pumps at pump suction locations considering anticipated low water levels;

(3) the location of the RSW pump house;

(4) the design features to assure that the requirements in Subsection 9.2.15.1.1(3) are met; and

(5) an analysis of a pipeline break and a single active component failure shall show that flooding shall not affect the main control room or more than one division of the RSW system.

430.201

9.2.17.5 Turbine Service Water System

The applicant shall demonstrate that all safety-related components, systems, and structures are protected from flooding in the event of a pipeline break in the TSW system. (See Subsection 9.2.16.3)

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SECTION 9.3

TABLES

ILLUSTRATIONS

9.3.3 Equipment and Floor Drainage System

The system which collects and transfers all radioactive liquid wastes is discussed in Subsection 9.3.8. The drainage systems for non-radioactive liquid wastes are not discussed because they are not a part of the ABWR Standard Plant.

9.3.4 Chemical and Volume Control System(PWR)

(Not applicable to a BWR)

9.3.5 Standby Liquid Control System

9.3.5.1 Design Bases

9.3.5.1.1 Safety Design Bases

The standby liquid control system (SLCS) has a safety-related function and is designed as a Seismic Category I system. It shall meet the following safety design bases:

(1) Backup capability for reactivity control shall be provided, independent of normal reactivity control provisions in the nuclear reactor, to be able to shut down the reactor if normal control ever becomes inoperative.

(2) The backup system shall have the capacity for controlling the reactivity difference between the steady-state rated operating condition of the reactor with voids and the cold shutdown condition, including shutdown margin, to assure complete shutdown from the most reactive conditions at any time in core life.

(3) The time required for actuation and effectiveness of the backup control shall be consistent with the nuclear reactivity rate of change predicted between rated operating and cold shutdown conditions. A fast scram of the reactor or operational control of fast reactivity transients is not specified to be accomplished by this system.

(4) Means shall be provided by which the functional performance capability of the backup control system components can be verified periodically under conditions approaching actual use requirements. Demineralized water, rather than the actual neutron absorber solution, can be injected into the reactor to test the operation of all components of the redundant control system.

(5) The neutron absorber shall be dispersed within the reactor core in sufficient quantity to provide a reasonable margin for leakage or imperfect mixing.

(6) The system shall be reliable to a degree consistent with its role as a special safety system; the possibility of unintentional or accidental shutdown of the reactor by this system shall be minimized.

9.3.5.2 System Description

The SLCS (Figure 9.3-1) is automatically initiated or can be manually initiated through the keyboard switches in the main control room to pump a boron neutron absorber solution into the reactor if the operator determines the reactor cannot be shut down or kept shut down with the control rods. Once the operator decision for initiation of the SLCS is made, the design intent is to simplify the manual process by providing dual keylocked switches. This prevents inadvertent injection of neutron absorber by the SLCS. However, the insertion of the control rods is expected to assure prompt shutdown of the reactor should it be required.

The keylocked control room switch is provided to assure positive action from the main control room should the need arise. Procedural controls are applied to the operation of the keylocked control room switch.

The SLCS is required only to shut down the reactor and keep the reactor from going critical again as it cools.

The SLCS is needed only in the improbable event that not enough control rods can be inserted in the reactor core to accomplish shutdown and cooldown in the normal manner.

The boron solution tank, the test water tank, the two positive displacement pumps, the two motor-operated injection valves, the two motor-

operated pump suction valves, and associated local valves, panel, and controls are located in the secondary containment outside the drywell and wetwell. The liquid is piped into the reactor vessel throughout the high pressure core flooder (HPCF) line downstream of the HPCF inboard check valve.

The boron absorbs thermal neutrons and thereby terminates the nuclear fission chain reaction in the uranium fuel.

The specified neutron absorber solution is sodium pentaborate ($Na_2B_{10}O_{16} \cdot 10H_2O$). It is prepared by dissolving stoichiometric quantities of borax and boric acid in demineralized water. An air sparger is provided in the tank for mixing. To prevent system plugging, the tank outlet is raised above the bottom of the tank.

At all times when it is possible to make the reactor critical, the SLCS shall be able to deliver enough sodium pentaborate solution into the reactor (Figure 9.3-2) to assure reactor shutdown. This is accomplished by placing sodium pentaborate in the standby liquid control tank and filling it with demineralized water to at least the low level alarm point. The solution can be diluted with water to within 14 inches of the overflow level volume to allow for evaporation losses or to lower the saturation temperature.

The minimum temperature of the fluid in the tank and piping shall be consistent with that obtained from Figure 9.3-3 for the solution temperature. The saturation temperature of the recommended solution is 59°F at the low level alarm volume and a lower temperature at 14 inches below the tank overflow volume (Figures 9.3-2 and 9.3-3). The equipment containing the solution is installed in a room in which the air temperature is to be maintained within the range of 50 to 100°F. An electrical resistance heater system provides a backup heat source which maintains the solution temperature at 75°F (automatic operation) to 85°F (automatic shutoff) to prevent precipitation of the sodium pentaborate from the solution during storage. High or low temperature, or high or low liquid level, causes an alarm in the control room.

Each positive displacement pump is sized to inject the solution into the reactor in 60 to 150 minutes, independent of the amount of solution in the tank. The pump and system design pressure between the injection valves and the pump and system design pressure between relief valves are approximately 1560 psig. To prevent bypass flow from one pump in case of relief valve failure in the line from the other pump, a check valve is installed downstream of each relief valve line in the pump discharge pipe.

The SLCS is automatically initiated after receiving an anticipated transient without scram (ATWS) signal or can be manually actuated by either of two keylocked, spring-return switches on the control room console. This assures that switching from the OFF position is a deliberate act. Changing either switch status to RUN starts an injection pump, opens one motor-operated injection valve, opens one pump suction motor-operated valve, and closes both of the reactor cleanup system outboard isolation valves to prevent loss of boron.

An ATWS condition exists when either of the following occurs:

(a) High RPV pressure (1125 psig) and average power range monitor (APRM) not down scale for 3 minutes, or

(b) Low RPV level (Level 2) and APRM not down scale for 3 minutes.

A light in the control room indicates that power is available to the pump motor contactor and that the contactor is deenergized (pump not running). Another light indicates that the contactor is energized (pump running).

Storage tank liquid level, tank outlet valve position, pump discharge pressure, and injection valve position indicate that the system is functioning. If any of these items indicates that the liquid may not be flowing, the operator shall immediately change the other switch to the RUN mode, thereby activating the redundant train of the SLCS. The local switch will not have a STOP position. This prevents the isolation of the pump from the control room. Pump discharge pressure and valve status are indicated in the control room.

Equipment drains and tank overflow are not piped to the radwaste system but to separate containers (such as 55 gallon drums) that can be removed and disposed of independently to prevent any trace of boron from inadvertently reaching the reactor.

Instrumentation consisting of solution temperature indication and control, solution level

9.3.9 Hydrogen Water Chemistry System

9.3.9.1 Design Bases

9.3.9.1.2 Safety Design Basis

The hydrogen water chemistry (HWC) system is non-nuclear, non-safety-related and is required to be safe and reliable, consistent with the requirement of using hydrogen gas. The hydrogen piping in the turbine building shall be designed to Seismic Category I requirements to comply with BTP 9.5-1.

9.3.9.1.2 Power Generation Design Basis

BWR reactor coolant is demineralized water, typically containing 100 to 200 parts per billion (ppb) dissolved oxygen from the radiolytic decomposition of water. To mitigate the potential for intergranular stress corrosion cracking (IGSCC) of sensitized austenitic stainless steels, the dissolved oxygen in the reactor water can be reduced to less than 20 ppb by the addition of hydrogen to the feedwater. The amount of hydrogen required is in the range of 1.0 to 1.5 ppm. The exact amount required depends on many factors including incore recirculation rates. The amount required will be determined by tests performed during the initial operation of the plant.

The concentration of hydrogen and oxygen in the main steam line and eventually in the main condenser is altered in this process. This leaves an excess of hydrogen in the main condenser that would not have equivalent oxygen to combine with in the offgas system. To maintain the offgas system near its normal operating characteristics, a flow rate of oxygen equal to approximately one-half the injected hydrogen flow rate is injected in the offgas system upstream of the recombiner.

The HWC system utilizes the guidelines given in EPRI report NP-5283-SR-A, "Guidelines for Permanent BWR Hydrogen Water Chemistry Installation".

9.3.9.2 System Description

The HWC system, illustrated in Figure 9.3-8, is composed of hydrogen and oxygen supply systems, systems to inject hydrogen in the feedwater and oxygen in the offgas and subsystems to monitor the effectiveness of the HWCS system. These systems monitor the oxygen levels in the offgas system, the feedwater system, the lower plenum region and the CUW inlet, hydrogen and pH levels in the feedwater system, the lower plenum region and the CUW inlet, and crack growth of pre-cracked samples in water from the lower plenum region.

The hydrogen supply system will be site dependent. Hydrogen can be supplied either as a high pressure gas or as a cryogenic liquid. Hydrogen and oxygen can also be generated on site by the dissociation of water by electrolysis. The HWC hydrogen supply system is integrated with the generator hydrogen supply system to save the cost of having separate gas storage facilities for both systems.

The oxygen supply system will be site dependent. A single oxygen supply system could be provided to meet the requirements of HWC system and the condensate oxygen injection system described in Subsection 9.3.10.

9.3.9.3 Safety Evaluation

The operation of the HCS is not necessary to assure:

(1) The integrity of the reactor coolant pressure boundary,

(2) The capability to shut down the reactor; or

(3) The capability to prevent or mitigate the consequences of events which could result in potential offsite exposures.

The HWC system is used, along with other measures, to reduce the likelihood of corrosion failures which would adversely affect plant availability. The means of storing and handling hydrogen shall utilize the guidelines in EPRI NP-5283-SR-A, "Guidelines for Permanent BWR Hydrogen Water Chemistry Installations".

9.3.9.4 Inspection and Testing Requirements

The HWC system is proved operable during the initial operation of the plant. During a refueling or maintenance outage, hydrogen injection is not required. System maintenance or testing can be performed during such periods.

9.3.9.5 Instrumentation and Controls

Automatic control features in the HWC system minimize the need for operator attention and improve performance. These are:

(1) Automatic variation of hydrogen and oxygen flow rates with reactor power level.

(2) Automatic oxygen injection rate change delay. This function is also augmented as a function of reactor power level.

(3) Automatic shutdown on several alarms.

(4) Isolation on system power loss, operator restart.

(5) Reprogrammable alarms and controller electronics.

(6) Hydrogen and oxygen flow monitor correction function to compensate for nonlinearities.

The recommended trips of the oxygen and hydrogen injection systems include:

(1) Reactor scram

(2) Low or high residual oxygen in the off-gas

(3) High area hydrogen concentration

(4) Low oxygen injection system supply pressure

(5) High hydrogen flow

The instrumentation provided includes:

(1) Flow monitors for measurement of hydrogen and oxygen flow rates.

(2) Hydrogen area monitor sensors to detect any hydrogen to the atmosphere.

(3) Pressure gages for measurement of hydrogen and oxygen supply pressures and instrument air pressure.

(4) An oxygen analyzer for measuring the percent oxygen leaving the offgas recombiner.

(5) Sensors for measuring dissolved oxygen content.

(6) Sensors for measuring pH and dissolved hydrogen.

(7) A system for verifying the effectiveness of HWC by measuring electrochemical potential (ECP) and crack growth rate.

9.3.10 Oxygen Injection System

9.3.10.1 Design Bases

The oxygen injection system is designed to add sufficient oxygen to the Condensate System to suppress corrosion and corrosion product release in the condensate and feedwater systems. Experience has shown that the preferred feedwater oxygen concentration is 20 to 50 ppb. During shutdown and startup operation the feedwater oxygen concentration is usually much above the 20 to 50 ppb range. However, during power operation, deaeration in the main condenser may reduce the condensate oxygen concentration below 20ppb, thus, requiring that some oxygen be added. The amount required is up to approximately 5 cubic feet per hour.

9.3.10.2 System Description

The oxygen supply consists of high pressure gas cylinders or a liquid tank. A condensate oxygen injection module is provided with pressure regulators and associated piping, valves, and controls to depressurize the gaseous oxygen and route it to the condensate injection modules. There are check valves and isolation valves between the condensate injection modules and the condensate lines downstream of the condensate demineralizers and the optional injection point upstream of the filters.

The flow regulating valves in this system are operated from the main control room. The oxygen concentration in the condensate/feedwater system is monitored by analyzers in the sampling system (Subsection 9.3.2). An operator will make changes in the oxygen injection rate in response to changes in the condensate/feedwater concentration. An automatic control system is not required because instantaneous changes in oxygen injection rate are not required.

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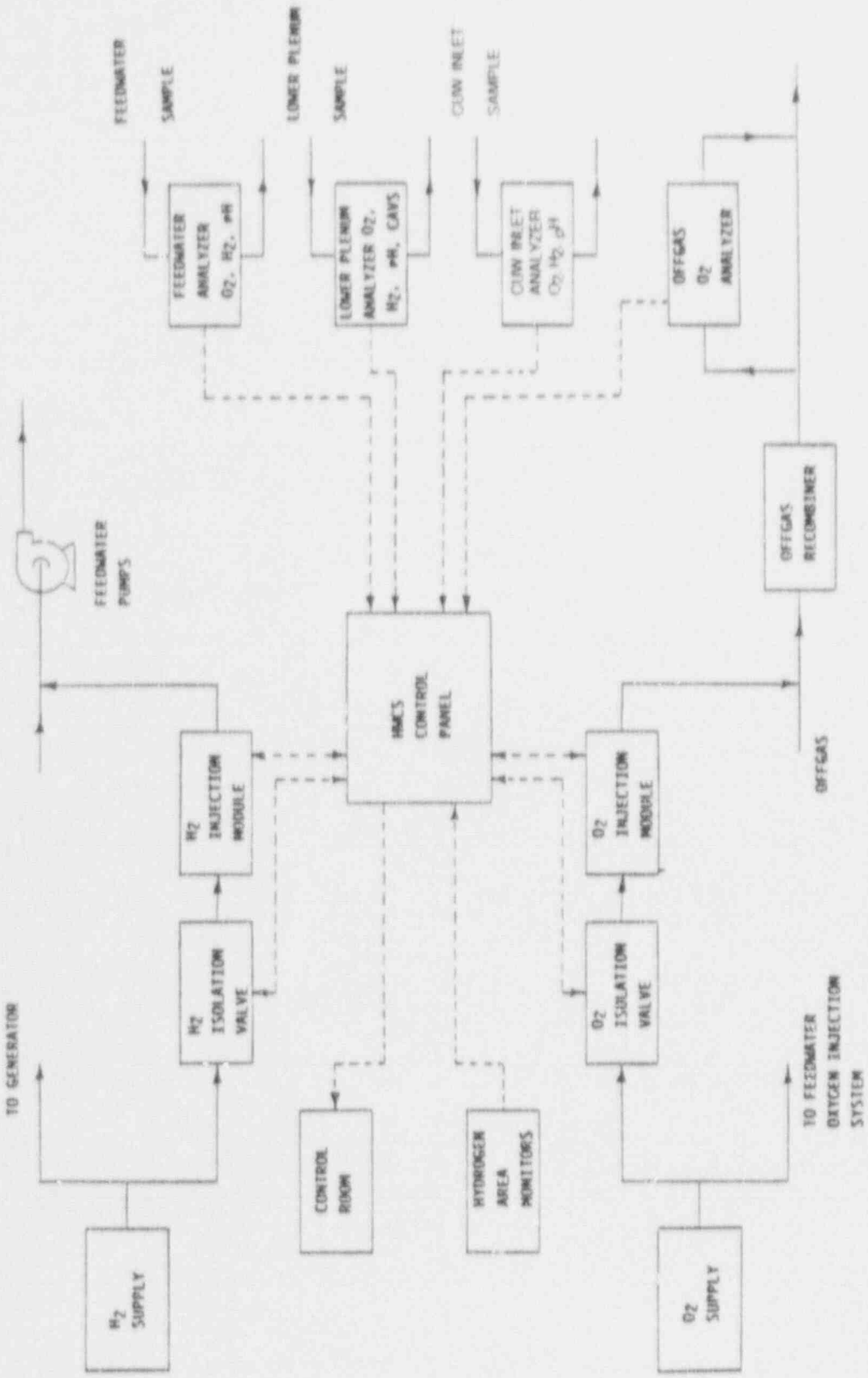

Figure 9.3-8 HYDROGEN WATER CHEMISTRY SYSTEM

9.4.9 Drywell Cooling System

9.4.9.1 Design Bases

The drywell cooling system shall have the capability to maintain the drywell temperature, during normal operation, at temperatures specified in Section 3.11.

The drywell cooling system shall be capable of controlling the temperature rise of the drywell during normal operational transients so that the average drywell temperature does not exceed 135 °F. The local temperature shall not exceed 165 °F in the CRD area or 149 °F elsewhere in the drywell.

The drywell cooling system is designed to provide sufficient air/nitrogen distribution so that proper temperature distribution can be achieved to prevent hot spots from occurring in any area of the drywell.

9.4.9.2 System Description

See Figures 9.4-8 and 9.4-9 for the flow diagram illustrating the drywell cooling system, and Table 9.4-1 for a listing of its components. It is a recirculating system consisting of three fan coil units. Normally, two of the three fan coil units are in operation. Each fan coil unit consists of two cooling coils, a drain pan, and a centrifugal fan. Cooling water comes from the RCW and HNCW systems. The two cooling coils are arranged in series. The return air passes over the first coil, which is cooled by RCW. Part of the cooled air is then cooled by the second coil, which is cooled by HNCW. This twice-cooled air is mixed with the air which bypassed the second cooling coil. Condensate that drips from the coils is routed to the DRW drain system via the leak detection system. Instrumentation is installed in the drain line, in front of the leak detection system connection that monitors cooler condensate flow.

The drywell cooling system supplies conditioned air to a common distribution header. The air/nitrogen is then ducted to areas within the drywell for equipment cooling. These areas consist of the drywell head area, upper drywell, lower drywell, shield wall annulus, and the wetwell air space. The drywell cooling system head loads are provided in Table 3.4-2.

Gravity dampers and adjustable volume dampers control distribution of the air/nitrogen to the drywell space.

High drywell temperatures are alarmed in the main control room, alerting the operator to take appropriate corrective action. During normal plant operation, two fan coil units are operated. During LOPA (when no LOCA signal exists), fan coil units shall restart automatically when power is available from the diesel generators. During LOPA, chilled water from the HNCW system will not be available. Chilling will only be available from the RCW coils. The fan coil units are not operated during LOCA.

9.4.9.3 Safety Evaluation

Operation of the drywell cooling system is not a prerequisite to assurance of either one of the following:

(1) integrity of the reactor coolant pressure boundary, or

(2) capability to safely shut down the reactor and to maintain a safe shutdown condition.

However, the system does incorporate features that provide reliability over the full range of normal plant operation. These features include the installation of redundant principal system components such as:

(1) electric power;

(2) fan coil units;

(3) sources of chilled water;

(4) ductwork;

(5) controls; and

(6) cross connection of all fan coil units.

9.4.9.4 Inspection and Testing Requirements

Equipment design includes provisions for periodic testing of functional performance and inspection for system reliability. Standby components are fitted with test connections so that system effectiveness, except for airflow or static pressure, can be verified without the units being online. Test connections are provided in the discharge air ducts for verifying calibration of the operating controls.

4.9.5 Instrumentation Applications

Drywell cooling unit function is manually controlled from the main control room. The instrumentation which monitors system performance is part of the atmospheric control system and the leak detection and isolation system.

FIGURE 9.4-1 CONTROL BUILDING HVAC PROCESS FLOW DIAGRAM

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FIGURE 9.4-3 SECONDARY CONTAINMENT HVAC SYSTEM

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FIGURE 9.4-4 ESSENTIAL ELECTRICAL EQUIPMENT HVAC SYSTEM

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FIGURE 9.4-5 REACTOR INTERNAL PUMP CONTROL PANEL ROOM HVAC SYSTEM

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Figure 9.4-6 ESSENTIAL DIESEL GENERATOR HVAC SYSTEM

power in accordance with RG 1.155. Adequate protection of the CTG against sabotage is provided by locating the unit inside the security protected area.

9.5.11.4 Tests and Inspections

The initial test qualification requirements described in IEEE 387, IEEE Standard Criteria for Diesel Units Applied as Standby Power Supplies for Nuclear Power Generating Stations, shall also be applied to the CTG in order to ensure adequate system reliability. However, the factory-test portion of this requirement may be waived if the identically designed unit has been shown capable of maintaining a reliability of 0.99 over a five-year period.

Site acceptance testing, periodic surveillance testing and preventive maintenance, inspections, etc., shall be performed in accordance with the manufacture's recommendations, including time intervals for parts replacement.

9.5.11.5 Instrumentation Requirements

The CTG is provided with local instrumentation and control systems suitable for manual start-up and shutdown, and for monitoring and control during operation. Automatic start-up and load sequencing is controlled via the control console located in the main control room.

Mechanical and electrical instrumentation linked to control room displays are provided to monitor starting, lubricating and fuel supply systems, the combustion air intake and exhaust system, and the excitation, voltage regulation and synchronization systems.

Generator output voltage, current, kVA, power factor, Hz, etc., are also displayed in the control room. Annunciators and computer logs provide early detection of abnormal behavior.

9.5.12 Lower Drywell Flooder

9.5.12.1 Design Basis

The function of the lower drywell flooder (LDF) is to flood the lower drywell with water from the suppression pool in the unlikely event of a severe accident where the core melts and causes a subsequent vessel failure to occur.

The equipment shall meet the following performance criteria:

(1) The LDF shall provide a flow path from the suppression pool to the lower drywell when the drywell air space temperature reaches 260°C.

(2) The LDF shall pass sufficient flow from the suppression pool to the lower drywell to quench all of the postulated corium, cover the corium, and remove the corium decay heat, as confirmed by severe accident analysis (Appendix 19E).

(3) The LDF shall operate automatically in a passive manner.

(4) The LDF outlet shall be at least one meter above the lower drywell floor.

(5) The LDF inlet shall be located as far below the bottom of the first horizontal drywell-to-wetwell vent as possible while still meeting the requirements for the location of the LDF outlet.

(6) The LDF shall not become a flow path from the suppression pool to the lower drywell during design basis accidents (DBAs) such as loss-of-coolant accidents (LOCAs) or during normal plant operation

(7) The LDF shall distribute flow evenly around the circumference of the lower drywell.

9.5.12.2 System Description

The LDF is shown schematically in Figure 9.5-3.

The LDF provides a flow path for suppression pool water into the lower drywell area during a severe accident scenario that leads to core meltdown, vessel failure, and deposition of molten corium on the lower drywell floor. Molten corium is a molten mixture of fuel, reactor internals, the vessel bottom head and control rod drive components. The flow path is opened when the lower drywell airspace temperature reaches 260°C.

The flow of suppression pool water into the lower drywell through the LDF quenches the

molten corium and subsequently removes the corium decay heat. This limits the drywell temperature to 260°C and avoids degradation of non-metallic penetration seals in the upper and lower drywell. Interaction between corium and the concrete floor is also stopped. This delays the time of fission product releases for the severe accident, which allows for more decay of fission products and results in lower release fractions.

The LDF consists of ten pipes that run from the vertical pedestal vents into the lower drywell. Each pipe contains a fusible plug valve connected to the end of the pipe that extends into the lower drywell by a flange. The fusible plug valves open when the drywell air space (and subsequently the fusible plug valve) temperature reaches 260°C. When the fusible plug valves open, a minimum of 10.5 l/sec of suppression pool water will be supplied through each flooder pipe (105 l/sec total) to the lower drywell to quench the corium, cover the corium and remove corium decay heat, which is estimated at 1% of rated thermal power. The flow rate is based on a minimum hydrostatic head of 200 mm above the flooder pipe inlet centerline and takes the frictional losses through the flooder pipe and fusible plug valve into account.

The inlet centerlines of the drywell flooder pipes are located 10.2 meters below the bottom of the vessel, and the outlets of the fusible plug valves are located at least one meter above the lower drywell floor.

The fusible plug valves are made from flanges welded to the end of the vent inside the lower drywell area. The inner diameter of the pipe is slightly enlarged to accommodate a stainless steel separation disk, an insulating disk and fusible metal. The insulating disk thermally insulates the fusible metal from the wetwell water to assure that the fusible metal is not cooled by wetwell water and prevented from melting during the severe accident high lower drywell temperature conditions. The end of the fusible plug valve is covered with a plastic cover that has a low melting point. The purpose of the cover is to avoid corrosion of the fusible metal material and to assure that any toxic components from the fusible metal material that might be released do not escape into the lower drywell area during normal plant operation.

The fusible plug valve is mounted in the vertical position, with the fusible metal facing downward, to facilitate the opening of the valve when the fusible metal melting temperature is reached.

The drywell flooder pipes are welded to the stainless steel vertical vent pipes in the pedestal and to the steel liner in the lower drywell.

9.5.12.3 Safety Evaluation

The LDF is a passive injection system and is maintained in an operable state whenever the reactor is critical. The system is never expected to be needed for safety reasons because of the extensive array of water injection systems available to maintain core cooling.

The LDF is safety-related because it is a structural extension of the blowdown vent system. The LDF is Seismic Category I. The quality control classification of the LDF components is the same as the pedestal and the blowdown vents. Therefore, it meets the same structural design, materials, welding fabrication, thermal and structural analysis, and quality assurance requirements as the reactor pedestal.

The LDF has sufficient redundancy that the failure of one fusible plug to open does not degrade the ability of the system to flood the drywell and quench the corium.

The design pressure of the LDF components is 1.1 kg/cm^{2}d.

The design temperature of the LDF components is 171°C. This value is the primary containment design temperature and considers DBA events. If the LDF components lose pressure integrity at higher temperatures during a severe accident, then the LDF function (i.e., drywell flooding) is performed. Therefore, the design temperature does not need to be higher than the temperature based on DBA events.

The LDF components have zero leakage when subjected to design differential pressure of 1.1 kg/cm^2 at a design temperature of 171°C.

