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1.0 INTRODUCTION AND GENERAL DESCRIPTION OF STATION

1.1 INTRODUCTION

This Updated Safety Analysis Report (USAR) for the Davis-Besse Nuclear Power Station Unit No. 1 was prepared in accordance with the requirements of the Code of Federal Regulations Title 10, Part 50 (10CFR50) Section 50.71(e) as published in the Federal Register on May 9, 1980. Davis-Besse Nuclear Power Station Unit No. 1 is located on Lake Erie in Ottawa County, Ohio, approximately six miles northeast of Oak Harbor.

The station has a pressurized water reactor nuclear steam supply system furnished by The Babcock & Wilcox Company. The Bechtel Corporation and its affiliate, The Bechtel Company, provided to Toledo Edison architect-engineering services for the station design, and construction management services for the construction.

The reactor design core power level is 2,817 MWt which is also the design power level for the station and all components. This design power level is also the rated power level for which an operating license was issued. At this core power level of 2,817 MWt, the net station electrical output is 925 MWe.

The containment system for the station utilizes a free-standing containment vessel surrounded by a reinforced concrete shield building. The general arrangement for this containment was designed by Bechtel with the Chicago Bridge & Iron Company being responsible for the detail design and construction of the containment vessel. Bechtel provided the detail design of the foundation, substructures, and shield building.

The station's Construction Permit was granted on March 24, 1971 and the station's Facility Operating License No. NPF-3 was issued by the Nuclear Regulatory Commission on April 22, 1977. Initial fuel loading was completed on April 27, 1977. Following a period of testing, full commercial operation began on July 31, 1978.

1.2 GENERAL STATION DESCRIPTION

1.2.1 Site and Station Arrangement

1.2.1.1 Site

The station site is located on the southwestern shore of Lake Erie in Ottawa County, Ohio and consists of 954 acres of which approximately 733 acres is marshland which is leased to the U.S. Government as a national wildlife refuge. The land area surrounding the site is generally agricultural with no major industry in the vicinity. The topography of the site and vicinity is flat with marsh areas bordering the lake and the upland area rising to only 10 to 15 feet above the lake low water datum level in the general surrounding area. The site itself varies in elevation from marsh bottom, below lake level, to approximately six feet above lake level.

The site areas surrounding the station structures have been built up from 6 to 14 feet above the existing grade elevation to an elevation of 584 feet above sea level, International Great Lakes Datum (IGLD), or 15.4 feet above Lake Erie Low Water Datum of 568.6 feet IGLD. All elevations herein are referenced to International Great Lakes Datum (I.G.L.D.) 1955. This provides flood protection from the maximum credible water level conditions of Lake Erie. The north, east and a small portion of the south sides of the station area with exposure to the lake are provided with a dike to elevation 591 feet IGLD to protect the facility from wave effects during the maximum credible water level conditions. The south end of the wave protection dike was modified to provide protection for future nuclear installations.

The station structures are located approximately in the center of the site 3,000 feet from the shoreline, which provides a minimum exclusion distance of 2,400 feet from any point on the site boundary. The low population zone has been established as an area within a radius of two miles from the center of the containment structures. The Emergency Response Facility is located near the western end of the site to permit a portion of the building to be outside the restricted area as described in the DBNPS Emergency Plan.

The site is underlain by dolomitic limestone, 14 to 22 feet below the original site grade. The upper surface of this bedrock is susceptible to solution activity and solution fissures and small cavities are present on the site. An extensive bedrock verification program was conducted prior to construction of any structures to provide positive assurance that all conditions of the bedrock underlying the station structures are known and that competent foundation conditions exist for these structures.

1.2.1.2 Station Structures

The general arrangement of the station structures is shown on Figures 1.2-1 through 1.2-13.

1.2.2 Nuclear Steam Supply System

1.2.2.1 Principal Design Criteria

1.2.2.1.1 General

The reactor coolant system consists of the reactor vessel, two vertical once-through steam generators, for shaft-sealed coolant circulating pumps, an electrically heated pressurizer, and interconnecting piping. The system is arranged as two heat transport loops, each with two circulating pumps and one steam generator. Reactor Coolant System design data and a

system schematic diagram are shown in Chapter 5. Elevation and plan views of the arrangement of the major components are shown in Chapter 5.