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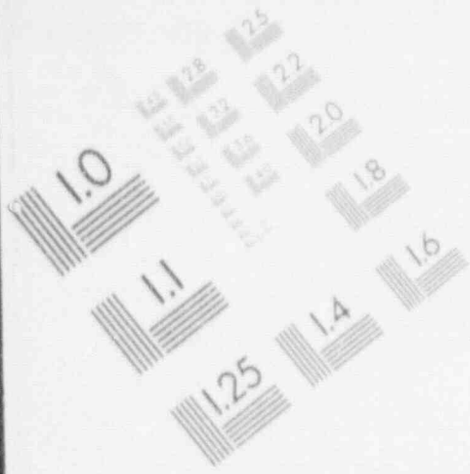

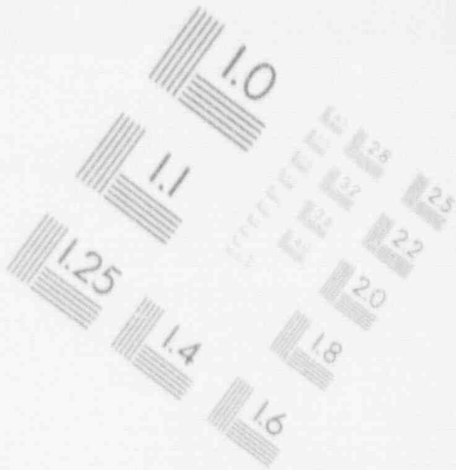

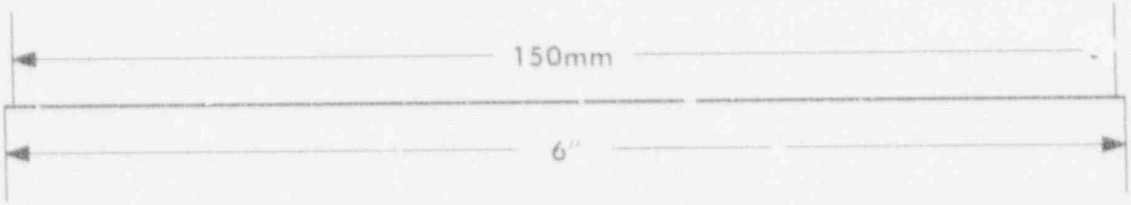

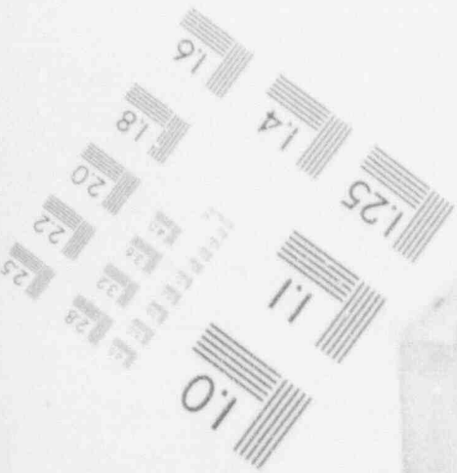

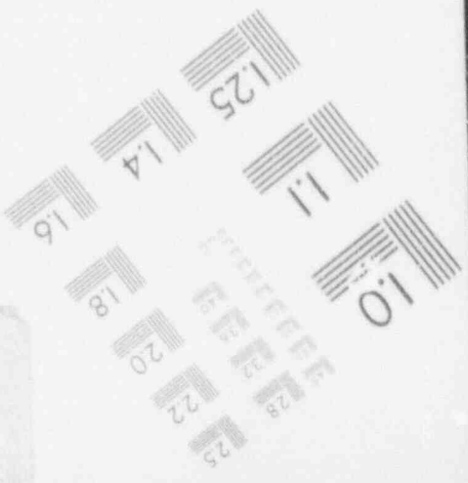

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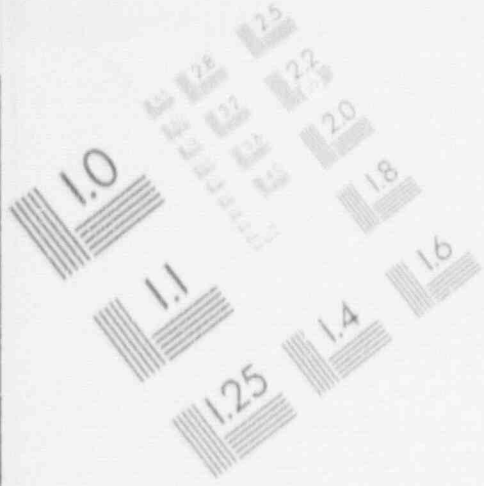

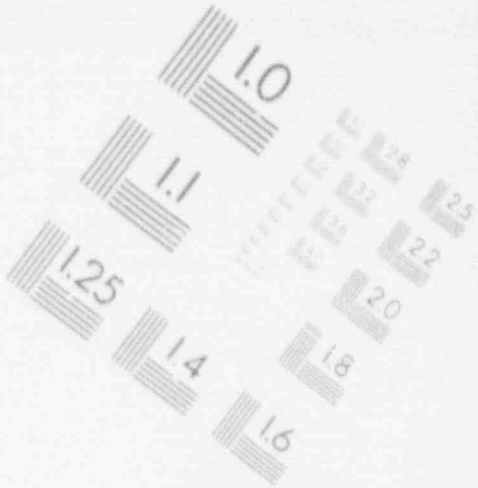

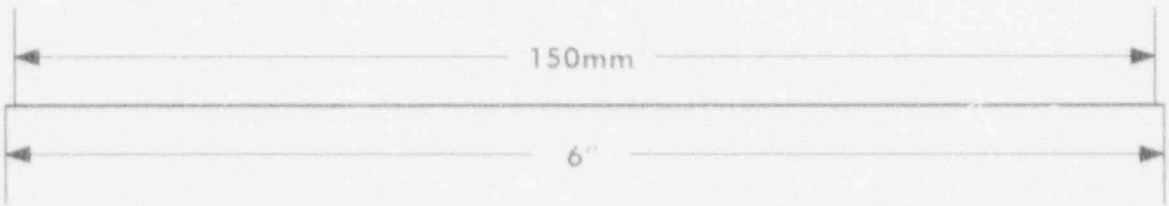

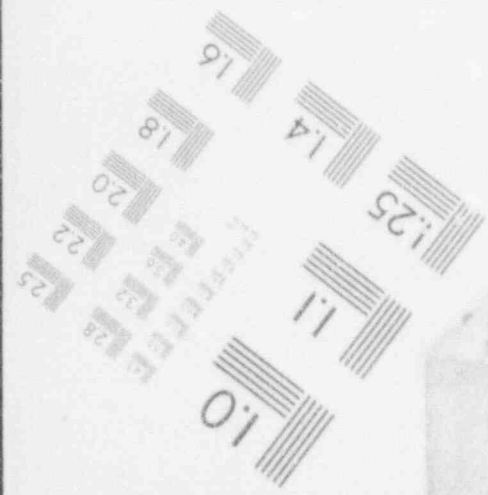

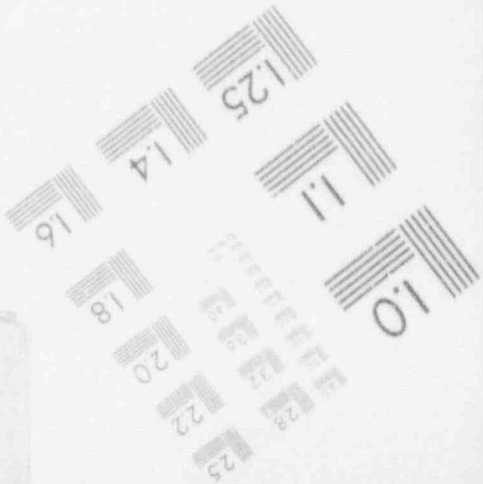

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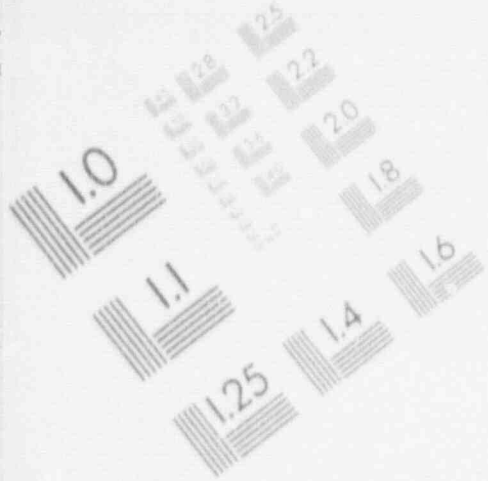

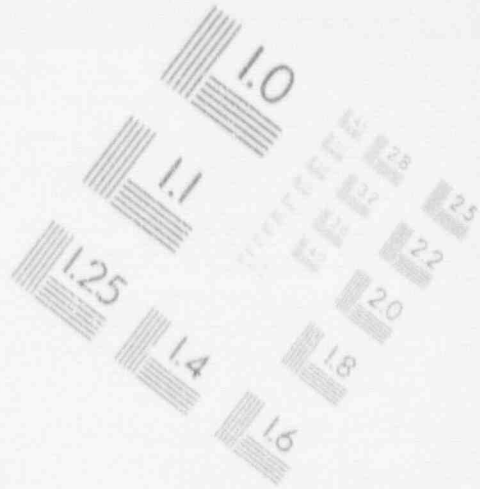

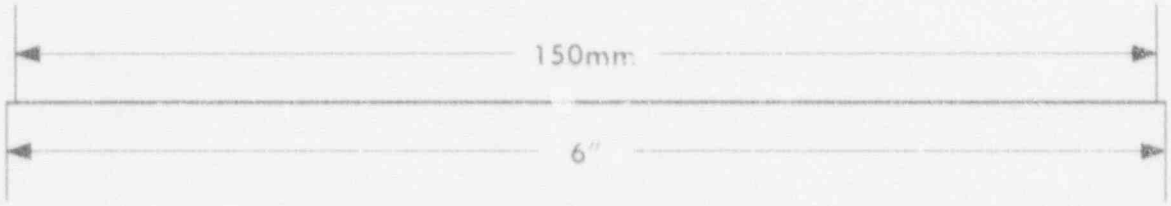

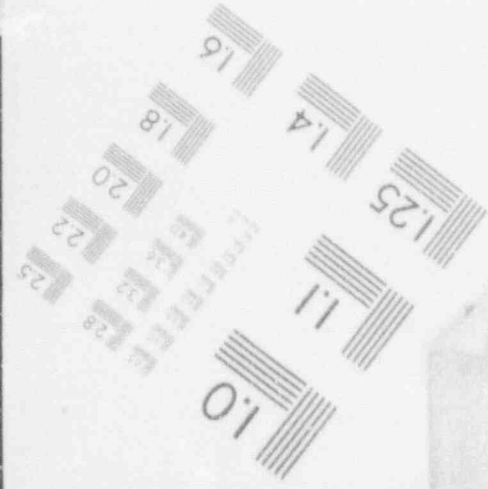

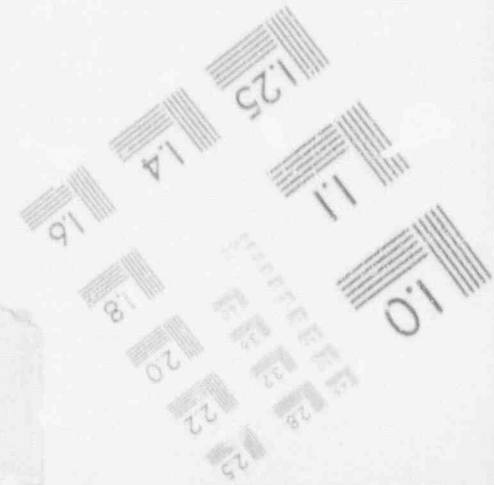

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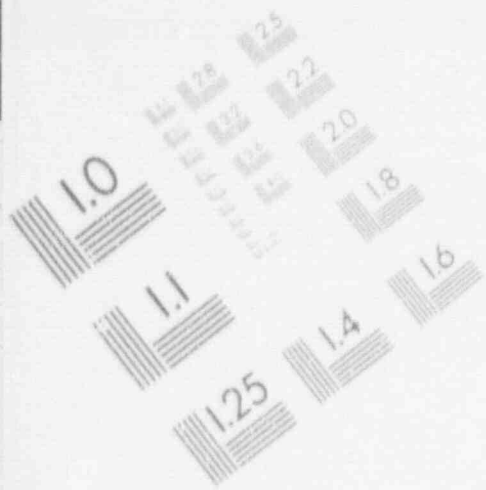

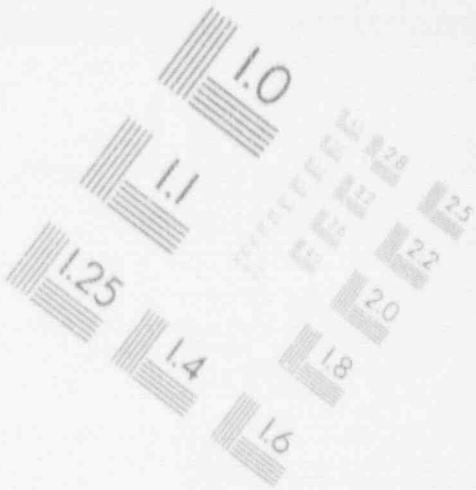

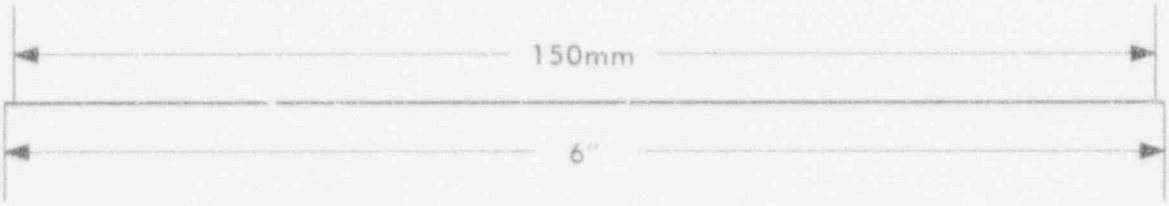

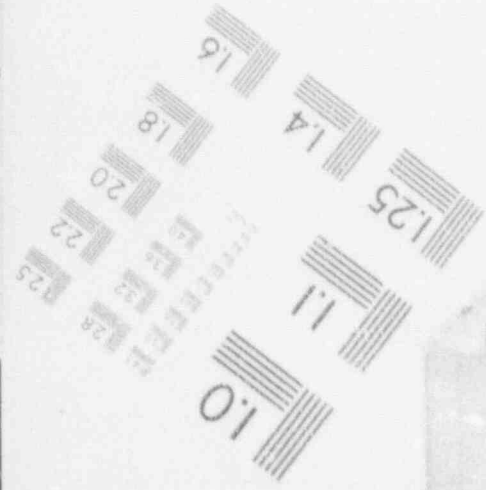

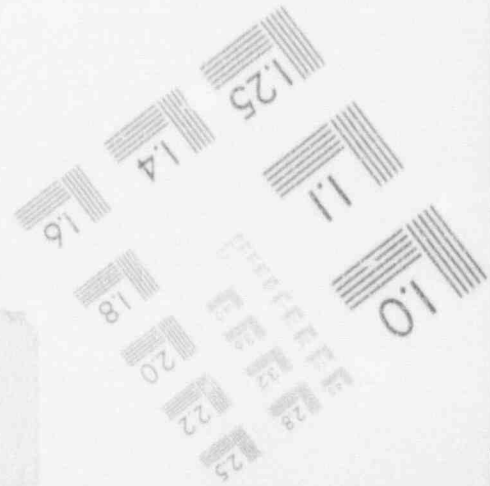

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10.4.6.1 Design Bases

10.4.6.1.1 Safety Design Bases

The CCS does not serve or support any safety function and has no safety design bases.

10.4.6.1.2 Power Generation Design Bases

Power Generation Design Basis One - The CCS continuously removes dissolved and suspended solids from the condensate to maintain reactor feedwater quality.

Power Generation Design Basis Two - The CCS removes corrosion products from the condensate and from drains returned to the condenser hotwell so as to limit any accumulation of corrosion products in the cycle.

Power Generation Design Basis Three - The CCS removes impurities entering the power cycle due to condenser circulating water leaks as required to permit continued power operation within specified water quality limits as long as such condenser leaks are too small to be readily located and repaired.

Power Generation Design Basis Four - The CCS limits the entry of dissolved solids into the feedwater system in the event of large condenser leaks, such as a tube break, to permit a reasonable amount of time for orderly plant shutdown.

Power Generation Design Basis Five - The CCS injects in the condensate such water treatment additives as oxygen and hydrogen as required to minimize corrosion/erosion product releases in the power cycle.

Power Generation Design Basis Six - The CCS maintains the condensate storage tank water quality as required for condensate makeup and miscellaneous condensate supply services.

10.4.6.2 System Description

10.4.6.2.1 General Description

The condensate cleanup system is illustrated in Figure 10.4-4. The CCS consists of six bead resin, mixed bed ion exchange polisher vessels arranged in parallel with, normally five in operation and 1 in standby. A strainer is installed downstream of each polisher vessel to preclude gross resin leakage into the power cycle in case of vessel underdrain failure, and to catch resin fine leakage as much as possible. The design bases influent concentrations are provided in Table 10.4-5. Based on the influent concentrations the condensate polisher effluent water quality is as reported in Table 10.4-6. The CCS components are located in the turbine building.

Provisions are included to permit mechanical ultrasonic washing and replacement of the ion exchange resin. Each of the polisher vessels has fail open inlet and outlet isolation valves which are remotely controlled from the local polisher control panel.

A system flow bypass valve is also provided which is manually controlled from the main control room. Pressure downstream of the polisher system is indicated and low pressure is alarmed in the main control room to alert the operator. The bypass is used only in emergency and for short periods of time until the polisher system flow is returned to normal or the plant is brought to an orderly shutdown. To prevent unpolished condensate from leaking through the bypass, double isolation valves are provided with an orificed leak-off back to the condenser.

10.4.6.2.2 Component Description

Codes and standards applicable to the CCS are listed in Subsection 3.2.2. The system is designed and constructed in accordance with quality group D requirements. Design data for major components of the CCS are listed in Table 10.4-4.

Condensate Polishers Vessels - There are six 20-percent-capacity polisher vessels (one on standby) each constructed of carbon steel and lined with natural rubber. Normal operation full load steady state design flowrate is 40 gpm per square foot of bed. Maximum flowrates are 50 and 60 gpm per square foot for steady state and transient operation respectively. The nominal bed depth is 40 inches.

10.4.6.2.3 System Operation

The CCS is continuously operated, as necessary to maintain feedwater purity levels.

Full condensate flow is passed through five of the six polisher vessels, which are piped in parallel. The sixth polisher is on standby or is in the process of

being cleaned, emptied or refilled. The service run for each polisher vessel is terminated by either high differential pressure across the vessel or high conductivity or sodium content in the polisher effluent water. Alarms for each of these parameters are provided on the local control panel.

The local control panel is equipped with the appropriate instruments and controls to allow the operators to perform the following operations:

(1) Remove an exhausted polisher from service and replace it with a standby unit

(2) Transfer the resin inventory of any polisher vessel into the resin receiver tank for mechanical cleaning or disposal.

(3) Process the as received resin through the ultrasonic resin cleaner as it is transferred from the receiver tank to the storage tank.

(4) Transfer the resin storage tank resins to any polisher vessel.

(5) Transfer exhausted resin from the receiver tank to the radwaste system.

On termination of a service run, the exhausted polisher vessel is taken out of service, and the standby unit is placed in service by remote manual operation from the local control panel. The resin from the exhausted vessel is transferred to the resin receiver tank and replaced by a clean resin bed that is transferred from the resin storage tank. A final rinse of the new bed is performed in the polisher by condensate full flow recycle to the condenser before it is placed in service. The rinse is monitored by conductivity analyzers, and the process is terminated when the required minimum rinse has been completed and normal clean bed conductivity is obtained.

Through periodic condensate makeup and reject, the condensate storage tank water inventory is processed through the CCS and tank water quality is maintained as required for condensate makeup to the cycle and miscellaneous condensate supply services. The diagram of the condensate storage and transfer system is illustrated in Figure 10.4-5.

The condensate cleanup and related support systems wastes are processed by the radwaste system as described in Chapter 11.

10.4.6.3 Evaluation

The CCS does not serve or support any safety function and has no safety design basis.

The condensate cleanup system removes some radioactive material, activated corrosion products and fission products that are carried-over from the reactor. While these radioactive sources do not affect the capacity of the resin, the concentration of such radioactive material requires shielding (see Chapter 12). Vent gases and other wastes from the condensate cleanup system are collected in controlled areas and sent to the radwaste system for treatment and/or disposal. Chapter 11 describes the activity level and removal of radioactive material from the condensate system.

The condensate cleanup system complies with Regulatory Guide 1.56, *Maintenance of Water Purity in Boiling Water Reactors*.

The condensate cleanup system and related support facilities are located in non-safety related buildings. As a result, potential equipment or piping failures can not affect plant safety.

10.4.6.4 Tests and Inspections

Preoperational tests are performed on the condensate cleanup system to ensure operability, reliability, and integrity of the system. Each polisher vessel and system support equipment can be isolated during normal plant operation to permit testing and maintenance.

10.4.6.5 Instrumentation Applications

Conductivity elements are provided for the system influent and for each polisher vessel effluent. System influent conductivity detects condenser leakage; whereas, polisher effluent conductivities provide indication of resin exhaustion. The polisher effluent conductivity elements also monitor the quality of the condensate that is recycled to the condenser after processing through a standby vessel before it is returned to service. Differential pressure is monitored across each polisher vessel and each vessel discharge resin strainer to detect blockage of flow. The flow through each polisher is monitored and used as control input to assure even distribution of condensate flow through all operating vessels and by correlation with the vessel pressure drop, to

permit evaluation of the vessel throughput capacity. Individual vessel effluent conductivity, differential pressure, and flow measurements are recorded at the system local control panel. A multipoint annunciator is included in the local panel to alarm abnormal conditions within the system. The local panel is connected to the main control room where local alarms annunicated by a global system alarm but can also be displayed individually if requested by the operators.

Other system instrumentation includes turbidity and other water quality measurements as necessary for proper operation of the polisher and miscellaneous support services, and timers for automatic supervision of the resin transfer and cleaning cycles. The control system prevents the initiation of any operation or sequence of operations which would conflict with any operation or sequence already in progress whether such operation is under automatic or manual control.

10.4.7 Condensate and Feedwater System

The function of the condensate and feedwater system (CFS) is to receive condensate from the condenser hotwells, supply condensate to the cleanup system, and deliver high purity feedwater to the reactor, at the required flow rate, pressure and temperature.

10.4.7.1 Design Bases

10.4.7.1.1 Safety Design Bases

The condensate-feedwater system does not serve or support any safety function and has no safety design bases.

10.4.7.1.2 Power Generation Design Bases

Power Generation Design Basis One - The CFS is designed to provide a continuous and dependable feedwater supply to the reactor at the required flow rate, pressure, and temperature under all anticipated steady-state and transient conditions.

Power Generation Design Basis Two - The CFS is designed to supply up to 115% of the rated feedwater flow demand during steady state power operation and for at least 10 seconds after generator step load reduction or turbine trip, and up to 75% of the rated flow demand thereafter.

Power Generation Design Basis Three - The CFS is designed to permit continuous long term full power plant operation with the following equipment out of service: one feedwater pump, one condensate pump or one heater drain pump or, one high pressure heater string with a slightly reduced final feedwater temperature.

Power Generation Design Basis Four - The CFS is designed to permit continuous long term operation with one LP heater string out of service at the maximum load permitted by the turbine manufacturer, approximately 85%, value which is set by steam flow limitation on the affected LP turbine. 430/98

Power Generation Design Basis Five - The CFS is designed to heat up the reactor feedwater to a nominal temperature of 420F during full load operation and to lower temperatures during part load operation.

Power Generation Design Basis Six - The CFS is designed to minimize the ingress or release of impurities to the reactor feedwater.

10.4.7.2 Description

10.4.7.2.1 General Description

The condensate and feedwater system is illustrated in Figure 10.4-6 and 10.4-7. The condensate and feedwater system consists of the piping, valves, pumps, heat exchangers, controls and instrumentation, and the associated equipment and subsystems which supply the reactor with heated feedwater in a closed steam cycle utilizing regenerative feedwater heating. The system described in this subsection extends from the main condenser outlet to the second isolation valve outside of containment. The remainder of the system, extending from the second isolation valve to the reactor, is described in Chapter 5. Turbine extraction steam is utilized for a total of six stages of closed feedwater heating. The drains from each stage of the low pressure feedwater heaters are cascaded through successively lower pressure feedwater heaters to the main condenser. The high pressure heater drains are pumped backward to the reactor feedwater pumps suction. The cycle extraction steam, drains and vents systems are illustrated in Figures 10.4-8 and 10.4-9.

The CFS consists of four 33% capacity condensate pumps, three manually operated 33-50% capacity reactor feedwater pumps (three normally operating and one on standby), four stages of low-pressure feedwater heaters, and two stages 430/98

of high-pressure feedwater heaters, piping, valves, and instrumentation. The condensate pumps take suction from the condenser hotwell and discharge the deaerated condensate into one common header which feeds the condensate filters and demineralizers. Downstream of the condensate demineralizers, the condensate is taken by a single header and flows in parallel through five auxiliary condenser/coolers, (one gland steam exhauster condenser and two sets of steam jet air ejector condensers and offgas recombiner condenser (coolers). The condensate then branches into three parallel strings of low pressure feedwater heaters. Each string contains four stages of low-pressure feedwater heaters. The strings join together at a common header which is routed to the suction of the reactor feedwater pumps.

Another input to the feedwater flow consists of the drains which are pumped backward and injected into the feedwater stream at a point between the fourth stage low-pressure feedwater heaters and the suction side of the reactor feed pumps. These drains, which originate from the crossaround steam moisture separators and reheaters and from the two sets of high-pressure feedwater heaters, are directed to the heater drain tank. The reheater and top heater drains are deaerated in the crossaround heaters so that, after mixing with condensate, the drains are compatible with the reactor feedwater quality requirements for oxygen content during normal power operation. The heater drain pumps take suction from their heater drain tank and inject the deaerated drains into the feedwater stream on the suction side of the reactor feed pumps.

The reactor feedwater pumps discharge the feedwater into two parallel high pressure feedwater heater strings, each with two stages of high-pressure feedwater heaters. Downstream of the high-pressure feedwater heaters, the two strings are then joined into a common header, which divides into two feedwater lines which connect to the reactor.

A bypass is provided around the reactor feedwater pumps to permit supplying feedwater to the reactor during early startup without operating the feedwater pumps, using only the condensate pump head.

Another bypass, equiped with a feedwater flow control valve, is provided around the high pressure heaters to perform two independent functions. During startup, the bypass and its flow control valve are used to regulate the flow of feedwater supplied by either the condensate pumps or the reactor feed pumps operating at their minimum fixed speed. During power operation, the heater bypass function is to maintain full feedwater flow capability when a high pressure heater string must be isolated for maintenance.

During power operation, the condensate is well deaerated in the condenser and continuous oxygen injection is used to maintain the level of oxygen content in the final feedwater as shown in Subsection 10.4.6.

To minimize corrosion product input to the reactor during startup, recirculation lines to the condenser are provided from the reactor feedwater pump suction header and from the high-pressure feedwater heater outlet header.

Prior to plant startup, cleanup is accomplished by allowing the system to recirculate through the condensate polishers for treatment prior to feeding any water to the reactor during startup.

10.4.7.2.2 Component Description

All components of the condensate and feedwater system that contain the system pressure are designed and constructed in accordance with applicable codes as referenced in Section 3.2.

Condensate Pumps - The four condensate pumps are identical, fixed speed motor driven pumps, three are normally operated, and the fourth is on standby. Valving is provided to allow individual pumps to be removed from service.

A minimum flow recirculation line is provided

downstream of the auxiliary condensers for condensate pump protection and for auxiliary condenser minimum flow requirements.

Low-pressure Feedwater Heaters - Three parallel and independent strings of four closed feedwater heaters are provided, and one string is installed in each condenser neck. The heaters have integral drain coolers, and their drains are cascaded to the next lower stage heaters of the same string except for the lowest pressure heaters which drain to the main condensers. The heater shells are either carbon steel or low alloy ferritic steel, and the tubes are stainless steel. Each low pressure feedwater heater string has an upstream and downstream isolation valve which closes on detection of high level in any one of the low pressure heaters in the string.

High-pressure Feedwater Heaters - Two parallel and independent strings of two high-pressure feedwater heaters are located in the high pressure end of the turbine building. The No. 6 heaters, which have integral drain coolers, are drained to the No. 5 heaters. The No. 5 heaters, which are condensing only, drain to their respective heater drain tanks. The heater shells are carbon steel, and the tubes are stainless steel. 4/30/90

Heater string isolation and by pass valves are provided to allow each string of high-pressure heaters to be removed from service, thus, slightly reducing final feedwater temperature but requireing no reduction in plant output. The heater string isolation and bypass valves are actuated on detection of high level in either of the two high pressure heaters in the string. 4/30/90

The startup and operating vents from the steam side of the feedwater heaters are piped to the main condenser except for the highest pressure heater operating vents which discharge to the cold reheat lines. Discharges from shell relief valves on the steam side of the feedwater heaters are piped to the main condenser.

Heater Drain Tank - Two heater drain tanks are provided. Drain tank level is maintained by the heater drain pump and control valves in drain pump disharge and recirculation line. 4/30/90

The heater drain tank is provided with an alternate drain line to the main condenser for automatic dumping upon detection of high level. The alternate drain line is also used during startup and shutdown when it is desirable to dump the drains for feedwater quality purposes.

The drain tanks and tank drain lines are designed to maintain the drain pumps available suction head in excess of the pump required minimum under all anticipated operating conditions including, particularly, load reduction transients. This is achieved mainly by providing a large elevation difference between tanks and pumps (approximately 50 feet) and optimizing the drain lines which would affect the drain system transient response, particularly, the drain pump suction line.

Heater Drain Pumps - Two motor-driven heater drain pumps operate in parallel, each taking suction from a heater drain tank and discharging into the suction side of the reactor feedwater pumps. The drain system design allows each heater drain pump to be individually removed from service for maintenance while the the balance of the system remains in operation while the affected string drains dump to the condenser.

Controlled drain recirculation is provided from the discharge side of each heater drain pump to the associated heater drain tank. This ensures that the minimum safe flow through each heater drain pump is maintained during operation. 4/30/90

Reactor Feedwater Pumps - Three identical and independent, 33-60% capacity reactor feed pumps (RFPs) are provided. The three pumps manually operate in parallel and discharge to the high-pressure feedwater heaters. The pumps take suction downstream of the last stage low-pressure feedwater heaters and discharge through the high-pressure feedwater heaters. Each pump is driven by an adjustable speed synchronous motor.

Isolation valves are provided which allow each reactor feed pump to be individually removed from service for maintenance, while the plant continues operation at full power on the two remaining pumps.

Controlled feedwater recirculation is provided from the discharge side of each reactor feed pump to the main condenser. This provision ensures that the minimum safe flow through each reactor feed pump is maintained during operation.

10.4.7.2.3 System Operation

NORMAL OPERATION - Under normal operating conditions, system operation is automatic. Automatic level control systems control the levels in all feedwater heaters, the heater drain tanks, and the condenser hotwells. Feedwater heater levels are controlled by modulating drain valves. Control valves in the discharge and recirculation lines of the heater drain pumps control the level in the heater drain tanks. Valves in the makeup line to the condenser from the condensate storage tank and in the return line to the condensate storage tank control the level in the condenser hotwells.

During power operation feedwater flow is automatically controlled by the reactor feedwater pump speed that is set by the feedpump speed control system. The control system utilizes measurements of steam flow, feedwater flow, and reactor level to regulate the feedwater pump speed. During startup, feedwater flow is automatically regulated by the high pressure heater bypass flow control valve.

Ten-percent step load and 5-percent per minute ramp changes can be accommodated without major effect in the CFS. The system is capable of accepting a full generator load rejection without reducing feedwater flow rate.

10.4.7.3 Evaluation

The condensate and feedwater system does not serve or support any safety function. Systems analysis show that failure of this system cannot compromise any safety-related systems or prevent safe shutdown.

During operation, radioactive steam and condensate are present in the feedwater heating portion of the system, which includes the extraction steam piping, feedwater heater shells, heater drain piping, and heater vent piping. Shielding and access control are provided as necessary (see Chapter 12). The condensate and feedwater system is designed to minimize leakage with welded construction utilized where practicable. Relief discharges and operating vents are channeled through closed systems.

If it is necessary to remove a process nent from service such as a feedwater heater, pump, or control valve, continued operation of the system is possible by use of the multistring arrangement and the provisions for isolating and bypassing equipment and sections of the system.

The majority of the condensate and feedwater piping considered in this section is located within the non-safety related turbine building. The portion which connects to the second valve outside the containment is located in the steam tunnel between the turbine and reactor buildings. This portion of the piping is analyzed for dynamic effect from postulated seismic events and safety-relief valve discharges. The entire condensate and feedwater system piping is analyzed for water hammer loads that could potentially result from anticipated flow transients.

10.4.7.4 Tests and Inspections

10.4.7.4.1 Preservice Testing

Each feedwater heater and condensate pump receives a shop hydrostatic test which is performed in accordance with applicable codes. All tube joints of feedwater heaters are shop leak tested. Prior to initial operation, the completed condensate and feedwater system receives a field hydrostatic and performance test and inspection in accordance with the applicable code. Periodic tests and inspections of the system are performed in conjunction with scheduled maintenance outages.

10.4.7.4.2 Inservice Inspections

The performance status, leaktightness, and structural leaktight integrity of all system components are demonstrated by continuous operation.

10.4.7.5 Instrumentation Applications

Feedwater flow-control instrumentation measures the feedwater discharge flow rate from each reactor feed pump and the heater bypass startup flow control valve. These feedwater system flow measurements are used by the feedwater control system to regulate the feedwater flow to the reactor to meet system demands. The feedwater control system is described in Subsection 7.7.1.4

Pump flow is measured on the pump inlet line and flow controls provide automatic pump recirculation flow for each reactor feedwater pump. Automatic controls also regulate the condensate flow

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ILLUSTRATIONS

12.2 RADIATION SOURCES

12.2.1 Contained Sources

12.2.1.1 Source Terms

With the exception of the vessel and drywell shields, shielding designs are based on fission product and activation product sources consistent with Section 11.1. For shielding, it is conservative to design for fission product sources at peak values rather than an annual average, even though experience supports a lower annual average than the design average (Reference 1). It should be noted that activation products, principally Nitrogen-16, control shielding calculations in most of the primary system. In areas where fission products are significant, conservative allowance is made for transient decay while at the same time providing for transient increase of the noble gas source, daughter product formation and energy level of emission. Areas where fission products are significant relative to Nitrogen-16 include: (1) the condenser off-gas system downstream of the steam jet air ejector; (2) liquid and solid radwaste equipment; (3) portions of the RWCS; and (4) portions of the feedwater system downstream of the hotwell including condensate treatment equipment.

For application, the design sources are grouped first by location and then by equipment type (e.g., reactor building, core sources). The following paragraphs represent the source data in various pieces of equipment throughout the plant. General locations of equipment are shown in the general plant arrangement drawings of the Section 1.2. Specific Acceptance Criterion II.6 of Section 12.2 provides that in addition to the location of contained sources, their approximate size and shape be shown. Though this has not always been included, the source strength or concentration has been provided in Chapter 12 tables and detailed geometry has been provided in Table 12.2-1 for the reactor and in Chapter 5 for the main steam. In Chapter 12 the reactor water concentrations were used to develop sources in equipment containing reactor water or steam.

12.2.1.2 Reactor, Radwaste, and Turbine Building Sources

The information in this section is divided into two categories, the reactor vessel sources (Subsection 12.2.1.2.1) and the sources from the remaining areas (Subsections 12.2.1.2.2 through 12.2.1.2.9). Included in these areas are the sources from the radwaste building (Subsection 12.2.1.2.6) and the turbine building (Subsection 12.2.2.3). Table 12.2-5 presents an overview of the radioactive sources found in the ABWR excluding the reactor pressure vessel. This table is divided into four sections. The first section lists all major radioactive sources, the table which provides the source term information for the component, and the figure in Section 12.3 (or Chapter 1) in which the component location is shown along with corrdinates for the component. In addition, the approximate geometry of the component is supplied. This geometry in most cases is only approximate and represents a generic application as compared to specific details for a vendor supplied component. The second section of Table 12.2-5 gives for each component the estimated source distribution in each component. Again this is estimated and will depend on final design parameters with vendor specific application. The third section of Table 12.2-5 lists room dimensions and wall thickness for each component. This data is taken from the arrangement drawings and represents minimal values. Part four of Table 12.2-5 lists pipe chases, the major pipe routing through these chases, and piping data. Only chases carrying significant radioactive sources are listed.

Some areas of the plant show shielded areas without any designation to any radioactive component. These are primarily areas found around the primary containment boundary. For example, in Figure 12.3-5 at coordinate (RF,R4) a shielded area is shown with break down walls without any designated component. This area represents shielded penetration areas for non-radioactive components and can be cross referenced to Figure 1.2-13. Reference to Figure 1.2-13a shows electrical penetrations from the primary containment into the shielded area at (RF,R4) on Figure 12.3-5.

12.2.1.2.1 Reactor Vessel Source

12.2.1.2.1.1 Radiation from the Reactor Core

12.2.1.2.1.1.1 General

The information in this section defines a re-

actor vessel model and the associated gamma and neutron radiation sources. This section is designed to provide the data required or calculations beyond the vessel. The data selected were not chosen for any given program, but were chosen to provide information for any of several shield program types. In addition to the source data, calculated radiation dose levels are provided at locations surrounding the vessel. These data are given as a potential check point for calculations by shield designers.

12.2.1.2.1.1.2 Physical Data

Table 12.2-1 presents the physical data required to form the model in Figure 12.2-1. This model was selected to contain as few separate regions as possible to adequately portray the reactor. Table 12.2-1 provides nominal dimensions and material volume fractions for each boundary and region in the reactor model. To describe the reactor core, Table 12.2-1 provides thermal power, power density, core dimensions, core average material volume factions and reactor power distributions. The reactor power distributions are given for both radial and axial distributions. These data contain

uncertainties in the volume regions near the edge of the core. The level of uncertainties for these regions is estimated at 20%.

12.2.1.2.1.1.3 Core Boundary Neutron Fluxes

Table 12.2-2 presents peak axial neutron multigroup fluxes at the core equivalent radius. The core-equivalent radius is a hypothetical boundary enclosing an area equal to the area of the fuel bundles and the coolant space between them. The peak axial flux occurs adjacent to the portion of the core with the greatest power. While the flux within any given energy group is not known within a factor of 2, the total calculated core boundary flux is estimated to be within %50%.

12.2.1.2.1.1.4 Gamma Ray Source Energy Spectra

Table 12.2-3 presents average gamma ray energy spectra per thermal per watt of reactor power in both core and noncore regions. In Table 12.2-3, part A, the energy spectra in the core are presented. The energy spectra in the core represent the average gamma ray energy released by energy group per watt of core thermal power. The energy spectra in MeV per sec per cm^3 per watt can be used with the total core power and power distributions to obtain the source in any part of the core.

The gamma ray energy spectra include the fission gamma rays, the fission product gamma ray and the gamma rays resulting from inelastic neutron scattering and thermal neutron capture. The total gamma ray energy released in the core is estimated to be accurate to within %10%. The energy release rate above 6 MeV may be in error by as much as a factor of %2.

Table 12.2-3, part B, gives a gamma ray energy spectrum in MeV/sec/W in spent fuel as a function of time after operation. The data were prepared from tables of fission product decay gamma fitted to integral measurements for operation times of 10^8 sec, or approximately 3.2 years. To obtain shutdown sources in the core the gamma ray energy spectra are combined with the core thermal power and power distributions. Shutdown sources in a single fuel element can be obtained by using the gamma ray energy spectra and the thermal power the element contained during operation.

Table 12.2-3, part C, gives the gamma ray energy spectra in the cylindrical regions of the reactor from the core through the vessel. The energy spectra are given in terms of MeV/cm^3/sec/W at the inside surface and outside surfaces of the region. This energy spectrum, multiplied by the core thermal power, is the gamma ray source. The point on the inside surface of the region is the maximum point within the region. In the radial direction, the variation in source intensity may be approximated by an exponential fit to the data on the inside and outside surfaces of the region. The axial variation in a region can be estimated by using the core axial variation. The uncertainty in the gamma ray energy spectra is due primarily to the uncertainty in the neutron flux in these regions. The uncertainty in the neutron flux is estimated to vary from approximately %50% at the core boundary to a factor of %3 at the outside of the vessel. The calculations were carried out with voids beyond the vessel.

12.2.1.2.1.1.5 Gamma Ray and Neutron Fluxes Outside the Vessel

Table 12.2-4 presents the maximum axial neutron and gamma ray fluxes outside the vessel. The maximum axial flux occurs on the vessel opposite the elevation of the core with the maximum outer bundle power level. This elevation can be located using the data from Table 12.2-1. The fluxes at this elevation are based on a mean radius core and do not show azimuths angle variations. The calculational model for these fluxes assumed no shield materials beyond the vessel wall. The presence of shield materials will significantly alter the neutron fluxes in the lower end of the neutron energy spectrum. The gamma ray calculations include gamma ray sources from all of the cylindrical regions between the center of the core and the edge of the vessel. While the uncertainties in a given energy group flux may be a factor of %3, the uncertainties in the total integral flux are estimated to be within a factor of two.

12.2.1.2.1.1.6 Deleted

carried out for a mean element and appropriate decay time.

12.2.1.2.9 Other Radioactive Sources

12.2.1.2.9.1 Reactor Startup Source

The reactor startup source is shipped to the site in a special cask designed with shielding. The source is transferred under water while in the cask and loaded into beryllium containers. This is then loaded into the reactor while remaining under water. The source remains within the reactor for it lifetime. Thus, no unique shielding requirements are required after reactor operation.

12.2.1.2.9.2 Radioactive Sources in the Control Rod Drive System

The control rod drive (CRD) source term data are provided in Table 12.2-18. The system is described in Subsection 3.9.4.

12.2.1.2.9.3 Radioactivity in the Transverse In-core Probe

The transverse in-core probe (TIP) system consists of a probe and a stainless steel cable which is run into and out of the core such that the probe and up to 12 feet of cable are activated. The probe is described in Subsection 7.7.1.6.1 and is automatically controlled and indexed to its in-core position. For maintenance, the probe is manually withdrawn into a shielded assembly area in which a shielded container is used to hold the probe. Both automatic logic control and mechanical stops prevent the probe and activated sections of the cable from withdrawl beyond the shielded room and container. Table 12.2-24 describes the levels of radioactivity expected from the probe and cable. Since there are two specific types of probes, a neutron and a gamma, both types are described in Table 12.2-24.

12.2.1.2.9.4 Radioactivity in the Reactor Internal Pumps

The reactor internal pumps, RIP, are located on the lower exterior portion of the pressure vessel and connect to an impeller located in the pressure vessel. A constant flow of clean water is maintained from the pump into the pressure vessel to minimize contamination of the lower pump housing and components. A complete description of the internal pump is given in Subsection 5.4.1. Contamination of the pump nevertheless occurs primarily on the upper impeller and components into the lower pump housing. Table 12.2-25 presents the expected levels of contamination based upon operating experience.

12.2.1.2.9.5 Radioactivity in the Standby Gas Treatment System

The standby gas treatment system (SGTS) is described in Subsection 6.5. For the determination of the potential activity associated with the operation of the SGTS, the primary containment source term developed in Subsection 12.2.2.1 for Table 12.2-19 was used as the basis for input to the SGTS. Six purges per year were assumed with a SGTS replacement life time of five years. The inventory is given in Table 12.2-30.

12.2.1.2.10 Post Accident Radioactive Sources

The ABWR general design criteria limits potential radiation exposure from accidents both to plant personnel and to the public by the use of containment and treatment of accident sources. The following describes those features of the ABWR germane to post accident radiation sources in the primary containment, reactor building, radwaste building, and the turbine building.

The primary containment is an inerted steel lined pressure boundary capable of containing all accidents sources with minimal leakage to the environment or other plant areas. Sufficient redundancy in the ECCS and spray systems exist to insure within a reasonable probability that this primary boundary will not exceed design criteria. In the case of a degraded core event additional passive features such as the suppression pool and passive flooder system have been incorporated to flood the containment and scrub airborne fission products. Therefore, for all but the most improbable accident scenarios, radioactive sources from the pressure vessel will be contained in the primary containment.

With respect to the reactor building, the overall plant design has divided the reactor building into three separate and independent divisions. ECCS components are contained in each division in separate isolated rooms such that the failure of one system in one division will not affect in any way components in another division. Releases of radioactive material either in the form of water or steam (airborne) are contained in and isolated to a large extent in the compartment in which it might occur by the use of water tight doors and area radiation monitors which isolate the HVAC system from the compartment. Divisional separation under such conditions is complete. Sumps are designed to detect and alarm in the event of leaks in excess of one gallon per minute establishing a threshold for leak before break on the larger water carrying piping systems. All connections to the primary containment not terminating in the reactor building meet GDC54, 55, 56, and 57. Therefore, in the event of an accident involving radioactive sources in the primary containment or reactor building such sources would be contained and isolated for further treatment and decontamination.

Likewise potential releases in the radwaste building will be contained by isolating the radwaste building atmosphere and sealing any water releases in the building which is seismically qualified and steel lined to prevent any potential water releases. Such potential releases are discussed in Section 15.7.

The turbine building contains no major sources of releasable radioactivity (discounting N-16 because of the 7.7 second half-life) and potential releases are limited to liquid releases of low activity water from the feedwater and condenser system. Two other sources exist which contain radioactivity species but in form not amenable for release. The potential for accident sources from these two sources, the offgas system and condenser demineralizers, is reduced due to heavy shielding and compartmentalizing these components.

Estimates on sources and location for limiting design basis events are found in Chapter 15 and sources for degraded core events as a function of probability are found in Chapter 19.

12.2.1.3 Turbine Building Sources

Turbine building sources are primarily dominated by N-16 in the steam flow from the pressure vessel. The N-16 source results in significant gamma shine from the main steam lines and steam bearing components (turbines, moisture separators, and reheaters) on the order of 20-50 rad/hr contact. Estimates of typical BWR sources and gamma shine are given in Reference 11. Since the geometry of the radiation source is dependent on the exact turbine configuration used, the specific details for the turbines and turbine reheaters are interface requirements for referencing applicant as called out in Subsection 12.2.4. Tables 12.2-26 through 12.2-28 provide estimates of inventories for the moisture separator, condenser, and condenser demineralizer. The offgas system is divided into three major components, steam jet air ejector (SJAE), recombiner, and charcoal tanks. The inventory in the SJAE is given in Table 12.2-29 while the inventories in the recombiner and charcoal tanks are given in Table 12.2-14. The offgas system is more fully described in Subsection 12.2.1.2.6.3.

12.2.2 Airborne and Liquid Sources for Environmental Consideration

This Subsection deals with the sources and parameters required to evaluate airborne and liquid releases during normal plant operations for compliance with 10CFR20 criteria

The following page is 12.2-6.1

TABLE 12.2-1

BASIC REACTOR DATA

A.	Reactor Thermal Power	3926 MW
B.	Average Power Density	50.57 W/cm^3
C.	Physical Dimensions	Fig. 12.2-1
	Radii	cm
1.	Core Equivalent Radius	258.13
2.	Inside Shroud Radius	274.955
3.	Outside Shroud Radius	286.035
4.	Inside Vessel Radius - Average	355.6
5.	Outside Vessel Radius - Average	374.015
6.	Shroud Head Inside Radius	568.96
7.	Outside Top Guide Radius	307.34
8.	Inside Radius of Shroud Head Flange	292.1
9.	Outside Radius of Shroud Head Flange	297.18
10.	Vessel Top Head Inside Radius	335.28
11.	Vessel Bottom Head Inside Radius	486.61
	Elevation	cm
12.	Outside of Vessel Bottom Head	-27.94
13.	Inside of Vessel Bottom Head	0.0
14.	Vessel Bottom Head Knuckle	164.46
15.	Bottom of Core Support Plate	506.34
16.	Top of Core Support Plate	511.42
17.	Bottom of Active Fuel	534.11
18.	Top of Active Fuel (144 inch)	904.95
	(150 inch)	915.11
19.	Bottom of Top Guide	933.85
20.	Top of Fuel Channel	951.63
21.	Shroud Head Knuckle	1068.29
22.	Inside of Shroud Head	1150.54
23.	Outside of Shroud Head	1155.62
24.	Normal Vessel Water Level	1342.06
25.	Top of Steam Dryer	1747.14
26.	Vessel Top Head Knuckle	1770.3
27.	Inside of Vessel Top Head	2105.58
28.	Outside of Vessel Top Head	2117.01

TABLE 12.2-1

BASIC REACTOR DATA
(Continued)

D. Material Densities* (gm/cm^3)

REGION	COOLANT	UO_2	ZIRCALOY	304L STAINLESS
A	0.740	0	0	0.178
B	0.338	0	0	4.35
C	0.318	2.33	0.978	0.056
C-1	0.597	0	0.166	1.70
C-2	0.234	0	1.10	0.255
D	0.240	0	1.00	1.21
E	0.390	0	0	0
F	0.669	0	0	0.200
G	0.036	0	0	0
H	0.740	0	0	0
I	0.740	0	0	0.260

See Figure 12.2-1 for Location Schematic.

Table 12.2-5

Radiation Sources

A. Radiation Sources

Source Table	For	Drawing	Location	Approximate Geometry
12.2-6	RHR Heat Exchanger	12.3-1	(R1,RF) (R6,RA) (R6,RF)	Rt Cylndr (r=0.9m,l=7m)
12.2-8	RCIC Turbine	12.3-1	(R6,RC)	Rt Cylndr (r=0.5m,l=0.7m)
12.2-9	CUW Filter Demineralizer	12.3-3	(R2,RB)	2 Tanks, Rt Cylndr (r=0.6m,l=3.3m)
12.2-10	RWCU Regen Heat Exchanger	12.3-2	(R1,RC)	Rt Cylndr (r=0.4m,l=6.8m)
12.2-11	RWCU Non-Regen Heat Exchanger	12.3-1	(R1,RC)	Rt Cylndr (r=0.4m,l=5.5m)
12.2-13.1	LCW Collector Tank	12.3-37	ITEM 7	2 Tanks, Rt Cylndr (r=4.m,l=9.4m)
12.2-13.2	LCW Filter	12.3-39	ITEM 12	Rt Cylndr (r=0.5m,l=2.5m)
12.2-13.3	LCW Demineralizer	12.3-39	ITEM 11	Rt Cylndr (r=0.6m,l=2.8m)
12.2-13.4	LCW Sample Tank	12.3-38	ITEM 8	2 Tanks, Rt Cylndr (r=4.m,l=9.4m)
12.2-13.5	HCW Collector Tank	12.3-37	ITEM 13	Rt Cylndr (r=2.2m,l=4.3m)
12.2-13.6	HCW Demineralizer	12.3-39	ITEM 20	Rt Cylndr (r=0.6m,l=2.8m)
12.2-14	Offgas	12.3-50	(TF,T2)	Tank 1, Rt Cylndr (r=0.6m,l=7.6m) Tanks 2-9, Rt Cylndr (r=1.1m,l=7.6m)
12.2-29	Steam Jet Air Ejector	12.3-51	(TF,T2)	Rt Cylndr (r=0.15m,l=4.6m) Rt Cylndr (r=0.76m,l=6.1m) Rt Cylndr (r=0.2m,l=4.6m)
12.2-14	Offgas Recombiner	12.3-51	(TF,T2)	Rt Cylndr (r=1.4m,l=7m)
12.2-15.1	CUW Backwash Receiving Tank	12.3-1	(R2,RB)	Rt Cylndr (r=2.2m,l=5.7m)
12.2-15.2	CF Backwash Receiving Tank	12.3-49	(TD,T4)	Rt Cylndr (r=2.2m,l=5.7m)
12.2-15.3	Phase Separator	12.3-38	ITEM 30	2 Tanks, Rt Cylndr (r=2.4m,l=6.0m)
12.2-15.4	Spent Resin Storage Tank	12.3-38	ITEM 31	Rt Cylndr (r=2.0m,l=5.7m)
12.2-15.5	Concentrated Waste Tank	12.3-37	ITEM 35	Rt Cylndr (r=1.5m,l=4.4m)
12.2-15.6	Sol Dryer Feed Tank	12.3-41	ITEM 39	Rt Cylndr (r=1.6m,l=3.2m)
12.2-15.7	Sol Dryer (outlet)	12.3-39	ITEM 55	Rt Cylndr (r=0.2m,l=3.2m)
12.2-15.8	Sol Pelletizer	12.3-38	ITEM 58	Rt Cylndr (r=0.4m,l=2.5m)
12.2-15.9	Sol Mist Separator (steam)	12.3-39	ITEM 56	Rt Cylndr (r=0.1m,l=2.8m)
12.2-15.10	Sol Condenser	12.3-40	ITEM 57	Rt Cylndr (r=0.2m,l=1.4m)
12.2-15.11	Sol Drum	12.3-39	(2,D)	Rt Cylndr (r=0.3m,l=0.8m) Box (1.5mx1.5mx1m)
12.2-16	FPC Filter Demineralizer	12.3-3	(R2,RB)	Rt Cylndr (r=0.7m,l=3.4m)

Table 12.2-5

Radiation Sources (Continued)

A. Radiation Sources (Continued)

Source Table	For	Drawing	Location	Approximate Geometry
12.2-17	Supp Pool Cleanup System[c]	12.3-3	(R2,RA)	Rt Cylndr (r = 0.7m,l = 3.4m)
12.2-18	Control Rod Drive System[a]	12.3-2	(R4,RF)	Distributed Source
12.2-24	Transverse Incore Probe	12.3-2	(R4,RB)	Distributed Source
12.2-25	Reactor Internal Pumps[b]	12.3-2	(RF,R1)	Distributed Source
12.2-25	RIP Heat Exchanger	1.2-3b	El 3000	Rt Cylndr (r = 0.322m,l = 2.9m)
12.2-26	Turbine Moisture Sep/Reheater	12.3-52	(T6,TE)	Rt Cylndr (r = 1.8m,l = 1.m)
12.2-27	Turbine Condenser	12.3-53	(TD,TG)	Distributed Source
12.2-28	Condenser Filter/Demineralizer			
	Filter	12.3-51	(TC-T2)	3 Tanks, Rt Cylndr (r = 1.4m,l = 6.1m)
	Demineralizer	12.3-51	(TC,T3)	6 Tanks, Rt Cylndr(r = 1.7m,l = 5.1m)
12.2-30	SGTS Filter Train	12.3-7	(R2,RB)	surface, (3.66m x 2.54m)[d]
Applicant	Spent Fuel Storage	12.2-6	(R4,RF)	See Drawings

Notes

a Maintenance Facility

b Maintenance Facility, see Figure 1.2-3B Elevation 3000 for drywell location

c Suppression pool clean up F/D uses second of Fuel Pool F/D

d Surface area of HEPA and charcoal filter

Table 12.2-5

Radiation Sources (Continued)

B. Source Geometry

Component	Assumed Shielding Source Geometry
RHR Heat Exchanger	Homogenous source over volume of heat exchanger
RCIC Turbine	Homogenous source over volume of turbine
CUW Filter Demineralizer	80% of source in first 15cm, remainder dispersed over volume
RWCU Regen Heat Exchanger	Homogenous source over volume of exchanger
RWCU Non-Regen Heat Exchanger	Homogenous Source over volume of exchanger
LCW Collector Tank	80% non solubles in slurry on tank bottom, rest evenly dispersed in volume
LCW Filter	Homogenous source over volume of filter
LCW Demineralizer	80% of source in first 15cm, rest evenly dispersed over volume
LCW Sample Tank	Homogenous source over volume of tank
HCW Collector Tank	Homogenous source over volume of tank
HCW Demineralizer	80% of source in first 15cm, rest evenly dispersed over volume
Offgas	90% of source in first tank in first (upper) 30 cm, rest evenenly dispersed. Remaining tanks, homogenous source over tank volume.
Steam Jet Air Ejector[b]	Homogenous source over volume of ejector
Offgas Recombiner[b]	Homogenous source over subcomponent, see Figure 12.2-14[a]
CUW Backwash Receiving Tank	80% non solubles in slurry on tank bottom, rest evenly dispersed in volume
CF Backwash Receiving Tank	80% non solubles in slurry on tank bottom, rest evenly dispersed in volume
Phase Separator	90% non-solubles in slurry on tank bottom, rest evenly dispersed in volume
Spent Resin Storage Tank	Homogenous source over volume of tank
Concentrated Waste Tank	90% non-solubles in slurry on tank bottom, rest evenly dispersed in volume
Sol Dryer Feed Tank	Source evenly dispersed over volume
Sol Dryer (outlet)	Source evenly dispersed over volume
Sol Pelletizer	Source evenly dispersed over volume
Sol Mist Separator (steam)	Source evenly dispersed over volume
Sol Condenser	Source evenly dispersed over volume
Sol Drum	Source evenly dispersed over volume
FPC Filter Demineralizer	90% insolubles in first 15 cm, rest of source evenly dispersed over volume
Suppression Pool Cleanup System	90% insolubles in first 15 cm, rest of source evenly dispersed over volume
Control Rod Drive System	Exposure dependent, assume evenly dispered over length of blade
Transverse Incore Probe	Point or line geometry, see Table 12.2-24
Reactor Internal Pumps	Cylindrical source coupled to water bearing components
RIP Heat Exchanger	Homogenous source over volume of exchanger
Turbine Mositure Sep/Reheater	Homogenous source over volume of component

Table 12.2-5

Radiation Sources (Continued)

B. Source Geometry (Continued)

Component	Assumed Shielding Source Geometry
Turbine Condenser	Homogenous source over volume of condenser
Condenser Filter/Demineralizer	
Filter	Source evenly dispersed over volume of filter
Demineralizer	90% insolubles in first 15 cm, rest of source evenly dispersed over volume
SGTS Filter Train	90% Particulates on HEPA filter, remaining on charcoal filter
Spent Fuel Storage	Applicant

Notes

a See Offgas Recombiner Description, Subsection 11.3, use inventory for preheater, recombiner, condenser and cooler for recombiner inventory for shielding applications.

b Radiation levels in SJAE and Recombiner highly dependent upon power level. Actual measurements on SJAE condenser contact dose rate are 20Rads/hr at 100% power and less than 5mRad/hr at 20% power.