The Reactor Coolant System is designed to contain and circulate reactor coolant at pressures and flows necessary to transfer the heat generated in the reactor core to the secondary fluid in the steam generators. In addition to serving as a heat transport medium, the coolant also serves as a neutron moderator and reflector, and as a solvent for the soluble boron utilized in chemical shim reactivity control.

1.2.2.1.2 Power Level

Licensed power for the reactor core is 2,817 MWt. Core performance analyses in this report are based on this power level. Postulated accidents that could release fission products to the environment have been analyzed on the basis of 2,817 MWt. An additional 17MWt is contributed to the cycle by the reactor coolant pumps, resulting in a net electrical output of about 925 MWe.

1.2.2.1.3 Peak Specific Power Level

A design peak specific power level in the fuel for operation at 2,817 MWt results in a maximum design nodal linear heat rate that is based on a total allowable design peak multiplied by the core average linear heat rate. Loss of Coolant Accident (LOCA) analysis based on maximum linear heat rates (core elevation and fuel type dependent) are specified in the Reload Report, Appendix 4B.

1.2.2.2 Operating Characteristics

The reactor core is designed to operate for a given number of full-power days for equilibrium cycles. The number of full-power days for each fuel cycle is specified in the Reload Report, Appendix 4B. The reactor coolant system contains and circulates reactor coolant at pressure and flows necessary to transfer the heat generated in the reactor core to the secondary fluid in the steam generators. The secondary fluid is completely separate from the reactor coolant and is used to transfer energy from the steam generators to the main turbine generator and auxiliary loads.

1.2.2.3 Safety Considerations

As the reactor coolant energy and radioactive material container, the Reactor Coolant System is designed to maintain its integrity under all operating conditions. While performing this function, the system serves the safety features objective of minimizing the release to the containment vessel of fission products that escape the primary barrier, the fuel cladding. All nuclear steam supply system components are designed and constructed to withstand the effects of earthquakes and loss-of-coolant accidents. Reactor and Reactor Coolant Systems are described in Chapters 4 and 5, respectively.

Chapter 15 is a safety evaluation summarizing the analyses which demonstrate the adequacy of the Reactor Protection System, and the engineered safety features systems. The consequences of various postulated accidents are within the guidelines set forth in the 10CFR100.

1.2.3 Engineered Safety Features and Emergency Systems

1.2.3.1 Principal Design Criteria

Chapter 6 presents design bases for safety features protection, equipment operational descriptions, design evaluation on equipment, failure analysis, and an operational testing program for systems used as engineered safety features. Compliance with IEEE 279-1971 is also covered in Chapter 7.

Applicable codes and standards for design, fabrication, and testing of components used as safety features are listed in Chapter 6, and seismic requirements are given in Chapter 3. The safety analyses presented in Chapter 15 demonstrate the performance of installed equipment in relation to functional objectives with assumed failures.

Some of the engineered safety features functions noted above are accomplished with the post-accident use of equipment serving normal functions. The design approach is based on the belief that regular use of equipment provides the best possible means for monitoring equipment availability and conditions.

1.2.3.2 Operating Characteristics

Engineered safety features are provided to fulfill three functions in the unlikely event of a serious loss-of-coolant accident:

a. Protect the fuel cladding.

b. Ensure containment vessel integrity.

c. Reduce the driving force for containment leakage.

Emergency injection of borated water to the Reactor Coolant System satisfies the first function above, while containment vessel atmosphere cooling satisfies the latter two functions. Each of these operations is performed by two or more systems which, in addition, employ multiple components to ensure operability. All equipment requiring electric power for operation is supplied by the emergency electric power sources as described in Chapter 8.

Equipment for the engineered safety features of the nuclear unit and its normal operation modes are listed in Table 1.2-1. The Emergency Core Cooling System (ECCS) includes the Core Flooding Tanks, High Pressure Coolant Injection, and Low Pressure Coolant Injection. Functional drawings of the ECCS are shown in Figures 6.3-1A, 6.3-2 and 6.3-2A.

Because some of the equipment used serves a normal function, the need for periodic testing is minimized. In cases where the equipment is used for emergencies only, the systems have been designed to permit meaningful periodic tests. Additional descriptive information and design details on equipment used for normal operation are presented in Chapter 9.