Table 12.2-5

Radiation Sources (Continued)

C. Shielding Geometry in meters

Component	Room Dimensions			Wall Thickness in meters[a]					
	Length	Width	Height	East	West	North	South	Floor	Ceiling
RHR Heat Exchanger	12.5	5.6	5.6	0.8	0.6	0.6	0.6	Ground	0.8
RCIC Turbine	14.6	7.8	5.6	0.8	2	0.6	0.6	Ground	0.8
CUW Filter Demineralizer	2.8	3	7.4	0.8	1	0.8	1	0.5	Hatch
RWCU Regen Heat Exchanger	7.7	3.6	6	1.4	1.4	1	1.4[a]	0.8	0.5
RWCU Non-Regen Heat Exchanger	7.4	4.4	5.6	1	1	1	1[a]	Ground	0.8
LCW Collector Tank	19	1	13	1.2	0.8	0.8	1.2	Ground	0.8
LCW Filter	16.4	10.6	8	0.8	0.8	0.8	0.8	0.8	0.8
LCW Demineralizer[b]	19.6	10.6	8	0.8	0.8	0.8	0.8	0.8	0.8
LCW Sample Tank	19	10	13	1.2	0.8	1.2	0.8	Ground	0.8
HCW Collector Tank	9	11.2	5.4	0.8	0.8	0.8	1.2	Ground	0.8
HCW Demineralizer[b]	19.6	10.6	8	0.8	0.8	0.8	0.8	0.8	0.8
Offgas	9.1	11	16	1	1	1	1	2.5	1
Steam Jet Air Ejector and Recombiner Room	9.1	14.2	7	1	1	1	1	1	1
CUW Backwash Receiving Tank	6.6	7.4	5.6	1	0.8	0.8	1	Ground	0.8
CF Backwash Receiving Tank	5	5	25	1	1	1	1	2.5	Hatch
Phase Separator	16	8.4	4.6	0.8	0.8	0.8	1.2	0.8	0.8
Spent Resin Storage Tank	6.4	6.4	4.6	0.8	0.8	0.8	0.8	0.8	0.8
Concentrated Waste Tank	4.6	5	5.4	0.8	0.8	1.2	0.8	Ground	0.8
Sol Dryer Feed Tank	9.4	7.2	6.2	0.8	0.8	0.8	0.8	0.8	0.8
Sol Dryer (outlet)[c]	9.2	5.2	8	0.8	0.8	0.8	0.8	0.8	0.8
Sol Pelletizer	9.2	5.4	6.8	0.8	0.8	0.8	0.8	0.8	0.8
Sol Mist Separator (steam)[c]	9.2	5.7	8	0.8	0.8	0.8	0.8	0.8	0.8
Sol Condenser	4.2	7.2	6.2	0.8	0.8	0.8	0.8	0.8	0.8
Sol Drum	3.2	3	8	0.8	0.8	0.8	0.8	0.8	0.8
FPC Filter Demineralizer	2.2	3.2	7.4	0.8	1	0.8	0.8	0.5	Hatch
Suppression Pool Cleanup Sys	3.2	3.2	7.4	0.5	0.8	0.8	0.8	0.5	Hatch
Control Rod Drive System[f]	7.6	33.4	5.8	0.6	0.6	0.6	0.6	0.8	0.6
Transverse Incore Probe	3	7.5	2.7	1	1	1	1	Mezz	0.6

Table 12.2-5

Radiation Sources (Continued)

C. Shielding Geometry in meters (Continued)

Component	Room Dimensions			Wall Thickness in meters[e]					
	Length	Width	Height	East	West	North	South	Floor	Ceiling
Reactor Internal Pump[f]	8.2	8.5	5.8	0.6	0.6	0.6	0.6	0.8	0.6
RIP Heat Exchanger	Primary Containment								
Turbine Moisture Sep/Reheater	12.4	47.6	8.5	1	1	1	1	1	1
Turbine Condenser	14.2	36	25	3.5	2.5	1	1	2.5	Turbine
Condenser Filter	5	21.1	8	2.5[a]	1	1	1	1	Hatch
Condenser Demineralizer	9.8	17.3	9	1	1	1	1.6	1	1
SGTS Filter Train	14.4	5	8.2	0.2	0.5	0.2	0.2	2	0.6
Spent Fuel Storage	9.4	14	4.1	2	2	2	2	2	7.4[d]

Notes

[a] Movenable Wall
[b] LCW and HCW Demineralizer share same room
[c] Solid dryer and Mist Separator share same room
[d] 7.4 meter water depth above fuel elements
[e] North refers to plant 0 degree orientation, east = 90 degrees
[f] Maintenance Facility

Table 12.2-5

Radiation Sources (Continued)

D. Pipe Chase Detail

Pipe Space (PS)	Level	Loc. ...g	System	Number Pipes	Size[a]	Source[b]	Shield Wall Thickness in meters East	West	North	South
RHR(A)	1F	(RC,R6)	RHR	1	273x237	RC	0.6	PC	0.6	0.6
			RCIC	1	168x140	RS	0.6	PC	0.6	0.6
	B1F	(RC,R6)	RHR	1	273x237	RC	0.6	PC	0.6	0.6
			RCIC	1	168x140	RS	0.6	PC	0.6	0.6
			RCIC	1	356X333	SP	0.6	PC	0.6	0.6
	B2F	(RC,R6)	RHR	1	273x237	RC	0.6	PC	0.6	0.6
			RCIC	1	168x140	RS	0.6	PC	0.6	0.6
			RCIC	1	356X333	SP	0.6	PC	0.6	0.6
	B3F	(RC,RA)	RHR	1	273x237	RC	0.6	PC	0.6	0.6
			RCIC	1	168x140	RS	0.6	PC	0.6	0.6
			RCIC	1	356X333	SP	0.6	PC	0.6	0.6
RHR(B)	1F	(RD,R2)	RHR	1	273x237	RC	PC	0.6	0.6	0.6
			HPCF	1	334x303	RC	PC	0.6	0.6	0.6
	B1F	(RD,R2)	RHR	1	273x237	RC	PC	0.6	0.6	0.6
			HPCF	1	334x303	RC	PC	0.6	0.6	0.6
	B2F	(RD,R2)	RHR	1	273x237	RC	PC	0.6	0.6	0.6
			HPCF	1	334x303	RC	PC	0.6	0.6	0.6
	B3F	(RE,R2)	RHR	1	273x237	[illegible]	PC	0.6	0.6	0.6
			HPCF	1	334x303	RC	PC	0.6	0.6	0.6
RHR(C)	1F	(RE,R6)	RHR	1	273x237	RC	0.6	PC	0.6	0.6
			HPCF	1	334x303	RC	0.6	PC	0.6	0.6
	B1F	(RE,R6)	RHR	1	273x237	RC	0.6	PC	0.6	0.6
			HPCF	1	334x303	RC	0.6	PC	0.6	0.6
	B2F	(RE,R6)	RHR	1	273x237	RC	0.6	PC	0.6	0.6
			HPCF	1	334x303	RC	0.6	PC	0.6	0.6
	B3F	(RE,R6)	RHR	1	273x237	RC	0.6	PC	0.6	0.6
			HPCF	1	334x305	RC	0.6	PC	0.6	0.6
FPC, CUW	2F	(RB,R3)	FPC	2	273x255	1% RC	1.2	1.2	1.2	1.2
	1F	(RB,R3)	FPC	2	273x255	1% RC	1.2	1.2	1.2	1.2
			CUW	1	219x189	RC	1.6	1.2	1.2	1.2
	B1F	(RB,R3)	FPC	2	273x255	1% RC	1.2	1.2	1.2	1.2
			CUW	1	219x189	RC	1.6	1.2	1.2	1.2
	B3F	(RB,R2)	CUW	2	168x140	RC	0.6	0.6	0.8	0.8
MSL/FDW	1F	(RB,R4)	MSL	4	711x640	RS	1.6	1.6	1.6	1.6
			FDW	4	550x480	10% RS[c]	1.6	1.6	1.6	1.6
SPCU	B2F	(RC,R2)	SPCU	1	219x203	SP	PC	0.8	0.8	0.8

Notes

a Pipe size given as outside diamter in millimeters and inside diameter in millimeters.

b Source is defined by RC = reactor coolant water, see Tables 11.2-2 through 11.2-5. RS is reactor steam, see Tables 11.2-1 and 4. SP = Suppression pool water = 10% RC (normal operations), Reg Guide 1.7 (LOCA conditions).

c No N-16 or noble gases in feedwater.

Table 12.2-6

FISSION PRODUCT GAMMA SOURCE STRENGTH IN THE RHR HEAT EXCHANGER

Energey Bounds (Mev)	Gamma Source (Mev/sec)
>4.0	0.0
3.0 - 4.0	2.3E 08
2.6 - 3.0	2.8E 10
2.2 - 2.6	8.2E 10
1.8 - 2.2	1.6E 11
1.4 - 1.8	1.1E 12
0.9 - 1.4	2.3E 12
0.4 - 0.9	3.5E 12
0.1 - 0.4	4.3E 11
0.0 - 0.1	5.4E 09

TABLE 12.2-13

LIQUID RADWASTE COMPONENT INVENTORIES

The inventory in the liquid radwaste components is provided in the following table for a deep bed system. The data in Table 12.2-13 were generated assuming a fission product release from the fuel equivalent to that required to produce 100,000 μCi/sec of offgas following a 30-min holdup period.

LCW COLLECTOR TANK

Source Volume = $90m^3$

Total Micro-Curies = 2.67E 07

Halogens		Soluble Fission Products		Insoluble Fission Products		Activation Products	
Isotope	μCI	Isotope	μCI	Isotope	μCI	Isotope	μCI
I 131	3.19E 06	RB 89	7.62E 03	Y 91	5.51E 04	NA 24	6.83E 05
I 132	4.57E 05	SR 89	1.37E 05	Y 92	1.11E 05	P 32	1.86E 05
I 133	2.98E 06	SR 90	1.13E 04	Y 93	1.85E 05	CR 51	7.24E 06
I 134	3.00E 05	Y 90	1.13E 04	ZR 95	1.12E 04	MN 34	1.10E 05
I 135	1.40E 06	SR 91	1.89E 05	NB 95	1.00E 04	MN 56	6.67E 05
		SR 92	1.44E 05	RU 103	2.61E 04	CO 58	2.87E 05
		MO 99	5.28E 05	RH 103M	2.61E 04	CO 60	6.37E 05
		TC 99M	5.28E 05	RU 106	4.76E 03	FE 55	1.61E 06
		TE 129M	4.95E 04	RH 106	4.76E 03	FE 59	4.01E 04
		TE 131M	1.25E 04	LA 140	3.45E 05	NI 63	1.63E 03
		TE 132	3.06E 03	CE 141	3.74E 04	CU 64	1.72E 06
		CS 134	4.33E 04	CE 144	4.72E 03	ZN 65	3.13E 05
		CS 136	1.61E 04	PR 143	2.72E 03	AG 110M	1.57E 03
		CS 137	1.18E 05			W 187	3.07E 04
		CS 138	3.24E 04				
		BA 140	3.45E 05				
		NP 239	1.85E 06				
TOTAL	8.33E 06	TOTAL	4.02E 06	TOTAL	8.25E 05	TOTAL	1.35E 07

TABLE 12.2-13 (Continued)

LIQUID RADWASTE COMPONENT INVENTORIES

LCW FILTER

Source Volume = 1.2m^3/Batch (Backwash)

Total Micro-Curies = 3.08E 06

Halogens		Soluble Fission Products		Insoluble Fission Products		Activation Products	
Isotope	µCI	Isotope	µCI	Isotope	µCI	Isotope	µCI
I 131	0.	RB 89	0.	Y 91	5.49E 04	NA 24	0.
I 132	0.	SR 89	0.	Y 92	5.56E 04	P 32	0.
I 133	0.	SR 90	0.	Y 93	9.59E 04	CR 51	0.
I 134	0.	Y 90	0.	ZR 95	1.12E 04	MN 54	5.74E 05
I 135	0.	SR 91	0.	NB 95	9.63E 03	MN 56	1.66E 05
		SR 92	0.	RU103	2.54E 04	CO 58	1.44E 05
		MO 99	0.	RH103M	2.54E 04	CO 60	3.35E 05
		TC 99M	0.	RU106	4.97E 03	FE 55	1.69E 06
		TE 129M	0.	RH106	4.97E 03	FE 59	3.93E 04
		TE 131M	0.	LA 140	2.91E 05	NI 63	8.57E 02
		TE 132	0.	CE 141	3.57E 04	CU 64	0.
		CS 134	0.	CE 144	4.92E 03	ZN 65	0.
		CS 136	0.	PR 143	2.31E 03	AG110M	1.63E 03
		CS 137	0.			W 187	1.71E 04
		CS 138	0.				
		BA 140	0.				
		NP 239	0.				
TOTAL	0.	TOTAL	0.	TOTAL	6.22E 05	TOTAL	2.45E 06

TABLE 12.2-13 (Continued)

LIQUID RADWASTE COMPONENT INVENTORIES

LCW DEMINERALIZER

Source Volume = 1.2m^3 (RESIN)

Total Micro-Curies = 5.37E 07

Halogens		Soluble Fission Products		Insoluble Fission Products		Activation Products	
Isotope	µCi	Isotope	µCi	Isotope	µCi	Isotope	µCi
I 131	5.61E 06	RB 89	7.62E 03	Y 91	2.88E 03	NA 24	7.34E 05
I 132	4.57E 05	SR 89	6.62E 05	Y 92	1.13E 03	P 32	4.54E 05
I 133	3.29E 06	SR 90	8.48E 04	Y 93	1.95E 03	CR 51	2.65E 07
I 134	3.00E 05	Y 90	8.48E 04	ZR 95	6.07E 02	MN 54	3.86E 05
I 135	1.42E 06	SR 91	1.96E 05	NB 95	4.23E 02	MN 56	3.37E 05
		SR 92	1.44E 05	RU 103	1.17E 03	CO 58	7.86E 05
		MO 99	6.54E 05	RH 103M	1.17E 03	CO 60	2.39E 06
		TC 99M	6.54E 05	RU 106	3.42E 02	FE 55	1.21E 05
		TE 129M	2.00E 05	RH 106	3.42E 02	FE 59	1.89E 03
		TE 131M	1.42E 04	LA 140	8.04E 03	NI 63	6.16E 03
		TE 132	3.89E 03	CE 141	1.52E 03	CU 64	1.83E 06
		CS 134	3.15E 05	CE 144	3.33E 02	ZN 65	2.13E 06
		CS 136	3.71E 04	PR 143	6.57E 01	AG 110M	1.10E 02
		CS 137	8.85E 05			W 187	3.48E 02
		CS 138	3.24E 04				
		BA 140	7.89E 05				
		NP 239	2.23E 06				
TOTAL	1.11E 07	TOTAL	6.99E 06	TOTAL	2.00E 04	TOTAL	3.56E 07

TABLE 12.2-13 (Continued)

LIQUID RADWASTE COMPONENT INVENTORIES

LCW SAMPLE TANK

Source Volume = 105m^3

Total Micro-Curies = 2.37E 05

Halogens		Soluble Fission Products		Insoluble Fission Products		Activation Products	
Isotope	µCI	Isotope	µCI	Isotope	µCI	Isotope	µCI
I 131	3.20E 04	RB 89	7.85E 01	Y 91	5.51E 01	NA 24	6.8[illegible]E 03
I 132	4.59E 03	SR 89	1.37E 03	Y 92	1.12E 02	P 32	1.86E 03
I 133	2.98E 04	SR 90	1.13E 02	Y 93	1.86E 02	CR 51	7.24E 04
I 134	3.03E 03	Y 90	1.13E 02	ZR 95	1.12E 01	MN 54	6.05E 02
I 135	1.40E 04	SR 91	1.89E 03	NB 95	1.00E 01	MN 56	3.68E 02
		SR 92	1.44E 03	RU103	2.61E 01	CO 58	1.58E 03
		MO 99	5.28E 03	RH103M	2.61E 01	CO 60	3.50E 03
		TC 99M	5.28E 03	RU106	4.76E 00	FE 55	1.61E 03
		TE 129M	4.95E 02	RH106	4.76E 00	FE 59	4.01E 01
		TE 131M	1.25E 02	LA 140	3.46E 02	NI 63	8.94E 00
		TE 132	3.06E 01	CE 141	3.74E 01	CU 64	1.72E 04
		CS 134	4.33E 02	CE 144	4.72E 00	ZN 65	3.13E 03
		CS 136	1.61E 02	PR 143	2.72E 00	AG 110M	1.57E 00
		CS 137	1.18E 03			W 187	3.09E 01
		CS 138	3.28E 02				
		BA 140	3.45E 03				
		NP 239	1.85E 04				
TOTAL	8.34E 04	TOTAL	4.02E 04	TOTAL	8.27E 02	TOTAL	1.13E 05

TABLE 12.2-13 (Continued)

LIQUID RADWASTE COMPONENT INVENTORIES

HCW COLLECTOR TANK

Source Volume = $15m^3$

Total Micro-Curies = 5.61E 04

Halogens		Soluble Fission Products		Insoluble Fission Products		Activation Products	
Isotope	µCi	Isotope	µCi	Isotope	µCi	Isotope	µCi
I 131	2.30E 03	RB 89	4.92E 01	Y 91	1.94E 01	NA 24	3.08E 03
I 132	2.88E 03	SR 89	4.92E 01	Y 92	6.94E 02	P 32	9.66E 01
I 133	1.14E 04	SR 90	3.45E 00	Y 93	9.95E 02	CR 51	2.96E 03
I 134	1.90E 03	Y 90	3.45E 00	ZR 95	3.88E 00	MN 54	3.45E 01
I 135	8.22E 03	SR 91	1.01E 03	NB 95	3.86E 00	MN 56	4.19E 03
		SR 92	9.02E 02	RU103	9.81E 00	CO 58	9.85E 01
		MO 99	8.76E 02	RH103M	9.81E 00	CO 60	1.95E 02
		TC 99M	8.76E 02	RU106	1.48E 00	FE 55	4.95E 02
		TE 129M	1.93E 01	RH106	1.48E 00	FE 59	1.47E 01
		TE 131M	3.80E 01	LA 140	1.90E 02	NI 63	4.95E-01
		TE 132	4.46E 00	CE 141	1.47E 01	CU 64	8.36E 03
		CS 134	1.33E 01	CE 144	1.48E 00	ZN 65	9.89E 01
		CS 136	8.76E 00	PR 143	1.45E 00	AG110M	4.94E-01
		CS 137	3.60E 01			W 187	1.08E 02
		CS 138	2.06E 02				
		BA 140	1.90E 02				
		NP 239	3.51E 03				
TOTAL	2.66E 04	TOTAL	7.79E 03	TOTAL	1.95E 03	TOTAL	1.97E 04

TABLE 12.2-13 (Continued)

LIQUID RADWASTE COMPONENT INVENTORIES

HCW DEMINERALIZER

Source Volume = $1.2 m^3$/(RESIN)

Total Micro-Curies = 3.85E 03

Halogens		Soluble Fission Products		Insoluble Fission Products		Activation Products	
Isotope	µCI	Isotope	µCI	Isotope	µCI	Isotope	µCI
I 131	2.79E 02	RB 89	4.92E-01	Y 91	1.40E 01	NA 24	4.61E 01
I 132	2.88E 01	SR 89	3.21E 01	Y 92	7.00E 00	P 32	2.04E 01
I 133	2.06E 02	SR 90	5.49E 00	Y 93	1.21E 01	CR 51	1.18E 03
I 134	1.90E 01	Y 90	5.49E 00	ZR 95	3.01E 00	MN 54	4.65E 01
I 135	8.94E 01	SR 91	1.23E 01	NB 95	1.89E 00	MN 56	4.20E 01
		SR 92	9.04E 00	RU103	5.31E 00	CO 58	8.00E 01
		MO 99	3.97E 01	RH103M	5.31E 00	CO 60	3.03E 02
		TC 99M	3.97E 01	RU106	2.05E 00	FE 55	7.47E 02
		TE 129M	9.09E 00	RH106	2.05E 00	FE 59	8.76E 00
		TE 131M	8.93E-01	LA 140	3.60E 01	NI 63	7.91E-01
		TE 132	2.32E-01	CE 141	6.72E 00	CU 64	1.15E 02
		CS 134	1.99E 01	CE 144	1.97E 00	ZN 65	1.27E 02
		CS 136	1.69E 00	PR 143	2.91E-01	AG110M	6.41E-01
		CS 137	5.73E 01			W 187	2.16E 00
		CS 138	2.06E 00				
		BA 140	3.60E 01				
		NP 239	1.37E 02				
TOTAL	6.23E 02	TOTAL	4.09E 02	TOTAL	9.76E 01	TOTAL	2.72E 03

Table 12.2-19

ANNUAL AIRBORNE RELEASES FOR OFFSITE DOSE EVALUATIONS (Curies)

Nuclide	R/B	Turbine	Radwaste	Mechanical Vacuum Pump	Turbine Seal	Offgas	Drywell
Kr-83m						5.4E-06	8.4E-04
Kr-85m	1.2E+00	9.9E+00				9.6E+00	3.6E-03
Kr-85					7.0E-03	5.7E+02	6.7E-04
Kr-87	7.9E-01	2.4E+01				4.9E-10	3.3E-03
Kr-88	1.6E+00	3.6E+01				8.7E-02	7.4E-03
Kr-89	7.9E-01	2.3E+02	1.2E+01				9.0E-04
Kr-90							3.3E-04
Xe-131m					6.0E-03	5.0E+01	3.3E-04
Xe-133m						8.5E-02	2.0E-03
Xe-133	4.4E+01	6.0E+01	8.7E+01	3.9E+02	3.0E+00	1.8E+03	1.2E-01
Xe-135m	2.4E+01	1.6E+02	2.1E+02		8.0E+00		8.8E-04
Xe-135	5.0E+01	1.3E+02	1.1E+02	1.5E+02	7.0E+00		2.7E-02
Xe-137	7.1E+01	4.0E+02	3.3E+01				1.3E-03
Xe-138	3.2E+00	4.0E+02	7.9E-01		2.5E+01		2.8E-03
Xe-139							4.1E-04
I-131	3.8E-02	1.5E-01	1.3E-02	5.5E-02	6.5E-04		2.6E-03
I-132	3.2E-01	1.3E+00	1.1E-01	4.7E-01			3.6E-04
I-133	2.5E-01	1.0E+00	8.9E-02	3.6E-01	4.3E-03		2.6E-03
I-134	5.5E-01	2.2E+00	2.0E-01	8.1E-01			2.4E-04
I-135	3.5E-01	1.4E+00	1.2E-01	5.1E-01			1.1E-03
H-3	3.0E+01	3.0E-01			6.0E+00		6.9E+00
C-14						9.3E+00	
Na-24							4.0E-03
P-32							9.2E-04
Ar-41						6.7E+00	
Cr-51	3.3E-04	7.4E-04	5.8E-04				3.3E-02
Mn-54	6.6E-04	4.9E-04	3.3E-03				4.5E-04
Mn-56							3.6E-03
Fe-55							6.5E-03
Fe-59	1.5E-04	8.2E-05	2.5E-04				1.8E-04
Co-58	1.6E-04	8.2E-04	1.6E-04				1.2E-03
Co-60	1.6E-03	8.2E-04	5.8E-03				2.6E-03
Ni-63							6.5E-06
Cu-64							1.0E-02
Zn-65	1.6E-03	4.9E-03	2.5E-04				1.3E-03
Rb-89							4.2E-05
Sr-89	4.9E-05	4.9E-03					5.9E-04
Sr-90	4.9E-06	1.6E-05					4.6E-05
Y-90							4.6E-05
Sr-91							1.0E-03
Sr-92							7.8E-04

Table 12.2-19

ANNUAL AIRBORNE RELEASES FOR OFFSITE DOSE EVALUATIONS (Curies) (Continued)

Nuclide	R/B	Turbine	Radwaste	Mechanical Vacuum Pump	Turbine Seal	Offgas	Drywell
Y-91							2.4E-04
Y-92							6.1E-04
Y-93							1.1E-03
Zr-95	4.9E-04	3.3E-05	6.6E-04				4.8E-05
Nb-95	1.6E-03	4.9E-06	3.3E-06				4.4E-05
Mo-99	9.9E-03	1.6E-02	2.5E-06				3.3E-03
Tc-99m							3.1E-04
Ru-103	3.3E-04	4.1E-05	8.2E-07				1.1E-04
Rh-103m							1.1E-04
Ru-106							1.9E-05
Rh-106							1.9E-05
Ag-110m	6.6E-07						1.8E-10
Sb-124	3.3E-05	8.2E-05	5.8E-05				
Te-129m							2.2E-04
Te-131m							7.7E-05
Te-132							1.9E-05
Cs-134							1.7E-04
Cs-136							8.1E-05
Cs-137							4.7E-04
Cs-138							1.7E-04
Ba-140	3.3E-03	8.2E-03	3.3E-06				1.8E-03
La-140							1.8E-03
Ce-141	3.3E-04	8.2E-03	5.8E-06				1.7E-04
Ce-144							1.9E-05
Pr-144							1.9E-05
W-187							1.9E-04
Np-239							1.2E-02

Table 12.2-29

Steam Jet Air Ejector Inventory in Curies

Isotope	1st Stage Ejector	Condenser	2nd Stage Ejector
Kr-83m	6.3E-04	2.0E-02	2.0E-03
Kr-85m	1.2E-03	3.7E-02	3.7E-03
Kr-85	4.0E-06	1.2E-04	1.2E-05
Kr-87	4.0E-03	1.2E-01	1.2E-02
Kr-88	4.0E-03	1.2E-01	1.2E-02
Kr-89	2.5E-02	7.5E-01	7.5E-02
Kr-90	4.5E-02	1.4E+00	1.4E-01
Kr-91	3.0E-02	8.9E-01	8.9E-02
Kr-92	1.5E-03	4.6E-02	4.6E-03
Kr-93	7.9E-05	2.4E-03	2.4E-04
Kr-94	2.1E-17	6.4E-16	6.4E-17
Kr-95	4.0E-10	1.2E-08	1.2E-09
Kr-97	1.5E-25	4.5E-24	4.5E-25
Total KR	1.1E-01	3.3E+00	3.3E-01
Xe-131m	3.0E-06	9.0E-05	9.0E-06
Xe-133m	5.8E-05	1.7E-03	1.7E-04
Xe-133	1.6E-03	4.9E-02	4.9E-03
Xe-135m	5.2E-03	1.5E-01	1.5E-02
Xe-135	4.4E-03	1.3E-01	1.3E-02
Xe-137	2.9E-02	8.7E-01	8.7E-02
Xe-138	1.8E-02	5.3E-01	5.3E-02
Xe-139	4.7E-02	1.4E+00	1.4E-01
Xe-140	3.6E-02	1.1E+00	1.1E-01
Xe-141	8.5E-04	2.6E-02	2.6E-03
Xe-142	5.0E-05	1.5E-03	1.5E-04
Xe-143	2.2E-13	6.7E-12	6.7E-13
Xe-144	1.1E-07	3.3E-06	3.3E-07
Total XE	1.4E-01	4.3E+00	4.3E-01
Noble Gas Total[a]	1.1E-01	7.6E+00	7.6E-01
N-16[a]	3.5E-01	1.3E+01	1.3E+00

Notes:

[a] Value given is estimated N-16 inventory at 100% power. Value varies in an unknown fashion with power. Based upon operating measurements, the value for N-16 at 20% power is close to zero. Multiply value by a factor of 4 for use with hydrogen water chemistry.

Table 12.2-30

Standby Gas Treatment System Inventory

Isotope	Curies	Isotope	Curies
I-131	1.5E-02	Y-91	8.5E-04
I-132	1.5E-03	Y-92	3.6E-04
I-133	1.1E-02	Y-93	6.7E-04
I-134	1.0E-03	Zr-95	1.9E-04
I-135	4.8E-03	Nb-95	1.0E-04
		Mo-99	2.0E-03
		Tc-99m	1.9E-04
Na-24	1.7E-02	Ru-103	2.9E-04
P-32	7.4E-03	Rh-103m	2.9E-04
Cr-51	4.4E-01	Ru-106	4.0E-04
Mn-54	5.4E-02	Rh-106	4.0E-04
Mn-56	1.5E-02	Ag-110m	1.1E-10
Fe-55	1.8E+00	Te-129m	4.9E-04
Fe-59	3.5E-03	Te-131m	4.6E-05
Co-58	3.7E-02	Te-132	1.2E-05
Co-60	9.6E-01	Cs-134	6.2E-03
Ni-63	4.7E-04	Cs-136	8.7E-05
Cu-64	6.0E-03	Cs-137	3.3E-02
Zn-65	1.8E-02	Cs-138	1.0E-04
Rb-89	2.5E-05	Ba-140	1.9E-03
Sr-89	1.9E-03	La-140	1.9E-03
Sr-90	3.2E-03	Ce-141	3.6E-01
Y-90	3.2E-03	Ce-144	3.1E-04
Sr-91	6.2E-04	Pr-144	3.1E-04
Sr-92	4.7E-04	W-187	1.1E-04
		Np-239	7.0E-03

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12.3 RADIATION PROTECTION DESIGN FEATURES

12.3.1 Facility Design Features

The ABWR Standard Plant is designed to meet the intent of Regulatory Guide 8.8 (i.e., to keep radiation exposures to plant personnel as low as reasonably achievable (ALARA)). This section describes the component and system designs in addition to the equipment layout employed to maintain radiation exposures ALARA. Consideration of individual systems is provided to illustrate the application of these principles.

Material application for primary coolant piping, tubing, vessel internal surfaces, and other components in contact with the primary coolant is discussed in the following pages. Typical nickel and cobalt contents of the principal materials applied are given in Table 12.3-2.

Carbon steel is used in a large portion of the system piping and equipment outside of the nuclear steam supply system. Carbon steel is typically low in nickel content and contains a very small amount of cobalt impurity.

Stainless steel is used in portions of the system such as the reactor internal components and heat exchanger tubes where high corrosion resistance is required. The nickel content of the stainless steels is in the 9 to 10.5% range and is controlled in accordance with applicable ASME material specifications. Cobalt content is controlled to less than 0.05% in the XM-19 alloy used in the control rod drives.

A previous review of materials certifications indicated an average cobalt content of only 0.15% in austenitic stainless steels.

Ni-Cr-Fe alloys such as Inconel 600 and Inconel X750, which have high nickel content, are used in some reactor vessel internal components. These materials are used in applications for which there are special requirements to be satisfied (such as possessing specific thermal expansion characteristics along with adequate corrosion resistance) and for which no suitable alternative low-nickel material is available. Cobalt content in the Inconel X750 used in the fuel assemblies is limited to 0.05%.

Stellite is used for hard facing of components which must be extremely wear resistant. Use of high cobalt alloys such as Stellite is restricted to those applications where no satisfactory alternative material is available. An alternative material (Colmonoy) has been used for some hard facings in the core area.

12.3.1.1 Equipment Design for Maintaining Exposure ALARA

This subsection describes specific components as well as system design features that aid in maintaining the exposure of plant personnel during system operation and maintenance ALARA. Equipment layout to provide ALARA exposures of plant personnel are discussed in Subsection 12.3.1.2.

(1) Pumps

Pumps located in radiation areas are designed to minimize the time required for maintenance. Quick change cartridge-type seals on pumps, and pumps with back pullout features that permit removal of the pump impeller or mechanical seals without disassembly of attached piping, are employed to minimize exposure time during pump maintenance. The configuration of piping about pumps is designed to provide sufficient space for efficient pump maintenance. Provisions are made for slushing and in certain cases chemically cleaning pumps prior to maintenance. Pump casing drains provide a means for draining pumps to the sumps prior to disassembly, thus reducing the exposure of personnel and decreasing the potential for contamination. Where two or more pumps conveying highly radioactive fluids are required for operational reasons to be located adjacent to each other, shielding is provided between the pumps to maintain exposure levels

ALARA. An example of this situation is the RWCS circulation pumps. Pumps adjacent to other highly radioactive equipment are also shielded to reduce the maintenance exposure, for example, in the radwaste system.

Whenever possible, operation of the pumps and associated valving for radioactive systems is accomplished remotely. Pump control instrumentation is located outside high radiation areas, and motor- or pneumatic-operated valves and valve extension stems are employed to allow operation from outside these areas.

(2) Instrumentation

Instruments are located in low radiation areas such as shielded valve galleries, corridors, or control rooms, whenever possible. Shielded valve galleries provided for this purpose include those for the RWCS, FPCC, and radwaste (cleanup phase separator, spent resin tank, and waste evaporator) systems. Instruments required to be located in high radiation areas due to operations requirements are designed such that removal of these instruments to low radiation areas for maintenance is possible. Sensing lines are routed from taps on the primary system in order to avoid placing the transmitters or readout devices in high radiation areas. For example, reactor water level as well as recirculation system pressure sensing instruments are located outside the drywell.

Liquid service equipment for systems containing radioactive fluids are provided with vent and backflush provisions. Instrument lines, except those for the reactor vessel, are designed with provisions for backflushing and maintaining a clean fill in the sensing lines. The reactor vessel sensing lines may be flushed with condensate following reactor blowdown.

(3) Heat Exchangers

Heat exchangers are constructed of stainless steel or Cu/Ni tubes to minimize the possibility of failure and reduce maintenance requirements. The heat exchanger design allows for the complete drainage of fluids from the exchanger, avoiding pooling effects that could lead to radioactive crud deposition. Connections are available for condensate or demineralized water flushing of the heat exchangers. For the reactor water clean up (CUW) system, separate connections are provided to chemically decontaminate both the heat exchangers (both regenerative and non-regenerative) and the pumps. The other main heat exchangers (RHR and RIP) are provided connections by which the exchangers can be flushed with clean water. The last main heat exchanger, the fuel pool heat exchanger, is downstream of the filter demineralizer and is therefore not subjected to flows containing significant amounts of fission or activation products. In all cases, the pumps directly involved with the heat exchangers are also inline for decontamination with the exchangers.. Instrumentation and valves are remotely operable to the maximum extent possible in the shielded heat exchanger cubicles, to reduce the need for entering these high radiation areas.

(4) Valves

Valve packing and gasket material are selected on a conservative basis, accounting for environmental conditions such as temperature, pressure, and radiation tolerance requirements to provide a long operating life. Valves have back seats to minimize the leakage through the packing. Straight-through valve configurations were selected where practical, over those which exhibit flow discontinuities or internal crevices to minimize crud trapping. Teflon gaskets are not used.

Wherever possible, valves in systems containing radioactive fluids are separated from those for "clean" services to reduce the radiation exposure from adjacent valves and piping during maintenance.

Pneumatic or mechanically operated valves are employed in high radiation areas, whenever practical, to minimize the need for entering these areas. For certain situations, manually operated valves are required, and in such cases extension valve stems are provided which are operated from a shielded area. Flushing and drain provi-

sions are employed in radioactive systems to reduce exposure to personnel during maintenance.

For areas in which especially high radiation levels are encountered, valving is reduced to the maximum extent possible with the bulk of the valve and piping located in an adjacent valve gallery where the radiation levels are lower.

alarmed to the control room. The TIP entry location into the room from the drywell is via the suppression pool instrumentation tunnel and then upward into the room. Area radiation monitors in both TIP room and cpooler room maintain a secondary surveillance of both rooms being alarmed to both the control rooms and locally in the TIP facility. An inadvertent withdrawal of the TIP will result in alarming both the position sensor and area radiation monitors resulting in local alarms to egress the area.

(3) ECCS Components

The ECCS systems are located in separately shielded cubicles. Shield labyrinths are provided to gain entry into the cubicles, and equipment removal doors are shielded with removable horizontally and vertically lapped concrete block. Piping to and from the ECCS system is routed through shielded pipe chases. Access into the cubicles is not required to operate the systems. In general, the radiation levels in the open corridors of the reactor building are less than 1 mR/hr. except during RHR shutdown cooling mode operation, when radiation levels may temporarily range between 1 and 5 mR/hr in areas near the RHR cubicles.

The RWC system pumps are located in a shielded cubicle designed to reduce the radiation levels in the adjoining open corridor to less than 1 mR/hr. The pumps are separated by shield walls to allow operation of one of the pumps while performing maintenance on the other. Dose rates at this pump due to the operating pump and piping are less than 5 mR/hr. A shielded valve gallery is employed to permit manual operation of the valves associated with the RWC system pumps without entering the pump area. Piping for the pumps is directly routed from the steam tunnel to the RWC system pump area.

The CRD maintenance room walls are designed to reduce dose rates in the adjoining corridor to less than 1 mR/hr during all CRD maintenance operations except CRD transfer, when dose rates in the corridor temporarily range between 1 and 5 mR/hr.

The main steam lines are located in the shielded steam tunnel. The steam tunnel reduces the dose rates from the steam lines to less than 1 mR/hr in all adjoining areas except the roof of the steam tunnel, which is less that 5 mR/hr.

(4) Fuel Components

The fuel storage pool is designed to insure that the dose rate in adjoining areas is less than 1 mR/hr. During normal operation, dose rates in the pump area are less than 1 mR/hr. During an isolation transient, however, dose rates in the area temporarily increase to 700 mR/hr. Due to the nature of the event, egress from the area can be accomplished well before dose rates reach this level. Access to equipment in this area is not required during this occurrence. An individual in this area will know that the dose rate is increasing since a local-mounted area radiation monitoring sensor, converter, indicating auxiliary unit, and audio alarm are provided.

(5) Control Room

The dose rate in the control room is much less than 0.6 mR/hr during normal reactor operating conditions. The outer walls of the control building are designed to attenuate radiation from radioactive materials contained within the reactor building and from possible airborne radiation surrounding the control building following a LOCA. The walls provide sufficient shielding to limit the direct-shine exposure of control room personnel following a LOCA to a fraction of the 5 Rem limit as is required by 10CFR50, Appendix A, Criterion 19. Shielding for the outdoor air cleanup filters is also provided to allow temporary access to the mechanical equipment area of the control building following a LOCA, should it be required.

(6) The main steam tunnel extends from the primary containment boundary in the reactor building through the control building up to the turbine stop valves. The primary purpose of the steam tunnel is to shield the plant complex from N-16 gamma shine in the main steam lines. A minimum of 1.6 meters

of concrete or its equivalent (other material or distance) is required on any ray pathway from the main steam lines to any point which may be inhabited during normal operations. The design of the steam tunnel is shown on Figures 1.2-14, 1.2-15, 1.2-20, 1.2-21, and 1.2-28. The tunnel is classified as Seismic Category 1 in the reactor building and in the control building and is designed to UBC Seismic Standards in the turbine building. The interface between the buildings provides for bayonet connection to permit differential building motion during seismic events and shielding in the areas between buildings. The exact details on the bayonet design are not shown on the referenced arrangement drawings but requires complete shielding in the building interface area. The tunnel also serves a secondary purpose as a relief and release pathway for high energy events in the reactor building. Any high energy event (line break) in the reactor building will, through a series of blow out panels, vent into the steam tunnel and from the steam tunnel through the tunnel vent shaft to the turbine building (see Figure 1.2-28) for processing to the plant stack. See Subsection 6.2.3.3.1 for more complete description of this function.

12.3.3 Ventilation

The HVAC systems for the various buildings in the plant are discussed in Section 9.4, including the design bases, system descriptions, and evaluations with regard to the heating, cooling, and ventilating capabilities of the systems. This section discusses the radiation control aspects of the HVAC systems.

12.3.3.1 Design Objectives

The following design objectives apply to all building ventilation systems:

(1) The systems shall be designed to make airborne radiation exposures to plant personnel and releases to the environment ALARA. To achieve this objective, the guidance provided in Regulatory Guide 8.8 shall be followed.

(2) The concentration of radionuclides in the air in areas accessible to personnel for normal plant surveillance and maintenance shall be kept below the limits of 10CFR20 during normal power operation.

The applicable guidance provided in Regulatory Guide 1.52 has been implemented for the ESF filter systems for the control building outdoor air cleanup system and the standby gas treatment system (STGS) as described in Subsections 6.5.1 and 9.4.1.

12.3.3.2 Design Description

In the following sections, the design features of the various ventilation systems that achieve the radiation control design objectives are discussed. For all areas potentially having airborne radioactivity, the ventilation systems are designed such that during normal and maintenance operations, airflow between areas is always from an area of low potential contamination to an area of higher potential contamination.

12.3.3.2.1 Control Room Ventilation

The control building atmosphere is maintained at a slightly positive pressure (up to 0.5 in. wg) at all times, except if exhausting or isolation are required, in order to prevent infiltration of contaminants. Fresh air is taken in via a dual inlet system, which has both intake structures on the roof of the building. The inlets are arranged with respect to the SGTS exhaust stack such that at least one of the intakes is free of contamination after a LOCA. Both inlets, however, can be submerged in contaminated air from a LOCA, but the calculated dose in the control room from such an eventuality is still below the limit of Criterion 19 of 10CFR50, Appendix A.

Outside air coming into the intakes is normally filtered by a particulate filter. If a high radiation level in the air is detected by the airborne radiation monitoring system, flow is automatically diverted to another filter train (an outdoor air cleanup unit) that has:

(1) a particular filter;

(2) a HEPA filter;

(3) a charcoal filter; and

(4) another HEPA filter.

Two redundant, divisionally separated radiation monitors and filter trains are provided. (See Subsection 9.4.1 for detailed description of the design.) Conservative calculations show that the filters keep the dose in the control room from a LOCA below the limits of Criterion 19 of 10CFR50, Appendix A.

The outdoor cleanup units are located in individual, closed rooms that help prevent the spread of any radiation during maintenance. Adequate space is provided for maintenance activities. The particulate and HEPA filters can be bagged when being removed from the unit. Before removing the charcoal, any radioactivity is allowed to decay to minimal levels, and is then removed through a connection in the bottom of the filter by a pneumatic transfer system. Air used in the transfer system goes through a HEPA filter before being exhausted. Dust masks can worn during maintenance activities, if desired.

12.3.3.2.2 Drywell

Access into the drywell is not permitted during normal operation. The ventilation system inside merely circulates, without filtering, the air. The only airflow out of the drywell into accessible areas is minor leakage through the wall.

During maintenance, the drywell air is purged before access is allowed.

12.3.3.2.3 Reactor Building

The reactor building HVAC system is divided into three zones, which are separated by leaktight, physical barriers. The zones include:

(1) secondary containment (this area contains equipment that is a potential source of radioactivity and if a leak occurs, the other accessible areas of the building are not contaminated);

(2) electrical equipment area, cable tunnels, cable spreading rooms, remote control panel area, diesel generator rooms, reactor internal pump panel rooms, and the heating and ventilating equipment rooms; and

(3) steam tunnel (this room also contains a potential source of radioactive material leakage).

Air pressure in the rooms in Zone 1 is maintained slightly below outside atmospheric pressure by a fresh air supply and exhaust system. The supply air is filtered by a particulate filter. The exhaust stream is monitored for radioactivity, and if a high activity level is detected, the exhaust stream is diverted to the SGTS.

Normally, exhaust air is drawn from the corridor and various rooms. The exhaust duct has two isolation valves in series and a radiation monitor. The valves isolate the system if high airborne radioactivity is detected by the radiation monitor.

Zone 2 of the reactor building is maintained at a positive pressure during normal operation.

For a description of the reactor building HVAC system, see Subsection 9.4.5.

12.3.3.2.4 Radwaste Building

The radwaste building is divided into two zones for ventilation purposes. The control room is one zone, and the remainder of the building is the other zone. The air pressure in the first zone is maintained slightly above atmospheric, while the air pressure in the second zone is maintained slightly below atmospheric. Air in the second zone is drawn from outside the building and distributed to various work areas within the building. Air flows from the work areas and is then discharged via the reactor building stack. An alarm sounds in the control room if the exhaust fan fails. The exhaust flow is monitored for radioactivity, and if a high activity level is detected, the potentially radioactive cells are automatically isolated, but airflow through the work areas continues.

If the exhaust flow high-radiation alarm continues to annunciate after the tank and pump rooms are isolated, the work area branch exhaust ducts are selectively manually isolated to locate the involved building area. Should this technique fail, because the airborne radiation has spread throughout the building, the control room air conditioning continues, but the air con-

ditioning for the balance of the building is shut down.

The work area's exhaust air is drawn through a filter unit consisting of a particulate filter, a HEPA filter, a charcoal filter, and then another HEPA filter, before being discharged to the reactor building stack. The air is monitored for radioactivity, and if a high level is detected, supply and exhaust is terminated, and the SGTS is started.

Maintenance provisions for the filters are similar to those for the control building HVAC system.

See Subsection 9.4.6 for a detailed discussion of the radwaste building HVAC system.

12.3.4 Area Radiation and Airborne Radioactivity Monitors

This section defines and describes the area radiation system that monitors the gamma radiation levels throughout the plant except within the containment. The gamma radiation levels within the containment (drywell and suppression chamber) are monitored continuously by the containment atmospheric monitoring system (CAMS) as described in Subsection 7.6.2. Four gamma sensitive ion chambers (two per divisions 1 & 2) are provided by CAMS to monitor for airborne radioactivity up to 10^7 rads per/hr. Those four sensors are located at the penetrations listed in Table 6.2-8. The area radiation monitoring system is classified as non-safety.

12.3.4.1 System Objectives

The purpose of the area radiation monitoring system is to warn plant personnel of excessive gamma ray levels in service areas including the areas where nuclear fuel is stored or handled, to record and indicate the monitored gamma radiation levels in the control room at selected locations within the various plant buildings, and to provide audible local alarms at key locations where abnormal radiation levels could endanger plant personnel.

12.3.4.2 System Description

The area radiation monitoring system consists of gamma sensitive detectors, associated digital radiation monitors, auxiliary units, local audible warning devices and multipoint recorders. The detector signals are digitized and optically multiplexed for transmission to the radiation monitors. Each monitor has two adjustable trip circuits for alarm initiation, one high radiation level trip and one downscale trip. The downscale trip circuit operates on loss of power or when gross equipment failure occurs. Auxiliary units are provided in local areas for radiation indication and for initiating the sonic alarms on abnormal levels. The electronics are powered from the non-1E vital 120 Vac source while the recorders are powered from the 120 Vac instrument bus.

12.3.4.3 System Design

The area radiation monitoring detectors provided in each plant building are listed in Tables 12.3-3 through 12.3-7 along with area location maps shown in Figures 12.3-56 through 12.3-73. Also, these tables specify the sensitivity range of each channel as designated below along with requirements for local area alarms.

The channel sensitivity covers the following ranges:

a) Range 10^{-2} to 10^2 mR/hr - H (High Sensitivity)

b) Range 10^{-1} to 10^3 mR/hr - M (Medium Sensitivity)

c) Range 1 to 10^4 mR/hr - L (Low Sensitivity)

d) Range 10^2 to 10^6 mR/hr - LL (Low Low Sensitivity)

e) Range 10^{-1} to 10^4 mR/hr - VL (Very Low Sensitivity)

There are two radiation detectors that are located in the fuel storage and handling area, one is positioned to monitor the radiation near the fuel pool and the other is placed in the fuel handling area to monitor the radiation that may result from accidental fuel handling. Criticality detection monitors for this area are not needed to satisfy the criticality accident requirements of 10CFR70.24, because the ABWR design utilizes specialized high density fuel storage racks that preclude the possibility of criticality accident under normal and abnormal conditions. The new fuel bundles are stored in racks that are placed at the bottom of the fuel storage pool. A full array of loaded fuel storage racks are designed to be subcritical by at least 5% delta k. Refer to Sections 9.1 and 9.2 for details.

The detectors and radiation monitors are responsive to gamma radiation over an energy range of 80 keV 7 MeV. The energy dependence

will not exceed 20% of point from 100eV to 3 MeV. The overall system design accuracy is within 9.5% of equivalent linear full scale recorder output for any decade.

The trip/alarm setpoints will be established in the field following equipment installation at the site. The exact settings will be based on sensor location, back ground radiation levels, expected radiation levels, and low occupational radiation exposures.

Each channel is calibrated based on a pseudo input signal to confirm accurate monitor response. The detectors are calibrated using standardized traceable radioactive source in order to establish the linearity and sensitivity of the channel for subsequent calibration. The area radiation monitoring system is designed to accommodate periodic surveillance testing.

The area radiation monitoring instrumentation is designed and properly located to provide early detection and warning for personnel protection to insure that occupational radiation exposures will be as low as is reasonably achieved (ALARA) in accordance with guidelines stipulated in Reg Guide 8.2 and 8.8.

The area radiation monitoring system includes instrumentation provided to assess the radiation conditions in crucial areas in the reactor building (the RHR equipment areas) where access may be required to service the safety related equipment during post LOCA per Reg Guide 1.97.

12.3.5 Post-Accident Access Requirements

The locations requiring access to mitigate the consequences of an accident during the 100-day post-accident period are the control room, the technical support center, the remote shutdown panel, the primary containment sample station (post accident sample system), the health physics facility (counting room), and the nitrogen gas supply bottles. Each area has low post LOCA radiation levels. The dose evaluations in Subsection 15.6.5 are within regulatory guidelines.

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Access to vital areas through out the reactor building/control building/turbine building complex is controlled via the service building. Entrance to the service building and access to the other areas are controlled via double locked secured entry ways. Access to the reactor building is via two specific routes, one for clean access and the second for controlled access. During a event such as a design basis accident, the service building/control building are maintained under filtered HVAC at a positive pressure with respect to the environment. Air infiltration is minimized by positive flow via double entry ways. Therefore, radiation exposure is limited to gamma shine from the reactor building, turbine building, main steam line access corridor, and skyline. This shine is minimized by locating highly populated areas below ground.

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During a design basis accident event, access to remote shutdown panel, nitrogen bottles, and the PASS and monitor systems is controlled from the service building via the controlled access way. These corridors are not maintained under filtered positive pressure so that personal protection equipment (radiation protection suits, breathing gear, etc.) will be required in the access corridor. Primary contamination would occur from leakage through the PASS system and air infiltration from the environment. Both pathways are considered minimal and minor contamination under even the most adverse conditions is expected.

The reactor building vital areas are all located off the controlled access way and contamination is limited to air infiltration from the environment and penetration leakage from the PASS system. Sources of radiation in each area are limited to gamma shine from the reactor building and potential leakage from monitor system such as the PASS. These sources are considered minimal including the stack monitor room which contains only instrumentation with their associated penetrations for monitoring stack effluent.

12.3.6 Post-Accident Radiation Zone Maps

The post-accident radiation zone maps for the areas in the reactor building are presented in Figures 12.3-25 through 12.3-36. The zone maps represent the maximum gamma dose rates that exist in these areas during the post-accident period. These dose rates do not include the airborne contribution in the reactor building.

Post-accident zone maps of the control building and turbine building are presented in Figures 12.3-54 and 55 respectively. The zone maps are designed to reflect the criteria established in Subsection 3.1.2.2.10.

12.3.7 Deleted

12.3.8 References

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2. J. H. Hubbell, *Photon Cross Sections, Attenuation Coefficients, and Energy Absorption Coefficients from 10 KeV to 100 GeV*, NSRDS-NBS20, U.S. Department of Commerce, August 1969.

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6. M.A. Capo, *Polynomial Approximation of Gamma Ray Buildup Factors for a Point Isotropic Source*, APEX-510, November 1958.

7. *Reactor Physics Constants*, Second Edition, ANL-5800, U.S. Atomic Energy Commission, July 1963.

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9. PDS-31 Cross Section Library, Oak Ridge National Laboratory.

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Table 12.3-1

COMPUTER CODES USED IN SHIELDING DESIGN CALCULATIONS

Computer Code	Description
QADF	A multigroup, multiregion, point kernal, gamma ray code for calculating the flux and dose rate at discrete locations within a complex source-geometry configuration.
GGG	A multigroup, multiregion, point kernal code for calculating the contribution due to gamma ray scattering in a hetrogeneous three-dimensional space
DOT.4	A discrete ordinates, two-dimersional transport code. Multigroup, multiregion neutron or gamma transport

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12.4 DOSE ASSESSMENT

Dose assessment is an important part of determining and projecting that the plant design and proposed methods of operation assures that occupational radiation exposure will be as low as reasonably achievable. Dose assessment depends upon estimates of occupancy, dose rates in various occupied areas, number of personnel involved in reactor operations and surveillance, routine maintenance, waste processing, refueling, in-service inspection, and special maintenance.

The goal is to reduce the exposure associated with each phase of plant operation and maintenance to the minimum level consistent with practical considerations for accomplishing each task. To achieve this goal, the ABWR design includes numerous significant design improvements to reduce occupational exposures from past experience. The design improvements include the elimination of recirculation piping and valves, improved water chemistry and low cobalt alloys at the cooling water boundary, reduced equipment maintenance and improved access, RHR discharge to the feedwater piping, overhaul handling and refueling devices, multiple main steam line plugs, automatic MSIV seat lapping system and reactor vessel stud tensioner. In assessing the collective occupational dose, each potentially significant dose-causing activity was evaluated. Values referred to as typical BWR operations are taken from references 1 through 4 which are a compendium of maintenance and work tasks for BWR-6, GESSAR.

12.4.1 Drywell Dose

The following provides the basis by which the drywell dose estimates for occupational exposure were made.

(1) The main steam isolation valves are located in the upper drywell area (4 valves) and in the reactor building outboard of the primary containment isolation wall (4 valves). These valves require periodic testing and maintenance to insure proper action and leak tightness. Typical values for BWR's for maintenance of these valves is 4,000 hours of drywell and 5,000 hours of reactor building work in effective radiation fields of 13.5 mRem/hr and 3.6 mRem/hr respectively. The ABWR design incorporates three specific features to reduce occupational exposure in the MSIV maintenance area: (1) improved water chemistry with lower overall contamination rates, (2) improved maintenance procedures with some procedures automated, and (3) reduced radiation fields, primarily due to the absence of the recirculation piping. Each area is discussed below.

Beginning in the early 1980's the BWR Owner's Group began an extensive study of the causes for failure of MSIV's to meet the technical leakage specification limits and extensive person-hours required to maintain these valves. As a result of these studies, the ABWR will use the latest technology for valve maintenance including mechanical aids for valve disassembly and assembly, automated lapping devices, and slightly relaxed leakage specifications to delete unnecessary maintenance. As a result of these aids, it is estimated that overall maintenance hours will be reduced by 50-60 percent.

Early studies on dose rates during MSIV maintenance showed increases in dose rate directly proportional to recirculation line activity. The ABWR has deleted the recirculation lines entirely thereby removing the singly most significant source of radiation in the drywell. The second most significant dose for MSIV operations will be the deposited and suspended activity in the feedwater lines. The deposited activity in the feedwater lines is expected to be lower than typical BWRs owing to an enhanced condensate system with full clean up of all condensate water, a 2% reactor water clean up system, and titanium condenser tubes. Additionally, the ABWR is designed to limit the use of cobalt bearing materials on moving components which have historically been identified as major sources of in water contamination. Overall the feedwater line radiation is expected to be a factor of three lower than current BWRs. Because of these factors, it is expected that the effective dose rate in the drywell will be 1.8mRem/hr and 1.3mRem/hr in the steam tunnel outboard of the primary containment.

(2) Drywell valve and pump maintenance other than the MSIVs consists primarily of maintaining the safety relief valves (SRVs) which for the most part consist of minor maintenance or removal of valves to a maintenance facility. Overall typical values for a BWR for these tasks are 1,450 person-hours per year in an effective radiation field of

17mRem/hr. In the ABWR, the primary source of radiation exposure, the recirculation lines and pumps, have been removed. Overall the reduction in drywell dose level is for these types of maintenance is expected to be a factor of two or 9mRem/hr. Overhead tracks and in place removal equipment is provided in the ABWR for an estimated person-power reduction to 1,150 person-hour per year broken down into 200 person-hours for 18 SRV maintenance at 6mRem/hr, 200 person-hours per year to pull and replace 3 RIPs with one heat exchanger at 20mRem/hr, and the remainder on miscellaneous valves at 4.5mRem/hr.

(3) Control rod drive maintenance is significantly reduced in the ABWR with the introduction of fine motion control rod drives (FMCRD). Based upon European experience, two FMCRDs will be replaced and repaired per outage along with twenty motors. Estimated work will consist of 64 person-hours under vessel preparation, 40 person-hours FMCRD removal and reinstallation, 200 person hours motor removal and installation, and 64 person-hours cleanup. Typical under vessel effective dose rates are 17mRem/hr but because of the removal of the recirculation pumps and lines has been reduced to 6.5mRem/hr.