1.2.3.3 Safety Considerations

Engineered safety features are employed to reduce the potential radiation dose to the general public from the Maximum Hypothetical Accident (Chapter 15) below the guideline values of 10CFR100. The potential dose is reduced by immediate automatic isolation of all containment

vessel fluid penetrations not required for limiting the consequences of the accident, thereby eliminating potential leakage paths. Long-term potential releases following the accident are minimized by rapidly reducing the containment vessel pressure to near-atmospheric within 24 hours, thereby reducing the driving potential for fission product escape.

In addition, the engineered safety features prevent core meltdown, should the worst postulated loss-of-coolant accident occur. This is accomplished by large capacity coolant injection systems, parts of which are operated for normal purposes and are immediately available for emergency duty. These systems, coupled with thermal, hydraulic, and blowdown characteristics of the reactor, reliably prevent excessive metal-water reactions (or core disfiguration into a geometry which could prevent further cooling and allow core melting).

1.2.4 Instrumentation and Control Systems

1.2.4.1 Principal Design Criteria

1.2.4.1.1 Reactor Protection System

The Reactor Protection System was designed to meet the requirements of the IEEE “Criteria for Nuclear Power Plant Protection System” (IEEE 279) dated August 1968. Equipment was subjected to qualification tests as required by that document.

1.2.4.1.2 Safety Features Actuation System

The Safety Features Actuation System (SFAS) was designed to meet the requirements of IEEE 279-1971.

1.2.4.1.3 Steam and Feed Rupture Control System

The Steam and Feed Rupture Control System (SFRCS) was designed to meet the requirements of IEEE 279-1971.

1.2.4.1.4 Other Instrumentation and Control Systems

The principal design criteria for the Control Rod Drive System, Integrated Control Systems, Nuclear Instrumentation System, Non-Nuclear Instrumentation System and Incore Instrumentation System are discussed in Chapter 7.

1.2.4.2 Operating Characteristics

1.2.4.2.1 Reactor Protection System (RPS)

The RPS monitors parameters related to safe operation, and trips the reactor to protect the reactor core against fuel rod cladding damage. It also assists in protecting against Reactor Coolant System damage caused by high system pressure by limiting energy input to the system through reactor trip action.

1.2.4.2.2 Safety Features Actuation System (SFAS)

The engineered SFAS monitors variables to detect loss of Reactor Coolant System boundary integrity. Upon detection of “out of limit” conditions on any of these variables, it initiates operation of the High Pressure Injection (HPI), Low Pressure Injection (LPI), Containment

Vessel Cooling and Isolation, Containment Vessel Spray, and Emergency Ventilation Systems as dictated by plant conditions. Additionally, it starts the emergency diesel generators 1 and 2. Refer to Section 7.3.

1.2.4.2.3 Steam and Feed Rupture Control System (SFRCS)

The SFRCS monitors Steam Generator level, Steam Generator pressure, Steam Generator to Feedwater differential pressure and the power supply to the Reactor Coolant pumps to initiate Auxiliary Feedwater and to isolate the Steam Generators as necessary.

1.2.4.2.4 Control Rod Drive Control System (CRDCS)

The Control Rod Drive Control System provides for controlled withdrawal, controlled insertion, and holding of the control rod assemblies. These functions establish and maintain the power level required to meet the unit load demand at a given reactor coolant boron concentration.

1.2.4.2.5 Integrated Control System (ICS)

The ICS provides the proper coordination of the reactor, steam generator feedwater control, and turbine-generator under all operating conditions. Proper coordination consists of producing the best load response to the unit load demand while recognizing the capabilities and limitations of the reactor, steam generator feedwater system, and turbine-generator.

The ICS maintains constant average reactor coolant temperature between low level power limits and 100% rated power and constant steam pressure at all loads.

1.2.4.2.6 Nuclear Instrumentation System (NI)

The NI system is designed to supply the reactor operator with neutron information over the full operating range of the reactor. It also supplies reactor power information to the RPS and to the ICS.