(4) The LPRM/TIP system assumes the servicing of two sensors per year and is based upon a total of 200 person-hours per year at an effective dose rate of 50mRem/hr which is typical for BWR operations.

(5) Inservice inspection consists of primarily NDE examination of vessel and piping systems and welds. Typical BWR values are 2400 person-hours per year at 12mRem/hr effective exposure rate. ABWR inservice inspection is estimated based upon the following:

Elimination of recirculation lines and pumps with the following savings:

Elimination of 14 nozzle inspections at 2 per year, saving 360 person hours

Elimination of shield penetration and shield plug removal saving 240 person hours per year

Reduction on weld inspection on recirculation lines estimated at 240 person hour per year

Reduction in drywell dose by 50% with the provision that the feedwater line dose is more than half the recirculation line dose and general drywell dose level and therefore removal of recirculation line inspection is estimated to be weighted at twice the general drywell dose rate.

Overall it is estimated that by use of automated turtles for inspection person-hour expended in ISI will be reduced by a factor of two.

The ABWR uses a forged ring pressure vessel in comparison to older plate welded vessels reducing the total vessel weld length inspection by 30% and the total weld inspection by 10%.

The ABWR design incorporates specific access into inspection areas past insulation areas with an estimated savings of 120 person-hours.

Overall person-hours reduction is 1,200 person-hours at approximately half the typical effective dose rate or 5.5mRem/hr.

(6) Other drywell work includes items such as minor valve maintenance, instrumentation work, and all other drywell work. Typical BWR work in this area estimates 5,500 person-hours per year at 17 mRem/hr. Overall reduction in this effort due to ABWR design improvements are:

Significant savings in total hours are estimated due to removal of the recirculation lines with miscellaneous recirculation line work such as line snubbers, fewer drywell cooling units, and less assembly/disassembly work on insulation due to the use of automated units. Overall it is estimated that 2,000 person hours savings can be made.

Overall reduction in the drywell radiation due to removal of the recirculation system results in the reduction of the overall upper drywell dose rate to 1.8mRem/hr and the lower drywell dose rate to 5.5mRem/hr since the components involved such as drywell coolers typically do not carry radioactive inventory. Assuming that of the remaining 3,500 person-hour, 2,000 is upper drywell work and 1,500 is lower drywell work at their respective effective dose rates.

12.4.2 Reactor Building Dose

The following provides the basis by which the reactor building dose estimates for occupational exposure were made.

(1) Vessel access and reassembly typically requires 4500 person-hours of work at an effective dose rate of 3 mr/hr. The ABWR work will involve the use of a stud tensioner for a 96 bolt top head. The projected time to remove 96 bolts with this equipment is between 600 to 1200 person-hours. Due to the larger ABWR vessel and expected reduced water contamination with the improved clean up system, the estimated projected effective dose rate is 1.5 mRem/hr.

(2) ABWR refuelling is accomplished via an automated refueiling bridge. All operations for refuelling are accomplished from an enclosed automation center off the refuelling floor. Time for refuelling is reduced from a typical 4,400 person-hours down to 2,000 person and from an effective dose rate of 2.5 mRem/hr to less than 0.2mRem/hr.

(3) RHR/CUW maintenance work consists of inspections for two pumps per year in each system. In the RHR system this consumes 150 person-hours per year at an effective dose rate of 40mRem/hr. In the CUW system this typically uses 1400 person-hours per year at an effective dose rate of 14mRem/hr. ABWR will use canned pumps for both system with an estimated reduction in maintenance to 100 person-hours per pump. With improved water chemistry and overall reductions in reactor water concentrations due to the two percent cleanup system the effective dose rate is estimated at twenty percent of the typical value for these system.

(4) FMCRD rebuilding estimates are taken from similar work done in Europe since no significant U.S. data exists to date. Two drives will be rebuilt at an effective dose rate of 4.5 mRem/hr and 30-60 hours per drive.

(5) Instrumentation work typically requires 1,000 person-hours of work per year at an effective dose rate of 5.0 mRem/hr. ABWR should take about the same effort in instrumentation, however because of the increased emphasis and improved water chemistry systems, should reduce the effective dose rate to two-thirds the typical value or 3.0mRem/hr.

(6) All other work in the reactor building typically takes 7,400 person-hours per year at an effective dose rate of 2.8mRem/hr. This work includes all valve work, RIP rebuild work, minor maintenance, and CRD hydraulic line work. The major task in this area is the hydraulic control units which require 5,000 person-hours per year at an effective dose rate of 3.3mRem/hr. With the use of the FMCRD units, an additional savings of 2,000 person-hours is anticipated. In addition, the ABWR reactor building has been designed to provide for ease of maintenance with overhead lifts, coordinated hatch ways and ample space to maintain in place equipment. In addition, with the exception of one tank and the pressure vessel, all the equipment in the reactor building is removable with those pieces which can be expected to be moved being palatalized. Because of these factors, an overall reduction in work of 1,000 person-hours is estimated. Because of the improved water chemistry the overall effective dose rate is anticipated at one-half the typical BWR dose rate.

12.4.3 Radwaste Building Dose

Radwaste building work consists of pump and valve maintenance, shipment handling, radwaste management, and general clean up activity. Typically, 6,700 hours are expended per year at an effective dose rate of 5.5mRem/hr. The ABWR radwaste building is designed along the same lines as newer radwaste facilities overseas. The building incorporates enhanced remote control and shielding for handling of resin materials which is expected to reduce overall maintenance by 1500 to 2000 hour per year at significantly reduced dose levels. In addition, radwaste pumps for ABWR are expected to utilize air driven, rack mounted pumps. Such pumps which are designed to handle slurries have been proven to show much longer life times between maintenance and being basically a very small portable pump, can be readily replaced. Replaced pumps are then subject to intense chemical decontamination prior to maintenance and repair. Overseas utilities have reported occupational exposures typically less than 1 person-rem per year using this design. For ABWR assuming 2,000 hours reduction in maintenance due to remote handling and an additional 500 hours reduction for pump replacement, 4,200 hours per year are estimated with

reduced effective dose rates of 2.5mRem/hr owing primarily to remotting those jobs involving high radiation exposure.

12.4.4 Turbine Building Dose

(1) Typical BWR valve maintenance in the turbine building uses 1,150 hours per year at an effective dose rate of 9.5mRem/hr. The valve maintenance requirements for ABWR do not vary significantly over current plants, therefore the total hours for this type of work is assumed as approximately the same excepting minor adjustments for improved valves, maintenance jigs, and automated devices will lower the estimated maintenance time to 1,000 hours. The effective dose rate of 9.5 mRem/hr is estimated at more than one half this value due to basically improvements in BWR fuel over the generation of fuel from which this data was taken bringing the effective dose rate down to 3.9mRem/hr. In addition, beta shielding is recommended for work on valving where possible which it is estimated will reduce the overall effective dose rate by an additional 10% to 3.5mRem/hr.

(2) In a similar fashion the turbine maintenance work typically requires 18,500 hours of work at an effective dose rate of 0.3mRem/hr. With additional operational improvements in automating turbine maintenance, overall work is estimated to be reduced to 15,500 hours. The effective dose rate for the turbine is not expected to be as sensitive to fuel performance as will the turbines but is estimated to reflect a decrease in dose to 0.2mRem/hr for turbine overhaul work.

(3) Work on the turbine hall condensate system typically requires 2,000 hours per year at an effective dose rate of 7.5mrem/hr. The condensate system in ABWR uses hollow-fiber filled filters which require half the maintenance of a typical system. In addition, with the plant incorporating Fe control in the feedwater system and a significant reduction in cobalt bearing materials, the overall effective dose rate is estimated at half the above value.

(4) Other work in the turbine building typically takes 13,140 hours per year at an effective dose rate of 0.1mRem/hr. Only minor changes can be assumed with ABWR with some remote operations and slight reductions in operating exposures. For the ABWR it is estimated that a 10% reduction can be realized with improving technology with no significant change in dose rate.

12.4.5 Work at Power

Work at power typically requires 5,000 hours per year at an effective dose rate of 6.6mRem/hr for the BWR. This category covers literally all aspects of plant maintenance performed during normal operations from health physics coverage to surveillance, to minor equipment adjustment, and minor equipment repair. Overall the ABWR has been designed with more automated and remotted equipment. It is expected that items of routine monitoring will be performed by camera or additional instrumentation. Most equipment in ABWR is palatalized which permits quick and easy replacement and removal for decontamination and repair. Therefore a reduction in actual hours need at power is estimate at 1,000 hours less than the typical value. In the area of effective dose rate, the ABWR is expected to have significantly lower general radiation levels over current plants owing to more stringent water chemistry controls, a full flow condensate flow system, a 2% clean up water program, titanium condenser tubes, Fe feedwater control, and low cobalt usage. In addition, the ABWR is the most compartmentalized BWR design which (1) perm[illegible] better shielding in specific work areas, and (2) l[illegible]s collateral radiation contamination. Overall then it is estimated that the effective dose rate for work at power will be slightly over two thirds the typical rate or 4.0mRem/hr.

12.4.6 References

1. Knecht, P.D., *BWR/6 Drywell and Containment Maintenance and Testing Access Time Estimates*, GE Report NEDE-23819, May 1978.

2. Knecht, P.D., *Maintenance Access Time Estimates, BWR/6 Radwaste Building*, GE Report NEDE-23996-2, May 1979.

3. Knecht, P.D., *Maintenance Access Time Estimates, BWR/6 Auxiliary and Fuel Buildings*, GE Report NEDE-23996-1, May 1979.

4. *Study of Advanced BWR Features, Plant Definition/Feasibility Results*, Volume III, Appendix Part G, GE NEDE-24679, Oct 1979.

Table 12.4-1

PROJECTED ANNUAL RADIATION EXPOSURE

Operation Task	SSAR Section	Hours per year	mRem/hr	Person-Rem/yr
Drywell				
MSIV	12.4.1(1)	4,200	1.5	6.3
SRV,RIP,etc	12.4.1(2)	1,150	7.5	8.6
FMCRD	12.4.1(3)	370	6.5	2.4
LPRM/TIP	12.4.1(4)	200	50.0	10.0
ISI	12.4.1(5)	1,200	5.5	6.6
Other	12.4.1(6)	3,500	3.5	12.3
Total		10,620		46.2
Reactor Building				
Vessel	12.4.2(1)	1,200	1.5	1.8
Refueling	12.4.2(2)	2,000	0.2	0.4
RHR/CUW	12.4.2(3)	400	8.0	3.2
FMCRD	12.4.2(4)	120	4.5	0.5
Instrument	12.4.2(5)	1,000	3.0	3.0
Other	12.4.2(6)	4,400	1.5	6.6
Total		9,120		15.5
Radwaste Building	12.4.3	4,200	2.5	10.5
Turbine Building				
Valve Maint	12.4.4(1)	1,000	3.5	3.5
Turbine Ovrhl	12.4.4(2)	15,500	0.2	3.1
Condensate	12.4.4(3)	1,000	3.5	3.5
Other	12.4.4(4)	11,800	0.1	1.2
Total		29,300		11.3
Work at Power	12.4.5	4,000	4.0	16.0
Totals		48,120		99.5

Specific testing to be performed and the applicable acceptance criteria for each preoperational test are in accordance with the detailed system specifications and equipment specifications for equipment in those systems. The tests demonstrate that the installed equipment and systems perform within the limits of these specifications.

The preoperational tests anticipated for the ABWR Standard Plant are listed and described in the following paragraphs. Testing of systems outside the scope of the ABWR Standard Plant, but that may have related design and therefore testing requirements, are discussed in Subsection 14.2.13, along with other interface requirements related to the initial test program.

14.2.12.1.1 Nuclear Boiler System Preoperational Test

(1) Purpose

To verify that all pumps, valves, actuators, instrumentation, trip logic, alarms, annunciators, and indications associated with the nuclear boiler system function as specified.

(2) Prerequisites

The construction tests have been successfully completed and the SCG has reviewed the test procedure and has approved the initiation of testing. All required interfacing systems shall be available, as needed, to support the specified testing and the appropriate system configurations.

(3) General Test Methods and Acceptance Criteria

Performance should be observed and recorded during a series of individual component and integrated system tests to demonstrate the following:

(a) verification that all sensing devices respond to actual process variables and provide alarms and trips at specified values;

(b) proper operation of system instrumentation and any associated logic, including that of the automatic depressurization system (ADS);

(c) proper operation of MSIVs and main steamline drain valves, including verification of closure time in the isolation mode, and test mode, if applicable;

(d) verification of SRV and MSIV accumulator capacity;

(e) proper operation of SRV air piston actuators and discharge line vacuum breakers;

(f) verification of the acceptable leak tightness and overall integrity of the reactor coolant pressure boundary via the leakage rate and/or hydrstatic testing as described in Section 5.2.4.6.1 and 5.2.4.6.2 respectively; and

(g) proper system instrumentation and equipment operation while powered from primary and alternate sources, including transfers, and in degraded modes for which the system and/or components are expected to remain operational.

Other checks should be performed, as appropriate, to demonstrate that design requirements, such as those for sizing or installation, are met via as built calculations, visual inspections, review of qualification documentation or other methods. For instance, SRV setpoints and capacities should be verified from certification or bench tests to be consistent with applicable requirements. Additionally, proper installation and setting of supports and restraints for SRV discharge piping will be verified as part of the testing described in 14.2.12.1.51.

14.2.12.1.2 Reactor Recirculation System Preoperational Test

(1) Purpose

To verify the proper operation of the reactor recirculation system at conditions approaching rated volumetric flow, including the reactor internal pumps (RIPs) and motors, and the equipment associated with the motor cooling, seal purge, and inflatable shaft seal subsystems.

(2) Prerequisites

The construction tests have been successfully completed and the SCG has reviewed the test procedure and has approved the initiation of testing. Cooling water from the reactor building cooling water system and seal purge flow from the CRD hydraulic system shall be available. The recirculation flow control system should be sufficiently tested

$$E = W_G H_G + \int_0^{H_B} \frac{W_B}{H_B} y \, dy =$$

$$W_G H_G + \frac{1}{2} W_B H_B$$

$$+ (350 \text{ lb}) \left(\frac{160}{12}\right) + \frac{1}{2} (617) \left(\frac{160}{12}\right) =$$

8780 ft-lb

As before, the energy was considered to be absorbed equally by the falling assembly and the impacted assemblies and the fraction available for clad deformation was 0.519. The energy available to deform clad in the impacted assemblies was:

$$E_c = (0.5)(8780 \text{ ft-lb})(0.519) = 2278 \text{ ft-lb}$$

and the number of failures in the impacted assemblies was

$$N_F = \frac{(2278 \text{ ft-lb})}{(250 \text{ ft-lb})} = 9 \text{ rods}$$

Since the rods in the dropped assembly were considered to have failed in the initial impact, the total failed rods in both impacts in 106 + 9 = 115.

15.7.4.4 (Not Used)

15.7.4.5 Radiological Consequences

Radiological analysis is based on conservative assumptions considered to be acceptable to the NRC for the purpose of determining adequacy of the plant design to meet 10CFR100 guidelines. The analysis is referred to as the "Design Basis Analysis".

The fission product inventory in the fuel rods assumed to be damaged in based on 1000 days of continuous operation at 4055 MWt. A 24-hr period for decay from the above power condition is assumed because it is not expected that fuel handling can begin within 24 hr following initiation of reactor shutdown. Figure 15.7-1 shows the leakage flow path for this accident.

15.7.4.5.1 Design Basis Analysis

The design basis analysis is based on Regulatory Guide 1.25. The specific models, assumptions and the program used for computer evaluations are described in Reference 1. Specific values or parameters used in the evaluation are presented in Table 15.7-8.

15.7.4.5.1.1 Fission Product Release from Fuel

Per the conditions in Regulatory Guide 1.25, The following conditions are assumed applicable for this event:

(1) Power Level - 4005 MWt for 3 years

(2) Plenum Activity - 10% of the radioactivity for iodine and noble gases except Kr-85 and 30% for Kr-85.

(3) Fission Product Peaking Factor - 1.5 for those rods damaged.

(4) Activity Released to Reactor Building - 10% of the noble gas activity and 0.1% for the iodine activity.

Based on the above conditions, the activity released to the reactor building is presented in Table 15.7-9.

15.7.4.5.1.2 Fission Product Transport to the Environment

Also, per the conditions of Regulatory Guide 1.25, it is assumed that the airborne activity of the reactor building (Table 15.7-9) is released to the environment over a 2-hr period via a 99% iodine efficient SGTS. The total activity released to the environment is presented in Table 15.7-10.

15.7.4.5.1.3 Results

The calculated exposures for the design basis analysis are presented in Table 15.7-11 and are within the guidelines of 10CFR100

15.7.5 Spent Fuel Cask Drop Accident

15.7.5.1 Identification of Cause

Due to the redundant nature of the crane, the cask drop accident is not believed to be a credible accident. However, the accident is assumed to occur as a consequence of an unspecified failure of the cask lifting mechanism, thereby allowing the cask to fall from the level of the refueling floor to ground level through the refueling floor maintenance hatch.

15.7.5.2 Radiological Analysis

The largest size BWR fuel cask is conservatively assumed to be dropped approximately 94 feet from the refueling floor level to ground level on transport from the decontamination pit out of the reactor building. It is conservatively assumed that all fuel rods are damaged and the fission gases in the fuel rod gap space are released to the reactor building and then to the environment over a two hour period. Table 15.7-12 provides the assumptions for this analysis and Table 15.7-13 radiological consequences. As can be seen from Table 15.7-13, the radiological releases are within guidelines.

15.7.6 References

1. D. Nguyen, et al., *Radiological Accident Evaluation - The CONAC03 Code*, December 1981 (NEDO-21143-1).

2. N.R. Horton, W.A. Williams, and K.W. Holtzclaw, *Analytical Methods for Evaluating the Radiological Aspects of General Electric Boiling Water Reactors*, March 1976 (APED-5756).

3. R.A. Head, *BWR Radioactive Waste Treatment System*, August 1976 (NEDO-21059).

15B.2 CONTROL ROD DRIVE SYSTEM

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15B.4 ESSENTIAL MULTIPLEXING SYSTEM

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SECTION 16.9

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18B.1 Introduction

Appendix 18B presents the differences and bases for the differences between the NRC approved BWROG EPG Revision 4 document[1] and the ABWR EPGs. For a given difference identified, the ABWR EPG step, BWROG EPG Revision 4 step, and the basis for the difference is given. The numbers used for the ABWR EPG steps correspond to those of the ABWR EPG Guidelines given in Appendix 18A.

[1] NRC letter, A. C. Thadani to D. Grace, Safety Evaluation of *BWR Owner's Group-Emergency Procedure Guidelines, Revision 4,* NEDO-31331, March 1987, dated September 12,1988.

18C.0 Introduction

This Appendix contains characterization of one operator interface system which has been designed to meet the design requirements as specified in section 18.4. The key features of the design are discussed. The design characterized in this Appendix does not necessarily represent the final design. The final design must be established based upon the requirements of Section 18.5, operator interface design implementation requirements, which is the responsibility of the applicant referencing the ABWR design.

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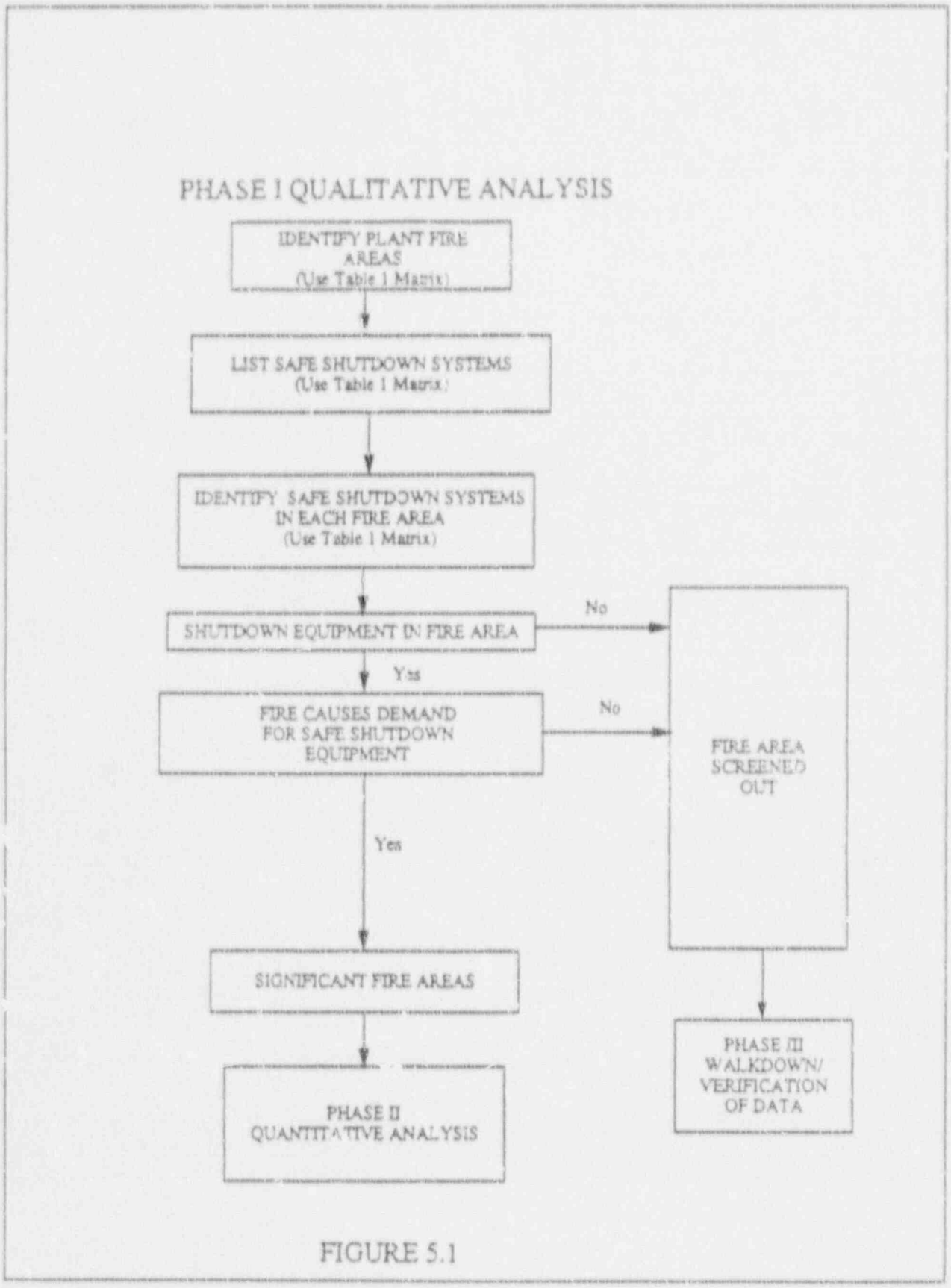

Figure 19M.3-1 PHASE I QUALITATIVE ANALYSIS FLOW CHART

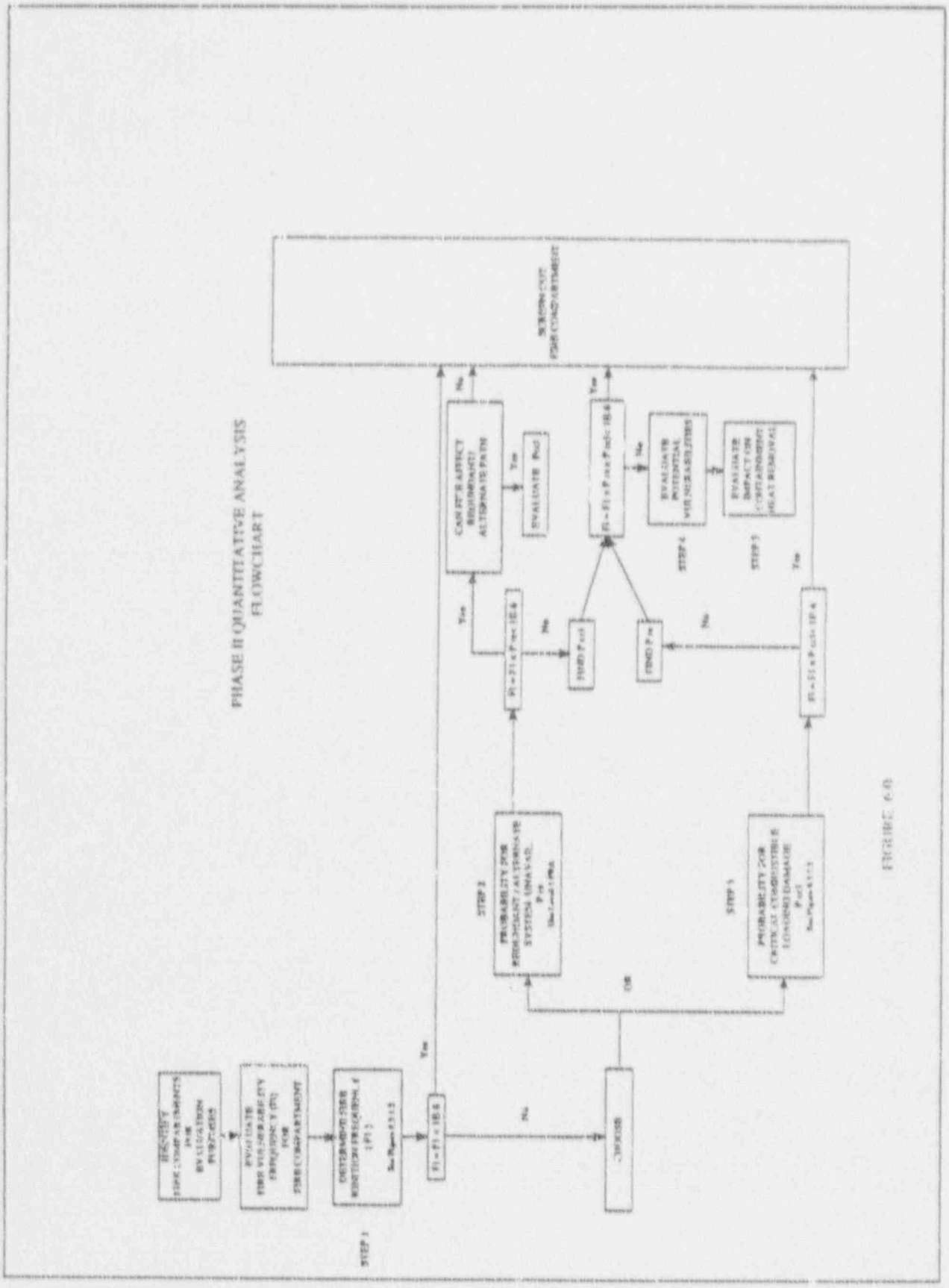

Figure 19M.3-2 PHASE II QUALITATIVE ANALYSIS FLOW CHART

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19M.7 INTERFACE

Section 19M.3 requires local manual control of RCIC as one means of mitigation in case of a control room fire. It is a requirement that a procedure for local operation of the RCIC be provided by the utility.

19M.8 REFERENCES

1. Fire Vulnerability Evaluation Methodology, FIVE, Plant Screening Guide, Electric Power Research Insitute, Preliminary Draft.

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		430.24	6.2	20.3.2	2
		430.25	6.2	20.3.2	2
		430.26	6.2	20.3.2	2
		430.27	6.2	20.3.2	2
		430.28	6.2	20.3.2	2
		430.29	6.2.3	20.3.2	2
		430.30	6.2	20.3.2	2
		430.31	6.2	20.3.2	2
		430.32	6.2	20.3.2	2
		430.33	6.2	20.3.2	2
		430.34	6.2	20.3.2	2
		430.35	6.2	20.3.2	2
		430.36	6.2	20.3.2	2
		430.37	6.2	20.3.2	2
		430.38	6.2	20.3.2	2
		430.39	6.2.4	20.3.2	2
		430.40	6.2	20.3.2	2
		430.41	6.2	20.3.2	2
		430.42	6.2	20.3.2	2
		430.43	6.2	20.3.2	2
		430.44	6.2	20.3.2	2
		430.45	6.2	20.3.2	2
		430.46	6.2	20.3.2	2
		430.47	6.2.5.3	20.3.2	2
		430.48	6.2.6	20.3.2	2
		430.49	6.2.6	20.3.2	2
		430.50	6.2.6	20.3.2	2
		430.51	6.2.6	20.3.2	2
		430.52	6.2.6	20.3.2	2
		430.53	6.2.6	20.3.2	2
		430.54	6.4	20.3.2	2
		430.55	6.5.1	20.3.2	2
		430.56	6.5.3	20.3.2	2
		430.57	6.7	20.3.2	2
		430.58	15.7.3	20.3.2	2
		430.59	10.1	20.3.11	10
		430.60	10.2	20.3.11	10
		430.61	10.2.2.2	20.3.11	10
		430.62	10.2	20.3.11	10
		430.63	10.2.2.4	20.3.11	10
		430.64	10.2.2.4	20.3.11	10
		430.65	10.2	20.3.11	10
		430.66	10.2	20.3.11	10
		430.67	10.3.2.1	20.3.11	10
		430.68	10.3.3	20.3.11	10
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NRC* Branch	Review Area	Question Number	SSAR Subsection	Response Subsection	RAI** Letter
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		430.73	10.4.1	20.3.11	10
		430.74	10.4.2	20.3.11	10
		430.75	10.4.2	20.3.11	10
		430.76	10.4.2	20.3.11	10
		430.77	10.4.2	20.3.11	10
		430.78	10.4.2	20.3.11	10
		430.79	10.4.2	20.3.10	10
		430.80	10.4.3	20.3.10	10
		430.81	10.4.3	20.3.10	10
		430.82	10.4.3	20.3.10	10
		430.83	10.4.3	20.3.10	10
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		430.86	10.4.7	20.3.10	10
		430.87	Chap 10	20.3.10	10
		430.88	Chap 10	20.3.10	10
		430.89	10.4.7	20.3.10	10
		430.90	10.4.7	20.3.10	10
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		430.159	11.2	20.3.13	13
		430.160	11.2	20.3.13	13
		430.161	11.2	20.3.13	13
		430.162	11.3	20.3.13	13
		430.163	11.3	20.3.13	13
		430.164	11.3	20.3.13	13
		430.165	11.3	20.3.13	13
		430.166	11.3	20.3.13	13
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		430.168	11.3	20.3.13	13
		430.169	11.4	20.3.13	13
		430.170	11.4	20.3.13	13
		430.171	11.4	20.3.13	13
		430.172	11.4	20.3.13	13
		430.173	11.4	20.3.13	13
		430.174	11.4	20.3.13	13
		430.175	11.4	20.3.13	13
		430.176	11.4	20.3.13	13
		430.177	9.1	20.3.15	15
		430.178	9.1	20.3.15	15
		430.179	9.1	20.3.15	15
		430.180	9.1	20.3.15	15
		430.181	9.1	20.3.15	15

20.2.2 Chapter 2 Questions

241.1

Table 2.01 in the Advanced BWR Standard Plant Safety Analysis Report (SSAR) gives an envelope of ABWR plant site design parameters. This table gives the minimum bearing capacity and the minimum shear wave velocity of the foundation soil. The table also gives the values of SSE and OBE and indicates (a) that the SSE response spectra will be anchored to Regulatory Guide (RG) 1.60, and (b) that the SSE time history will envelope SSE response spectra. The following additional information/clarification should be provided in the SSAR:

a. While the SSE (PGA) of 0.3g anchored to RG 1.60 could, in general, be considered conservative for many sites in the Central and Eastern United States, the SSAR should recognize and reflect the fact that localized exceedances if this value cannot be ruled out categorically and that adequate provisions will be made in the seismic design to consider site-specific geological and seismological factors.

b. The SSAR gives an OBE (PGA) value of 0.10g and states that, "for conservatism, a value of 0.15g is employed to evaluate structural and component responses in Chapter 3." The staff, however, considers the OBE value to be 0.15g as per criterion 2 of 10 CFR 50 Appendix A and paragraph V of 10 CFR 100 Appendix A which require, in part, that for seismic design considerations thew OBE shall be no less than one-half of the SSE.

c. The SSAR should indicate the procedures that would be adopted to evaluate the liquefaction potential at selected soil sites. It is not sufficient to say that the liquefaction potential will be "none at plant site resulting from OBE and SSE."

451.1

What are the bases (including references) for the site envelope of the ABWR design meteorological parameters listed in Table 2.0-1? Are these values intended to reflect the indicated maximum historical values for the contiguous USA? What is the combined winter precipitation load for the addition of the 100-year snow pack and the 48-hour probable maximum precipitation? What is the duration of the design temperature and wind speed values? What gust factors are associated with the extreme winds? Are any other meteorological factors (e.g., blowing dust) considered in the ABWR design?

451.2

Short-term dispersion estimates for accidental atmospheric releases are not provided explicitly in Section 2.4.3. If your X/Q values which are listed in Chapter 15 represent an upper bound for which the ABWR is designed; what is the bases for their selection?

COMPARISON OF BWR/6s
"WEEPING" SRVs

PLANT	"WEEPING" SRVs/TOTAL NO
Clinton	3/16
River Bend	12/19
Grand Gulf	11/20
Grand Gulf (after all valves changed during 1st refueling)	6/20
Perry	18/19

The continuous "weeping" of the SRV has the potential to degrade SRVs and increase the frequency of use of RHR heat exchangers.

How will the ABWR SRVs resolve the generic problem stated above?

(2) Valve and valve operator type and/or design. Include discussion of improvements in the air actuator, especially materials used for components such as diaphragms and seals. Discuss the safety margins and confidence levels associated with the air accumulator design. Discuss the capability of the operator to detect low pressure in the accumulator(s). Provide detailed description of safety and relief mode of operation/function of the SRV.

(3) Specifications. What new provisions have been employed to ensure that valve and valve actuator specifications include design requirements for operation under expected environmental conditions (esp. temperature, humidity, and vibration)?

(4) Testing. Prior to installation, safety/relief valves should be proof tested under environmental conditions and for time period representative of the most severe operating conditions to which they may be subjected.

(5) Quality Assurance. What new programs have been instituted to assure that valves are manufactured to specifications and will operate to specifications.

(6) Valve Operability. Provide a summary of the surveillance program to be used to monitor the performance of the safety/relief valves. Identify the information that will be obtained and how these data will be utilized to improve the operability of the valves.

(7) Valve Inspection and Overhaul. Operating experience has shown that safety/relief valve failure may be caused by exceeding the manufacturer's recommended service life for the internals of the safety/relief valve or air actuator. At what frequency do you intend to visually inspect and overhaul the safety/relief valve? For both safety/relief and ADS modes, what provisions exist to ensure that valve inspection and overhaul are in accordance with the manufacturer's recommendations and that the design service life would not be exceeded for any component of the safety/relief valve?

440.18

Address the following TMI-2 action items related to SRVs.

(a) II.K.3.16
(b) II.B.1
(c) II.D.3
(d) II.K.3.28
(e) II.D.1

440.19

Explain in detail how the spring and relief modes of the SRV works. Are they an difference from the SRVs currently used in operating BWRs?

440.20

What ATWS considerations have you given for sizing SRVs?

440.21

In Subsection 5.2.2.2.2.3 the reclosure pressure setpoint (% of operating setpoint) for both modes are given as 98 and 93. Explain the significence of these numbers.

440.22

In Figure 5.1.3a the SRV solenoid valves are not shown as DC powered as they should be. Note 8 states that "valve motor operators and pilot solenoids are ac operated unless otherwise specified."

440.24

Confirm that SRVs are designed to meet seismic and quality standards consistent with the recommendations of Regulatory Guides 1.26 and 1.29.

440.28

In SSAR Table 1.8-19, it is stated that branch technical position RSB 5-2 is applicable for ABWR. How does the ABWR design comply with BTP RSB 5-2?

440.29

Describe the methods planned for performing hydrostatic tests on ABWR RPV vessel after the initial start-up. Can you perform hydrostatic tests and leak tests without using critical heat?

440.30

In SSAR Chapter 5.4.1.4, it is stated "During various moderately frequent transient, various Reactor Internal Pump (RIP) operating modes will be required such as: Bank of five RIPs runback to 30% speed; trip from current speed conditions; or runback to 30% speed and subsequent trip. These control actions are all produced through control actions of the Recirculation Flow Control System (RFCS)."

RSB 5-1 requires that the suction and discharge valves interfacing with the RCS shall have independent diverse interlocks to prevent the valves from being opened unless the RCS pressure is below the RHR design pressure.

Confirm that the high/low pressure interface with RCS satisfies the requirements of RSB 5-1.

440.72

NRC Bulletin 88-04 dated May 5, 1988, discusses the potential safety related pump loss. The first concern involves the potential for the dead-heading of one or more pumps in safety related systems that have a miniflow line common to two or more pumps or other configurations that do no preclude pump-to-pump interaction during miniflow operation. A second concern is whether or not the installed miniflow capacity is adequate for even a single pump in operation.

In the ABWR design, HPCS pump miniflow lines and test return lines to the suppression pool are routed through the RHR 'c' loop test and minimum flow lines. How does the ABWR design satisfy the concerns given in NRC Bulletin No. 88-04?

440.73

In RHR process diagrams 5.4-11b, RHR heat exchanger removal capacity for different modes is not given. Revise the process diagram to include the heat removal capacity.

440.74

In Figure 5.4-10b, (I-12) flammability system (T-49) is cross-tied to the RHR system. What is the purpose of this cross-tie to the RHR system?

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other systems should be implemented. Challenges to the relief valves should be reduced substantially (by an order of magnitude).

220.16

Generic safety issue 82 "Beyond Design Basis Accidents in Spent Fuel Pools" is concerned with the loss of the pool water which may result in a fire in the pool causing a release of fission products. In the ABWR resolution, it is indicated that the spent fuel pool will be designed to withstand a design basis earthquake without pool drainage, and will be arranged to prevent cask movement over the pool, which will be accomplished through the use of a separate cask loading pit. Was a cask drop in the cask loading pit considered? Since the cask loading pit is adjacent to the spent fuel pool. In addition, it appears that the fuel pool is near the staging area for the reactor vessel head, indicate the effect on the fuel pool of vessel head drop on the adjacent staging area. (19B.2.14

220.17

Generic Safety Issue No. 103 "Design for Probable Maximum Precipitation" (PMP) is concerned with the difference in the determination of PMP. BY using the recently developed NOAA/NWS procedures which are believed to be more realistic, PMP estimates larger than those obtained by previously used methodologies may lead to higher flood levels. Therefore, in ABWR resolution on Page 19B 2-47, specify that the recently developed NOAA/NWS procedures will be used for determining PMP for a specific site. (19B.2.17)

252.16

(1) The applicant should define bolting in detail. Bolting in this context should include bolts, studs, embedments, machine/cap screws, threaded fasteners, and associated nuts and washers.

(2) Define high strength bolting and medium strength bolting in terms of material and mechanical properties.

(3) Provide bolting manufacture process (e.g., heat treated, quenched, tempered, etc.).

(4) Provide bolting manufacture process (e.g., equipment and piping systems) where the high strength bolting or medium bolting will be used.

(5) Discuss how to avoid the intergranular stress corrosion cracking (IGSCC) of bolting in a BWR hydrogen environment.

(6) Identify thread lubricants that will be used and identify chemical compond(s) in them.

(7) The applicant discussed the ALWR Resolution initiated by the Atomic Industrial Forum/Metal Properties Council Task Group and BWR Requirements in the EPRI-ALWR Requirements Document. It is unclear whether the applicant will follow the resolutions and requirements. (19B.2.12)

260.4

The ALWR Resolution Summary for issues I.F.1 and II.F.5 states:

(1) The designer shall identify any structures, systems, or components (items) that are not safety related but for which provisions beyond normal industry practice are judged to be needed to provide desired reliability and availability.

(2) At the same time, specific surveillance, maintenance provisions (appropriate for specific item and desired reliability and availability) shall be identified for those items.

The NRC evaluation is that ALWRs should have a Reliability Program to ensure that the facility is operated and maintained within enveloping PRA assumptions throughout its life. The NRC anticipates that these new (Reliability Program) requirements will effectively subsume the I.F.1 and II.F.5 issues and these issues can be considered resolved.

The ABWR Resolution states:

(1) The ABWR application of quality system requirements satisfies the ALWR resolution.

(2) An interface requirement (Section 19B.3.1) is included to ensure that quality system requirements will be provided during construction and operation.

(3) Therefore, this issue is resolved for the ABWR.

REQUEST FOR ADDITIONAL INFORMATION 1. It is not clear to the staff that the ABWR SSAR describes how points 1 and 2 of the ALWR Resolution Summary (above) are to be satisfied. That is, how is the ABWR designer identifying items for which provisions beyond normal industry practice are judged to be needed? And how are specific surveillance/maintenance provisions being identified for those items? SSAR Table 3.2-1 is used to show the quality assurance that is applied to plant items. The table indicates that a quality assurance program meeting 10CFR50 Appendix B either does or does not apply. In some instances, where Appendix B does not apply, there is reference to a footnote regarding quality assurance. Such references are neither wide-spread enough nor specific enough to really meet an objective of the classification system which is to assign appropriate Quality Control and Quality Assurance measures.

The SSAR should be clarified in this regard, or justification should be given for not doing so. For example, footnote "u" regarding quality assurance for non-safety-related fire protection items should make it clear that a quality assurance program meeting the guidance of Branch Technical Position CMEB 9.5-1 (NUREG-0800) will be applied to each such item. Similarly, for non-safety-related radioactive waste management items, a footnote should make it clear that a quality assurance program meeting the guidance of Regulatory Guide 1.143 will be applied during design and construction. The safety parameter display system (or its equivalent), though not safety-related, should have a quality assurance program beyond normal industry practice applied, and this should be clear in Table 3.2-1. Generic Letter 85-06, "Quality Assurance Guidance for ATWS Equipment That Is Not Safety Related," is also applicable. If GE has not already done so, it should ascertain whether there are other ABWR plant items within the scope of points 1 and 2 of the ALWR Resolution summary (above) and revise Table 3.2-1 accordingly if required. Then the ABWR Resolution should reference Table 3.2-1 to show how GE has resolved TMI issues I.F.1. and II.F.5 for the ABWR.(19B.2.1)

260.5

The statement in the ABWR SSAR, "Applicants referencing the ABWR design shall have a Quality Assurance Program satisfying the requirements of Section 19.B.2.1(2) including the right to impose additional

closed) and through RHR subsystem A check valve (normally closed) into feedwater line A.

Thus potential intersystem leakage from the reactor coolant system is only postulated to occur into RHR subsystems B and C discharge lines resulting from leakage through normally closed discharge check valves and normally closed injection valves of RHR subsystems B and C. Test points and test valves are located between the discharge check valves and the discharge injection valves of RHR subsystems B and C to be used to specifically test for the leak tightness of the discharge check valves and injection valves (both normally closed). Substantial leakage through the discharge check valve and closed injection valve of either RHR subsystem B or C would result in pressurization of the discharge lines which would lead to a control room alarm. Significant pressurization of the discharge piping of RHR subsystems B or C, resulting from postulated intersystem leakage, would be discharged to the suppression pool via pressure relief valves.

Item 1 of Table 1.II of the SPR addresses the components of the safety injection systems that are connected to the Reactor Coolant System.

For the ABWR, the low pressure safety injection system is the low pressure core flood (LPCF) mode of the RHR system. The connections between the reactor coolant system and the discharge lines at the RHR subsystems were previously discussed above. In the LPCF mode of operation of the RHR subsystems, inlet suction flow is drawn from the suppression pool and not from the reactor coolant system.

For the ABWR, the high pressure safety injection system consists of the two high pressure core flood (HPCF) systems B and C (and also RCIC which was previously discussed). There is no connection between the reactor coolant system and the inlet suction of HPCF systems B and C. Both systems draw their suction flow from the condensate storage pool (or the suppression pool) and not from the reactor coolant system.

The discharge lines of HPCF systems B and C connect to the reactor coolant system through discharge check valves and injection valves. Potential intersystem leakage from the reactor coolant system is only postulated to occur into HPCF systems B and C discharge lines resulting from leakage through normally closed check valves and normally closed injection valves. Test points and test valves are located between the discharge check valves and the injection valves of HPCF systems B and C to be used to specifically test for the leak tightness of the discharge check valves and the normally closed injection valves.

Normal lineup for both HPCF systems B and C is through normally open suction valves connected to the condensate storage pool. These suction lines will fill with water down to the condensate storage pool suction check valves. The discharge lines for HPCF systems B and C are maintained full of water with water sourced from the makeup water system (condensed). Substantial (potential) leakage from the reactor coolant system through closed discharge check valves and closed injection valves into either HPCF B or C discharge lines would result in pressurization of both the discharge line and the HPCF pump suction line which would lead to a control room alarm indicating high HPCF B (or C) pump suction pressure. Significant pressurization of the suction piping for either HPCF pump B or C, resulting from postulated intersystem leakage, would be discharged to the suppression pool via a pressure relief valve.

QUESTION 430.3

Discuss compliance of reactor coolant leak detection systems with Regulatory Guide (RG) 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems", Positions C4, C5, C6, C8, and C9 with respect to the following items: (5.2.5)

(a) Indicators for abnormal water levels or flows in all the affected areas in the event of intersystem leakages.

(b) Sensitivity and response time of leak detection systems used for unidentified leakages outside the drywell.

(c) Qualification relating to seismic events for drywell equipment drain sump monitoring system and leak detection systems outside the drywell.

(d) Testing Procedures - Monitoring sump levels and comparing them with applicable flow rates of fluids in the sumps.

(e) Inclusion of reactor building and other areas floor and equipment drain sumps in ABWR Technical Specifications for leak detection systems.

Note that a few of the questions above arise because in Subsection 5.2.5.4.1 you state that the total leakage rate includes leakages collected in drywell, reactor building and other area floor drain an equipment drain sumps.

RESPONSE 430.3

As noted above, several questions arose because in Subsection 5.2.5.4.1 it was stated that:

"The total . . . leakage rate consists of all leakage, identified and unidentified, that flows to the drywell, *reactor building and other area* floor drain and equipment drain sumps."

The italicized wording was incorrectly included with the text of Subsection 5.2.5.4.1. Subsection 5.2.5.4.1 has been revised accordingly.

Historically, total leakage rate limit, as established by Plant Technical Specifications, have been associated only with the potential leakage into the reactor primary containment (drywell) as collected by the drywell floor and equipment drain sumps and as monitored by different drywell leakage detection systems, e.g., drywell atmosphere (gaseous and/or particulate) radioactivity monitoring, drywell sump/level monitoring and drywell air coolers condensate flow monitoring. Also, the recommendations and regulatory positions of Regulatory Guide (RG) 1.45 have been interpreted in the past as applying only to reactor coolant leakage into the primary containment. RG 1.45 Positions C1, C2 and C3 specifically address leakage to the primary reactor containment and indication of leakage to the containment.

RESPONSE 430.3a

This questions addresses compliance with RG 1.45 Position C4 which recommends that provisions should be made to monitor systems connected to the RCPB for signs of intersystem leakage and also suggests that monitoring and indicators to show abnormal water level or flow in the affected areas. Specifically, this questions requests discussion of compliance with RG 1.45 Position C4 with respect to the "indicators for abnormal water level or flows in all the affected areas in the event of intersystem leakages."

As indicated in the Subsection 5.2.5.9 discussion that is related to RG 1.45 compliance, radiation monitoring of the reactor building cooling water coolant return lines from the RHR, RIP, CON and SPCV heat exchangers is the monitoring method used for determining potential intersystem leakage from the RCPB within these heat exchangers.

Also, in the discussion related to the response to part C of Question 430.2, it is indicated that

QUESTION 440.19

Explain in detail how the spring and relief modes of the SRV works. Are they an difference from the SRVs currently used in operating BWRs?

RESPONSE 440.19

It is currently anticipated that the basic ABWR SRV configuration will be very similar to that of direct acting SRVs used on BWR/6 plants. The spring is used to provide the force to the SRV stem, which in turn forces the disk down on the valve seat, capturing the steam on the inlet side of the valve. When the steam pressure increases so that it's force is equal to the downward force provided by the spring, the disk and stem begin to lift. As this happens, the steam pressure acts over a larger area of the disk and the valve opens quickly. This actuation using pressure to overcome spring force is safety mode actuation. In relief mode actuation, an actuator system uses a large pneumatic piston to lift a lever which in turn lifts the valve stem against the force of the spring. This system is driven by one or more solenoid operated pneumatic valves and normally has an intermediate stage pneumatic valve (one driven by each solenoid valve) to provide the required pneumatic flow rate to the piston. This pneumatically driven method of operating the SRV (the relief mode of operation) must never interfere with operation of the SRV in the safety mode.

The main spring of the ABWR SRV may be a double spring configuration or a set of Belleville washers. The main spring(s) will be controlled by specification requirements and manufacturing standards to provide smooth operating and relative motion characteristics and to minimize SRV set point drift due to spring relaxation. The valve stem will be guided to preclude any tendency to cock or gall.

The ABWR could use a pilot operated valve. This type of valve has a pilot stage which acts as a small direct acting valve. When the pilot stage opens, steam is redirected inside the valve so that the main disk opens. The relief mode actuator is connected to the pilot stage.

QUESTION 440.20

What ATWS considerations have you given for sizing SRVs?

RESPONSE 440.

The most limiting ATWS event (i.e., MSIV closure) has been analyzed. The peak vessel bottom pressure for this case is 1300 psig, which is well below the ATWS overpressure criterion of 1500 psig.

QUESTION 440.21

In Subsection 5.2.2.2.3 the reclosure pressure setpoint (% of operating setpoint) for both modes are given as 98 and 23. Explain the significance of these numbers.

RESPONSE 440.21

Subsection 5.2.2.2.2.3 has been revised to reflect SRV safety mode reclosure points consistent with new ABWR requirements. These new requirements are 96% of nameplate opening setpoint (4% blowdown) to 90% of nameplate setpoint (10% blowdown). The lower reclosure limit (10% blowdown) is based on JIS Standard B8210-1986 and reflects a requirement imposed by MITI on plants built in Japan. It is a reasonable lower limit for two reasons: (1) It makes sense, economically, not to let more steam escape from the reactor system than is necessary to protect the system within a reasonable safety margin. (2) It provides an effective lower limit from a system standpoint so that SRV opening

and reclosure in the safety mode occur at a higher pressure than the respective "normal" opening and reclosure in the relief mode (i.e., as normally initiated by pressure sensors in the steam lines).

The upper reclosure limit (reclosure point at 96% of opening setpoint) is a reasonable upper limit which will serve to limit the number of times the SRV will open and reclose in case of a pressure transient causing valve operation in the safety mode. It permits the valve to remain open longer and cycle less often (as compared with prior allowed upper reclosure limits, which were set at 97% and 98% of opening setpoint in the past).

The 96% upper limit also provides an extra measure of insurance that deviations in manufacturing tolerances, actual back-pressure in service, and other such variables do not result in an SRV with negative blowdown, in which buildup of backpressure would reclose the valve before it could perform its pressure relief function.

QUESTION 440.22

In Figure 5.1.3a the SRV solenoid valves are not shown as DC powered as they should be. Note 3 states that "valve motor operators and pilot solenoids are ac operated unless otherwise specified."

RESPONSE 440.22

At the next revision, Figure 5.1.3a will be revised to show that the SRV solenoids are DC powered.

QUESTION/RESPONSE 440.23

This question number not used.

QUESTION 440.24

Confirm that SRVs are designed to meet seismic and quality standards consistent with the recommendations of Regulatory Guides 1.26 and 1.29.

RESPONSE 440.24

(1) The SRVs are classified as Quality Group A and Seismic Category I as shown in Table 3.2-1. The SRVs are designed to meet Regulatory Guides 1.26 and 1.29. Tests required by ASME Code Section III for Class I valves are imposed in the ABWR SRV equipment specification. Analyses equivalent to those required by ASME III are performed in accordance with the requirements of MITI-501 (the Japanese equivalent of ASME III).

(2) SRV's are Class IE (active, safety related, electrically driven). It is currently planned to impose a complete environmental qualification program on the entire SRV, including both electrically and pneumatically driven components of the actuator system. This program includes dynamic qualification of operability following the Japanese equivalent of an SSE. This program will be in compliance with NUREG-0588 requirements.

QUESTIONS/RESPONSE 440.25 through 440.27

These questions numbers not used.

QUESTION 440.28

In SSAR Table 1.8-19, it is stated that branch technical position RSB 5-2 is applicable for ABWR. How does the ABWR design comply with BTP RSB 5-2?

rerouted to the vessel should system initiation be required during CST to CST testing. There would also be additional interlocks needed to prevent pumping suppression pool water to the CST. Complexity and cost would also increase from the required maintenance of the additional hardware, instrumentation and logic.

Suppression pool water quality will be maintained by the suppression pool cleanup system which is designed to be operated continuously. Although this quality may be somewhat less than that of the CST, it will be consistent with infrequent filling of RCIC piping during testing and possible injection to the RPV and therefore the reference draining, flushing and filling of the system is not necessarily required. Additionally, and decrease in personnel exposure realized by performing CST to CST testing (assuming draining, flushing and filling were required) might be fully or partially offset by an increase from the additional maintenance considerations.

QUESTION 440.42

Why are the power supply for valves F063, F064, F076, and F078 standby AC instead of DC?

RESPONSE 440.42

For the ABWR RCIC, only the steam supply inboard isolation valves F035 and F048 are powered from AC source. F036 and all other MOV's are DC powered. Figure 5.4-8 has been updated and indicates the correct power supply; also, valves F063, F064 and F076 have been re-designated F035, F036 and F048, respectively.

Valves F077 and F078 have recently been removed from the ABWR RCIC design. The line where these valves were located performed a vacuum breaking function of the turbine exhaust line and had a separate containment penetration. The current ABWR RCIC configuration eliminated F077 and F078 since the vacuum breaking function is now inside containment and has no separate penetration that mandates provision for F077 and F078.

The use of AC power source for F035 and F048 is considered technically acceptable for the following reasons:

(1) DC motors require considerably more maintenance than AC motors. Since they cannot be maintained during plant operation if they are located inside the drywell, DC MOVs would be far less reliable than AC.

(2) During loss of AC power RCIC system will remain operable since these valves are normally open.

QUESTION 440.43

Address the following TMI-2 action Items related to RCIC

(a) II.K.1.22

(b) II.K.3.13

(c) II.K.3.15

(d) II.K.3.22

(e) II.K.3.24

RESPONSE 440.43

Response to this question is provided in Appendix 1A.

QUESTION 440.44

Confirm that the RCIC system meets the guidelines of Regulatory Guide 1.1 regarding pump Net Positive Suction Head (NPSH).

RESPONSE 440.44

The key requirement of Regulatory Guide 1.1 is that no credit be taken for containment pressurization when establishing the NPSH conditions for ECCS pumps. The RCIC meets this requirement. New Table 5.4-1a provides the numerical evaluation of RCIC NPSH conditions assuming no containment pressurization and 77°C suppression pool water temperature. In summary, the RCIC pump will have over 0.85 meter NPSH margin at the most limiting condition.

Note that NPSH calculation is based on suppression pool temperature of 77°C. This is the maximum temperature RCIC is expected to operate.

The following summarizes the transient/accident events which can result in increasing suppression pool water temperature. It summarizes the basis for concluding that RCIC NPSH conditions (1.03 kg/cm^{2}abs containment pressure, 77°C suppression pool water) are acceptable.

EVENT	RCIC NPSH ASSESSMENT *
Reactor Isolation Event	Maximum pool temperature well below 77°C (approx. 49°C)
Large Break LOCA	Rapid vessel depressurization. RCIC not required.
Intermediate Size LOCA	Rapid vessel depressurization. Reactor pressure less than 10.5 kg/cm^{2}g before pool temperature reaches 77°C.
Small Break LOCA	RCIC operation not required when pool temperature reaches 77°C.
Station Black Out Event (8 hours capability)	RCIC suction is taken from the condensate storage tank (CST) with a capacity of 8 hour operation. Suppression pool (S/P) water is not expected to be used during this event. However, if the automatic transfer of suction from the CST to S/P were to occur due to high S/P water level, a manually controlled override switch is operated to continue taking suction from the condensate storage tank.

*RCIC design basis requires 100 percent system flow only for reactor pressure > 10.5 kg/cm^{2}g.

QUESTION 440.45

SRP 5.4.6 identifies GDCs 5, 29, 33, 34 and 54 in the acceptance criteria. Confirm that the RCIC system, described in Chapter 5.4.6 of the SSAR, meets the requirements of the above GDCs.