1.2.4.2.7 Non-Nuclear Instrumentation System (NNI)

The NNI provides the required input signals of process variables for the reactor protection, regulating and auxiliary systems. It performs the required process control functions in response to those systems and provides instrumentation for startup, operation, and shutdown of the reactor system under normal and emergency conditions.

1.2.4.2.8 Incore Monitoring System (IMS)

The IMS provides neutron flux detectors and core outlet thermocouples to monitor core performance. Incore self-powered neutron detectors measure the neutron flux in the core to provide a history of power distribution during operation. The thermocouples measure the temperature of the RC leaving the top of the core to provide a record of core exit temperature. Data obtained provides power distribution information and fuel burn up data to provide assistance in fuel management. The station computer provides for normal system readout and a back-up readout system is provided for selected detectors.

1.2.4.3 Safety Considerations

1.2.4.3.1 Reactor Protection System (RPS)

Trip setpoints are consistent with the safety limits that have been established from the analyses described in Chapter 15. The setpoint for each input, which must initiate a trip of the RPS, was established at a level that ensures the control rods are inserted in sufficient time to protect the reactor core. Factors such as the rate at which the sensed variable can change, instrumentation and calibration inaccuracies, trip element trip times, circuit breaker trip times, and control rod travel times have been considered in establishing the margin between the trip setpoints and the safety limits that have been derived.

While operation on two reactor coolant pumps is provided for in the design of the reactor protection system, power operation with only two reactor coolant pumps running is not allowed by Davis-Besse License Condition C.3.a.

1.2.4.3.2 Safety Features Actuation System

The reactor coolant pressure, and containment vessel pressure, are the parameters to initiate safety features action. These parameters are shown in Chapter 15 to be the proper actuating parameters for the SFAS.

1.2.4.3.3 Steam and Feed Rupture Control System

SFRCS is designed to isolate the affected Steam Generator and to automatically start the Auxiliary Feedwater System in the event of a main steam line or main feedwater line rupture, to automatically start the Auxiliary Feedwater System on the loss of both main feed pumps or the loss of all four RC pumps, and to prevent steam generator overfill and subsequent spillover into the main steam lines. The SFRCS also provides a trip signal to the Anticipatory Reactor Trip System (ARTS).

1.2.4.3.4 Control Rod Drive Control System

The Control Rod Assemblies (CRA) are inserted into the core upon receipt of reactor protection system trip signals. Trip command has priority over all other commands. No single failure can inhibit the protective action of the Control Rod Drive Control Systems.

1.2.4.3.5 Other Instrumentation and Control Systems

Although the ICS, NNI, and IMS are not designed as safety systems, portions of these systems are used in the performance of safety functions as described in Chapter 7.

1.2.5 <u>Electrical Systems and Emergency Power</u>

1.2.5.1 Description

The generator output at 25kV is stepped down by a unit auxiliary transformer to 13.8kV to supply normal power to the station auxiliaries. Two startup transformers supply offsite power to station auxiliaries during startup, shutdown, and loss of normal power source. The station distribution system consists of essential (safety related) and nonessential (station auxiliaries) supplied by: 13.8kV buses, 4.16kV buses, 480V unit substations, 480 VAC motor control

centers, DC motor control centers, DC to AC power supply systems and regulated AC power supply systems. The station distribution system is described in Section 8.3.1 and depicted in electrical one line diagrams as listed in Section 1.5.3.1. On-site standby power is provided by two redundant emergency diesel generators each connected to its respective 4.16kV essential bus and one non-class 1E diesel generator which can be aligned to power either 4.16kV essential bus in the event of a station blackout. DC power is obtained from 2-250/125 VDC batteries. A complete description of the electrical power distribution system is given in Chapter 8.

1.2.5.2 Principal Design Criteria

a. Power Systems

The electrical power systems are designed to efficiently deliver the electrical power generated to the transmission systems.

b. Power Sources

Sufficient, reliable, redundant, adequate, and independent power sources are provided for handling all normal and emergency conditions.

c. Electrical Systems and Associated Equipment

All electrical systems and associated equipment important to safety are classified as IEEE-308 Class 1E and are designed to ensure that any design basis event does not cause:

1. A loss of electric power to a number of engineered safety features sufficient to jeopardize the safety of the station.