RESPONSE 440.45

Evaluations of the reactor core isolation system against the applicable General Design Criteria (GDC) are provided in Subsection 3.1.2 (statement to this effect has been added to Subsection 5.4.6). Based on the evaluations in Subsection 3.1.2 it is concluded that the RCIC system meets the requirements of the applicable GDCs.

QUESTION 440.46

In SSAR Chapter 5.4.6.3, it is stated "The analytical methods and assumptions in evaluating the RCIC system are presented in Chapter 15 and Appendix 15A." Identify the section in Chapter 15 where the analytical methods and assumptions evaluation the RCIC systems are given.

RESPONSE 440.46

This is given in Subsection 15.6.5.3 which, in turn, makes reference to Section 6.3.

QUESTION 440.47

Normally the RCIC pump takes suction from the condensate storage tank (CST). But the CST is not seismically qualified or safety related. Confirm that the system piping and level transmitters, which interface with CST, will be designed and installed such that the automatic switchover to the suppression pool takes place without failure.

RESPONSE 440.47

Four redundant Class 1E level transmitters are seismicall installed in the condensate storage tank (CST). One level sensor in each electrical Division (Div. 1, Div. 2, Div. 3 and Div. 4) provides input signal and is processed in 2-out-of-4 logic configuration. If the amount of water in the CST decreases below the setpoints of the level sensors, on the CST failed seismically, a signal will be sent to perform the automatic switchover to the suppression pool. The system piping from the suppression pool to the RCIC pump is seismically qualified.

QUESTION 440.48

The equipment and component description given in 5.4.6.2.2 is very brief. What type of turbine is used in the ABWR? Is it the same type as the Terry Turbines used in current BWRs? Is the turbine

testing done by Terry Co. with water applicable to the ABWR? Describe in detail the components, especially the turbine and the pump.

RESPONSE 440.48

The ABWR RCIC equipment specification does not specify the type of turbine, rather, its performance requirements. Performance testing will be performed with water applicable to the ABWR Standard Plant design. The equipment and component description given in Subsection 5.4.6.2.2 is commensurate with a standard design. The depth of information provided in this subsection is the same as that provided for GE's standard BWR/6 Nuclear Island design. This information is reflected in the RCIC equipment specification. The amount of information provided is sufficient to delineate the performance requirements of the RCIC without restricting its supply by qualified equipment vendors.

QUESTION 440.49

To the best of our knowledge, the steam isolation valves F063 and F064 in currently operating BWRs are not tested with a steam pipe break downstream and with actual operating conditions (pressure 1000 psig and temperature 546 degrees F). There is no guarantee that the steam isolation valves will close during a break. We require that a proper testing of the valves be performed before the final design approval. (Reference Generic Issue GI-87 "Failure of HPCI Steam Line Without Isolation.")

RESPONSE 440.49

The ABWR RCIC equipment specification requires that the valves in question close within a specified time under actual operating conditions. Since this is a standardized design it is not possible to indentify a specific equipment vendor and test the valve before the final design approval. However, GE will closely follow the current valve testing in support of GI-87 and , if necessary, will make appropriate modifications to the equipment specification prior of issuance of the final SER. Figure 5.4-8 has revised designation of valves F063 and F064 to F035 and F036 respectively.

QUESTION 440.50

Steam isolation valves F063 and F064 are to be opened in sequence to reduce water hammer and for slow warm-up of the piping. F064 and F076 are opened first. The valves logic should prevent the operator from opening the valves out of sequence. Confirm that the valves control logic includes an interlock.

RESPONSE 440.50

The inboard (F035) and outboard (F036) isolation valves are provided with keylock switches as protective features in addition to several administrative constraints. Administratively, the valve control switch key must be obtained, then (1) the key must be inserted into the lock to enable the maintained contact switch and (2) the switch must be turned from OPEN to CLOSE to enable reset of the sealed-in isolation signal from the leak detection system. (An interlock for the isolation valve to be in CLOSE position before the leak detection system isolation signal can be reset is in compliance with NUREG-0737. NUREG-0737 Item II.E.4.2 Position 4 requires that isolation valves must not open automatically upon reset of the isolation signal and must only be opened by a deliberate operator action).

Upon reset of the leak detection system, the outboard isolation valve (F036) is allowed to open by placing the control switch in the OPEN or STOP (intermediate position for throttling) position to drain trapped condensate between the inboard and the ourboard isolation valves. Then the inboard

RESPONSE 410.46

The surge volume of the system is within the condensate storage tank (CST). The capacity requirements of the CST are in Table 9.2-3. Section 3.4 demonstrates that failure of the CST will not lead to unacceptable results. "HPCF pumps" means the high pressure core flooder pumps.

QUESTION 410.47

Describe the design features provided in the system and/or interfacing components to ensure automatic switchover of the suction of the applicable pumps to safety-related water sources, if so required. (9.2.9)

RESPONSE 410.47

Level sensing elements and transmitters are provided for the condensate storage tank (CST). Signals are sent to the HPCF and RCIC pumps to provide automatic switchover to the suppression pool when sufficient water is not available in the CST. The switchover of the SPCU pumps is manual.

QUESTION 410.48

Discuss conformance of the MUWC systems design with the requirements 10 CFR 50.63, "Loss of all Alternating Current Power." Specifically include the system's capacity and capability to ensure core cooling by removing decay heat independent of preferred and onsite emergency ac power in the event of a station blackout for the specified duration, in accordance with guidelines of Regulatory Guide 1.55, "Station Blackout," Positions C.3.2 through C.3.5, as applicable. (9.2.9)

RESPONSE 410.48

The condensate storage tank (CST) is designed to provide approximately 150,000 gallons of water for use during station blackout. Other consumers of condensate are switched to other water sources so that this volume of water is always available during power operation. This volume of water is sufficient for operation of the RCIC system to remove decay heat during the first eight hours of station blackout.

QUESTION 410.49

Discuss compliance of the system with Positions C1 and C2 of Regulatory Guide 1.29. (9.2.9)

RESPONSE 410.49

The normal secured source of water for decay heat removal is the suppression pool. The condensate storage tank (CST) is used in preference to the suppression pool because the water quality is normally better. As a result the CST is not required to be Seismic Category I.

QUESTION 410.50

Provide P&IDs for the Demineralized Water Makeup System (i.e., Makeup Water System (Purified) (MUWP)). (9.2.10)

RESPONSE 410.50

The MUWP P&ID is provided as Figure 9.2-5.

QUESTION 410.51

Clarify which portion of the MUWP is within the ABWR scope. Also, identify the system interfaces which include temperature, chemistry, system capacity (i.e., tank volume) and treatment. (9.2.10)

RESPONSE 410.51

See response to Question 410.52.

QUESTION 410.52

Provide the water quality characteristics for the MUWP water (SSAR Section 9.2.10.1, Item 3, refers to Section 9.2.8 which in turn refers to Section 9.2.16. However, Section 9.2.16 does not give the water quality characteristics). (9.2.10)

RESPONSE 410.52

The response to this question is provided in new Table 9.2-2a.

QUESTION 410.53

Discuss compliance of the system with Position C1 (e.g., containment penetration portions) and Position C2 of Regulatory Guide 1.29. (9.2.10)

RESPONSE 410.53

The MUWP line which enters primary containment has a locked closed manual valve outside of containment and a check valve inside of containment. The containment penetration is Seismic Category I and Quality Group B. (9.2.10)

QUESTION 410.54

Verify that flooding analyses have been performed for a failure of the nonseismic Category I demineralized water makeup system where the piping runs through safety-related structures and tunnels containing safety-related equipment. (9.2.10)

RESPONSE 410.54

Section 3.4 studies of MUWP piping run failures through safety-related structures and tunnels containing safety-related equipment have shown that flooding will not have an adverse effect on these structures and equipment.

QUESTION 410.55

With respect to the capability of the Reactor Building Cooling Water System for detection, control, and isolation of system leakage, and radioactive leakage: (9.2.11)

1. Identify the isolation valves which isolate the non-essential loads from the essential supply headers and describe their isolation function in the event of a LOCA or in the event of a leak detected in the non-essential system piping.

QUESTION 420.87

The response noted that RIP trips have mostly been caused by noise in the adjustable speed drive (ASD). Describe the changes that have been made to reduce the susceptibility of the RIP's or the reduction in noise of the ASD's.

RESPONSE 420.87

The primary cause of faulty trips of Adjustable Speed Drives (ASD) in the European plants with Reactor Internal Pumps (RIPs) is due to electromagnetic interference (EMI) and harmonic effects on the improperly installed ASD power and control cables. Several preventive changes have been made to the ASD design and installation requirements to reduce their occurrence. These changes can be summarized as below:

(1) Interconnection cables within the ASD are designed with consideration for EMI effects. Factory tests will be performed on the ASD equipment to demonstrate that EMI effects are within the tolerance limits of the ASD control functions. Proper isolation between the power and the control cables is specified to eliminate propagation of EMI noises. Both the ASD equipment requirements specifications and the installation specification reflect the requirements of the "Special Wire and Cable Specification" (see Subsection 1.1.3) for interconnection of static converter device.

(2) The ASD is designed to ensure that current and voltage distortions due to harmonics on the output power waveform is tolerable by both the connected loads and the input power distribution system. Phase-shifted isolation transformers are utilized to reduce harmonic contents on the power supply circuits. These specifications have been defined as interface requirements on the affected equipment designs.

QUESTION 420.88

List the criteria or standards for surge withstand capability to be applied to the equipment. ANSI/IEEE C62.45-1987 "Guide on Surge Testing Equipment Connected to Low-Voltage AC Power Circuits" is an example of criteria currently being applied to limit the possible affects from, line surges. (7)

RESPONSE 420.88

The answer to this question is included in the response to Question 420.7.

QUESTION 420.89

List the design goals for the survivability and continued operation of safety systems equipment in the presence of line switching transients, lightning induced surges and other induced transients within the systems as installed. (7)

RESPONSE 420.89

Surge withstand capability, and associated testing criteria, is discussed in Sections 7A.2 [Response (4)] and 7A.3 [Response (8)] of Appendix 7A.

QUESTION 420.90

Address the possible effects of electrostatic discharge (ESD) at keyboards, keyed switches and other exposed equipment components. (7)

RESPONSE 420.90

If appropriate countermeasures are not taken, then electrostatic discharge (ESD) can cause damage to electronic components. High impedance devices using MOS (metal-oxide semiconductor) technology are particularly subject to damage. The discharge from an electrically charged human body, when certain areas of electronic equipment are touched (keypads, switches), may open the junctions of CMOS devices or other semiconductors.

However, modern CMOS and other MOS components have internal protection against ESD in the form of diode clamping arrays and current limiting resistors that conduct the discharge away from the junction. In addition, good circuit design practices will include the use of other devices such as transient suppressors [for example, metal-oxide varistors (MOVs), Zener diodes] across critical circuit inputs and outputs that are directly exposed to external transients.

Other precautions against the effects of ESD take the form of adequate insulation or proper grounding. Keypads generally have insulating material in the form of a thick plastic covering over the metallic switch contacts. Toggle switches and other controls should have insulating knobs. Various metallic chassis components (front panel, handles, deck, connector shells) should be solidly grounded to each other (the effects of painted and plated surfaces should be considered) and the chassis should be grounded to the appropriate panel or instrument ground bus by metallic ground straps. Panel and instrument mounting hardware should not be depended upon for solid grounds. Printed circuit boards must have the signal commons and ground plane commons properly connected to the common busses and to the low voltage logic power supplies.

Microprocessor-based control equipment for ABWR is designed under the assumption that users will have taken no precautions against static charge buildup before attempting to operate the equipment. The equipment is designed to tolerate an electrostatic discharge without damage, partly by employing insulation (with no air gaps) over exposed metallic components, but primarily by providing an alternative path for current flow other than through sensitive circuit paths. As discussed previously, this means that all exposed metallic components of the system must be grounded. Low inductance multipoint grounds are used where ESD current flow is desired and single-point grounds where discharge flow is not wanted.

The low power requirements of ABWR control equipment ensure that the integrity of the equipment enclosures is not compromised by large ventilating holes or slots. Special attention is given to hinges, joints, and seams so that the continuity of shielding is maintained.

In the system configuration, where shielded cables transfer data between the equipment enclosures, the cables must be prevented from propagating ESD currents and voltages between system units. For ABWR safety systems, the problem has been minimized by using fiber optic cables as the transmission medium for most critical signals. While the cables may contain metallic supporting members or protective shields, these will not be electrically connected to any equipment or circuit. For certain functions where hardwired cable is required, solid grounding of cable shields to the equipment chassis and bypass capacitors at all inputs and outputs shall be used to divert ESD currents to ground.

These hardware solutions shall be supplemented with firmware ESD solutions to protect against potential upsets such as system lockup if ESD noise causes memory or data flow errors. The methods used are discussed as part of the fault-tolerance issues.

The susceptibility of ABWR control equipment to electrostatic discharges shall be established using the test procedures included in IEC Publication 801-2, Electromagnetic Compatibility for Industrial-Process Measurement and Control Equipment. Part 2: Electrostatic Discharge Requirements. The test procedures of paragraph 8 of this document shall be performed up to and including Severity Level 4, as defined in the document. The following acceptance criteria shall be used:

(1) No change in trip output status shall be observed during the test.

(2) Equipment shall perform its intended functions after the test.

Note that the safety system control equipment for ABWR has inherent protection against transient ESD effects in that data is continually refreshed throughout the system, including trip, display and indicator status. Further protection is provided by the asynchronous, four-division, 2-out-of-4 channel configuration. Temporarily corrupted data in one division cannot cause an inadvertent trip or permanently disable a required trip. When bad data or equipment damage is detected, the affected division can be bypassed until repaired. In the reactor protection system (RPS) and main steam isolation valve (MSIV) channels, where the final trip outputs are also in a 2-out-of-4 configuration, both the sensor input and trip output sides of each equipment division can be bypassed, thus preventing failure from any cause in one channel from inhibiting or inadvertently causing a trip.

QUESTION 420.91

Most of the I&C system microprocessor equipment is likely to be located in a mild environment, but survivability requirements or limitations on the voltage potential buildup by humidity control or other measures is not discussed. Also, the data concentrators are provided at remote locations where the environmental control is not clearly described. Identify the criteria, design limits and testing program for this area of ESD controls. (7)

RESPONSE 420.91

The environmental qualification requirements for systems and equipment are described in Section 3.11 and in the design documents referenced in Subsection 1.1.3 (in particular, BWR Requirements - Equipment Environmental Interface Data and the Safety System Logic & Control Design Specification).

Voltage potential buildup will be limited by proper grounding of equipment and use of appropriate static control materials and dielectric barriers to ensure that high potentials cannot be coupled to sensitive semiconducter devices (see the response to Question 420.90). Humidity controls are provided by the normal and emergency HVAC systems; when relative humidity is restricted to the ranges specified for the mild environment locations where the microprocessor equipment will be installed, there will be no unusual static charge buildup.

The thermal design environments for the SSLC panels themselves are discussed in the response to Question 420.008. The Remote Multiplexing Units (i.e., "data concentrators") of the Essential Multiplexing System are located within the "clean" areas of the Reactor Building outside the secondary containment. The panels containing this equipment will be environmentally qualified and tested in accordance with Regulatory Guide 1.89 and IEEE 323 for the areas in which they are located.

I&C microprocessor equipment will be required to meet the requirements of IEC Standard Publication 801-2, "Electromagnetic Compatibility for Industrial Process Measurement and Control Equipment, Part 2 (Electrostatic Discharge Requirements)". Test equipment shall have the following minimum capabilities:

20.3.14 Response to Fourteenth RAI-Reference 14

QUESTION 620.1

Describe GE's human factors design team, the staff's human factors expertise, and its responsibilities for human factors on the ABWR design.

RESPONSE 620.1

The team is responsible for the compliance of the plant design with all regulatory requirements related to Human Factors Engineering and the inclusion of good human factors engineering practice in all aspects of the plant design. The five members of the team have a total of 99 years of human factors engineering, managerial, quality assurance, plant operating and design experience on BWRs.

Descriptions of the individual members of the ABWR human factors team are as follows.

Chairman: Twenty seven years of human factors engineering, licensing and design experience at GE in nuclear energy. Responsible for the development of the man-machine interface design for the ABWR and SBWR projects. Experience includes human factors engineering for the General Electric Emergency Response Information System (SPDS) and GEPAC Plus and NUMAC computer products

Second Team Member: Fifteen years experience at GE in nuclear energy. Developed Human Factors Engineering Plan and documentation system for ABWR Project. Currently technical leader of an international study group working on design optimization for the SBWR Project.

Third Team Member: Thirty one years experience at GE in nuclear energy. Experience includes the evaluation, development and implementation of quality assurance requirements and procedures for the ABWR certification program in the United States and the ABWR Project in Japan, the audit and review of GE quality and management activities both domestical and international and the review and development of both BWR and fast reactor fuel.

Fourth Team Member: Twelve years experience at GE in nuclear energy. Experience includes instrumentation and control system design, human factors engineering of nuclear power plant control rooms, containment transient analysis and program management. Conducted detailed control room design reviews for several domestic and foreign operating and requisition nuclear plants.

Fifth Team Member: Fourteen years experience at GE in nuclear energy. Responsible for preparation of Chapter 18, *"Human Factors Engineering"*, of the ABWR SSAR. Principal contributor to the development of plant automatic operation and control room design. Developed emergency operating procedures, system operating procedures and integrated operating procedures for the ABWR.

QUESTION 620.2

Both Hitachi and Toshiba are designing main control room workstations which, although based upon the common engineering studies, may result in two different workstation design implementations within one two-unit control room. Describe the process that GE will use to actually implement high-level, single-unit workstation requirements and design selection, including the decision process to be followed in selecting the Hitachi or Toshiba approach, a hybrid, or a different design.

RESPONSE 620.2

The control room design definition documented in the ABWR Standard Safety Analysis Report (SSAR) is specifically independent of any particular equipment vendor's details of design implementation. The main

20.3.17 Response to Seventeenth RAI-Reference 17

QUESTION 210.51

The information in this section should be revised to more nearly reflect the current status of this issue. GSI II.E.6.1 originally consisted of the following sub-issues:

(1) In-situ testing of motor operated valves (MOV)

(2) In-situ testing of pressure isolation valves (PIV)

(3) Reevaluation of thermal overload protection devices for motor operated valves.

(4) In-situ testing of check valves

Sub-issues 1, 2 and 3 are no longer considered to be part of II.E.6.1. Sub-issue 1 was subsumed by the staff's evaluation of responses to Generic Letter 89-10, "Safety Related MOV Testing and Surveillance". Sub-issue 2 was subsumed by Generic Safety Issue 105, "Interfacing Systems LOCA in Light Water Reactors". Sub-issue 3 is considered to be resolved for the ABWR on the basis of the unconditional commitment in the SSAR Table 1.8-20 to Regulatory Guide 1.106, "Thermal Overload Protection for Electric Motors on Motor-Operated Valves". Sub-issue 4 remains unresolved at this time. During a meeting on April 7,1986 between the staff and industry representatives, it was agreed that industry would initiate an aggressive program to resolve the check valve issue. Since that time, the Institute (EPRI), the Nuclear Power Operation (INPO), the Electric Power Research Institute (PERI), the Nuclear Industry Check Valve Group (NIC) and the staff have made some progress in addressing this issue. However, as stated in a letter to Mr. Z. T. Pate, President of INPO, dated April 20, 1990, the staff continues to find weakness in the efforts of individual licensees to improve the performance of check valves. To assist the staff in its continuing evaluations and perspectives regarding the resolution of the check valve issue. The staff has not yet received a complete response to this request.

The staff does not agree that the information in the "ABWR Resolution" of Subsection 19B.2.2 in the SSAR is sufficient to resolve this issue for the ABWR. The exceptions to position indication of check valves will require some clarification. However, the staff prefers that this type of information be included as a part of the ASME Section XI Inservice Test Program for safety-related pumps and valves which is discussed in the SSAR, Subsection 3.9.6. Therefore, GE is requested to revise Subsection 19B.2.2 related to sub-issue 4 to reflect a more broad commitment to the collective industry and NRC activities relative to implementation of the resolution of issues on in-situ testing of check valves. In addition, the staff will need to complete its review of the ABWR Inservice Testing Program before this issue can be considered resolved.

Since sub-issue 1 has been subsumed, Subsection 19B.2.2 should also include a commitment to provide a response to Generic Letter 89-10 which will be applicable to the ABWR. (19B.2.2)

RESPONSE 210.51

This response is contained in revised Subsection 19B.2.2.

The ABWR resolution to in-situ testing of valves is presented in Subsection 3.9.6. A plan of periodic testing that implements the ASME Code, Section XI, Subsection IWV, for safety related valves is outlined.

To insure MOV operability when subjected to the design basis conditions considered during both normal operation and abnormal events, the detail design is committed to a test program that is responsive to Generic Letter 89-10. GL 89-10 is an interface requirement listed in Table 1.8-22.

QUESTION 210.52

Recent BWR operating experience indicates that the isolation valves between the RCS and low pressure interfacing systems may not adequately protect against overpressurization of low pressure systems.

For ABWRs, pressure isolation valve instrumentation and controls are provided to (1) prevent opening shutdown cooling connections to the vessel in any loop when the pool suction valve, discharge valve, or spray valves are open in the same loop, (2) prevent opening the shutdown connections to and from the vessel whenever the RCS pressure is above the shutdown range, (3) automatically close shutdown connections when RCS pressure rises above the shutdown range, and (4) prevent operation of shutdown suction valves in the event of a signal that the water level in the reactor is low.

The ABWR has been designed to minimize the possibility of an interfacing system LOCA in the following ways. The low pressure systems directly interfacing with the RCS are designed with 500 psig piping which provides for a rupture pressure of approximately 100 psig. In addition, the high/low-pressure motor-operated isolation valves have safety-grade, redundant pressure interlocks. Also, the motor-operated emergency core cooling system (ECCS) valves will only be tested when the reactor is at low pressure. All inboard check valves on the ECCS will be testable and have position indication. Additionally, design criteria used by GE require that all pipe designed to 1/3 or greater of reactor pressure requires two malfunctions to occur before the pipe would be subjected to reactor system pressure. The pipe designed to less than 1/3 reactor pressure requires at least three malfunctions before the pipe would be subjected to reactor system pressure.

Position

Since ABWR low pressure systems are designed only for 500 psig rather than the full RCS design pressure of 1250 psig, the ABWR design should provide (1) the capability for leak testing of the pressure isolation valves, (2) valve position indication that is available in the control room when isolation valve operators are deenergized and (3) high-pressure alarms to warn control room operators when rising RCS pressure approaches the design pressure of attached low-pressure systems and both isolation valves are not closed. It is the staff's position that GE should confirm that the above design features are incorporated into the ABWR design.

GI-96 was related to PWRs which considers the failure of the low pressure isolation valves between the RCS and RHR system in PWRs. The issues contained in GI-96 now are incorporated into GI-105. (19B.2.15)

RESPONSE 210.52

(1) The response for leak testing of reactor coolant pressure isolation valves is contained in the revised Subsection 3.9.6.

(2) All PIV's in RHR, HPCF, and RCIC systems have position indication in the control room. The SLC system outboard valve (motor-operated) is provided with local as well as control room position indication.

(3) The RHR system "Low Pressure Flooder" (LPFL) High/low interface is provided with a high pressure alarm at the downstream of the pump discharge check valve. A rising reactor coolant pressure will trigger the alarm when it approaches the set point pressure (less than the low pressure piping design pressure) and both PIV's are open to pressurize the piping.

The RHR system "Shutdown Cooling" (SDC) high/low interface is provided with a high pressure alarm between the outboard PIV and the pump suction valve. A rising reactor coolant pressure will trigger the alarm when it approaches the set point pressure (less than the piping design pressure) and both PIV's are open to pressurize the piping.

RESPONSE 252.16

(1,5,7) The response is contained in the changes to Subsection 19B.2.12.

(2,3,4,6) The response is an interface requirement contained in the revised Subsection 19B.2.12.

QUESTION 260.4

The ALWR Resolution Summary for issues I.F1 and II.F.5 states:

(1) The designer shall identify any structures, systems, or components (items) that are not safety related but for which provisions beyond normal industry practice are judged to be needed to provide desired reliability and availability.

(2) At the same time, specific surveillance, maintenance provisions (appropriate for specific item and desired reliability and availability) shall be identified for those items.

The NRC evaluation is that ALWRs should have a Reliability Program to ensure that the facility is operated and maintained within enveloping PRA assumptions throughout its life. The NRC anticipates that these new (Reliability Program) requirements will effectively subsume the I.F.1 and II.F.5 issues and these issues can be considered resolved.

The ABWR Resolution states:

(1) The ABWR application of quality system requirements satisfies the ALWR resolution.

(2) An interface requirement (Section 19B.3.1) is included to ensure that quality system requirements will be provided during construction and operation.

(3) Therefore, this issue is resolved for the ABWR.

REQUEST FOR ADDITIONAL INFORMATION 1. It is not clear to the staff that the ABWR SSAR describes how points 1 and 2 of the ALWR Resolution Summary (above) are to be satisfied. That is, how is the ABWR designer identifying items for which provisions beyond normal industry practice are judged to be needed? And how are specific surveillance/maintenance provisions being identified for those items? SSAR Table 3.2-1 is used to show the quality assurance that is applied to plant items. The table indicates that a quality assurance program meeting 10CFR50 Appendix B either does or does not apply. In some instances, where Appendix B does not apply, there is reference to a footnote regarding quality assurance. Such references are neither wide-spread enough nor specific enough to really meet an objective of the classification system which is to assign appropriate Quality Control and Quality Assurance measures.

The SSAR should be clarified in this regard, or justification should be given for not doing so. For example, footnote "u" regarding quality assurance for non-safety-related fire protection items should make it clear that a quality assurance program meeting the guidance of Branch Technical Position CMEB 9.5-1 (NUREG-0800) will be applied to each such item. Similarly, for non-safety-related radioactive waste management items, a footnote should make it clear that a quality assurance program meeting the guidance of Regulatory Guide 1.143 will be applied during design and construction. The safety parameter display system (or its equivalent), though not safety-related, should have a quality assurance program beyond normal industry practice applied, and this should be clear in Table 3.2-1. Generic Letter 85-06, "Quality Assurance Guidance for ATWS Equipment That Is Not Safety Related," is also applicable. If GE has not already done so, it should ascertain whether there are other ABWR plant items within the scope of points 1 and 2 of the ALWR Resolution summary (above) and revise Table 3.2-1 accordingly if required. Then the ABWR Resolution should reference Table 3.2-1 to show how GE has resolved TMI issues I.F.1. and II.F.5 for the ABWR.(19B.2.1)

RESPONSE 260.4

(1) Question

How is the ABWR designer identifying items for which provisions beyond normal industry practice are judged to be needed?

(1) Response

Clarified as requested in response to questions (3) through (9).

(2) Question

How are specific surveillance/maintenance provisions being identified for those items?

(2) Response

Clarified as requested in response to questions (3) through (9).

(3) Question

Table 3.2-1 is used to show the quality assurance that is applied to plant items. The table indicates that a quality assurance program meeting 10 CFR 50 Appendix B either does or does not apply. In some instances, where Appendix B does not apply, there is reference to a footnote regarding quality assurance. Such references are neither wide-spread enough or specific enough to really meet an objective of the classification system which is to assign appropriate Quality Control and Quality Assurance measures. The SSAR should be clarified in this regard, or justification should be given for not doing so.

(3) Response

Table 3.2-1 footnote "e" has been revised and will apply the new "E" QA requirement to all NNS items as indicated in Table 3.2-1.

(4) Question

Footnote "v" regarding quality assurance for non-safety-related fire protection items should make it clear that a quality assurance program meeting the guidance of Branch Technical Position CMEB 9.5-1 (NUREG-0800) will be applied to each such item.

(4) Response

Table 3.2-1 footnote "t" has been revised. (Footnote "u" applies to other systems in addition to Fire Protection, whereas "t" applies only to the fire protection system.)

(5) Question

For non-safety-related radioactive waste management items, a footnote should make it clear that a quality assurance program meeting the guidance of Regulatory Guide 1.143 will be applied during design and construction.

(5) Response

Table 3.2-1 footnote "p" has been revised.