2. A loss of electric power to equipment that could cause significant damage to the fuel or reactor coolant system.

d. Variations of Voltage and Frequency

Variations of voltage and frequency in the Class 1E electric systems during any design basis event do not degrade the performance of any load important to safety.

e. Component Failure

No single component failure jeopardizes the operation of the required engineered safety features.

f. Electrical Systems Important to Safety

Electrical systems important to safety are designed to permit appropriate periodic testing and inspection.

g. Components of the Systems

All components of the systems are sized for operation under normal and emergency conditions. Electrical and physical separation of equipment related to safety are ensured.

1.2.5.3 Operating Characteristics

The main turbine-generator is designed to remain in operation upon a load rejection from full load down to auxiliary load.

Each of the two startup transformers is supplied from a different 345kV bus section and is the reserve source for one redundant emergency bus. If either startup transformer is out of service, the other startup transformer can supply power to all engineered safety features, as long as both bus tie transformers remain in service.

Upon loss of all onsite power sources a minimum of two offsite circuits are available in sufficient time to supply station auxiliaries. When a loss-of-coolant accident occurs, at least one of the three offsite circuits is available in a few seconds.

Fast automatic transfer schemes are provided between the normal and reserve sources.

Each emergency diesel generator is designed to start and reach stabilized voltage and frequency approximately ten seconds after receiving the starting signal. Upon receiving a safety feature actuation signal, the diesel generator is automatically isolated and loaded according to a predetermined sequence (See Chapter 8). All safety loads are assumed within 35 seconds including the 10-second starting interval.

The station batteries are maintained in a float condition. In the event all AC sources are lost, the batteries automatically supply DC power to DC essential loads and to inverters for essential 120 VAC loads.

1.2.5.4 Safety Considerations

The design of the electrical systems has been carried out with safety as a prime consideration. Reliable, redundant, and independent electric power sources are provided for all safety functions. These power sources are designed to be available immediately if called for by a safety features signal. Protection system channels are designed to maintain necessary functional capability under extremes of conditions relating to environment, malfunctions, power supply shifting and accidents. Channels are designed to be tested during power operation. Equipment and components related to safety are of a quality that is consistent with minimum maintenance requirements and low failure rates. Any single failure within the protection system does not prevent proper protective action at the system level when required.

For a list of safety functions and safety loads see Chapter 8.

1.2.6 <u>Power Conversion System</u>

1.2.6.1 Principal Design Criteria

The steam and power conversion system is designed to remove heat energy from the reactor coolant in two steam generators and convert it to electric energy via the turbine generator. The system is designed to accept 10 percent step load rejection without safety valve action or turbine bypass action. The turbine-generator equipment conforms to the applicable ANSI, ASME, and IEEE Standards. The main steam line piping design rating is 1050 psig at 600°F.

It meets the requirements of ASME Section III for Nuclear Class II piping from the steam generators up to and including the main steam isolation valves.

The steam and power conversion subsystems are designed to the applicable Codes and Standards of ASME, HEI, ASTM, SHI, and ANSI.

1.2.6.2 Operating Characteristics

The steam and power conversion system provides steam for driving the main turbine and the main feed pump turbines. Steam is also used for the auxiliary feed pump turbines, gland sealing, condenser inventory heating, steam jet air ejector, turbine reheater steam heating, building heating (steam supplied unit heaters), station heating heat exchangers and outdoor tank heating. The maximum expected gross turbine-generator capacity is 960 megawatts electrical. The turbine is operated as a combined reactor-following, turbine-following unit with the turbine header pressure setpoint varied in proportion to megawatt error between reactor and generator. The turbine governor valves change position to vary load and steam pressure. The system can be adjusted to permit controlled variations in steam pressure by limiting the effect of megawatt error on the steam pressure setpoint. This procedure can achieve any desired rate of turbine response to megawatt demand.

1.2.6.3 Safety Considerations

The features that support safety-related functions incorporated into the system design include those in the areas of loss of full load, turbine trip, overpressure protection, turbine overspeed protection, and turbine missile protection. Each of these areas are discussed in Section 10.1.3.

1.2.7 Fuel Handling and Storage

1.2.7.1 Principal Design Criteria

The Fuel Handling System, shown in Figures 9.1-7 and 9.1-8, is designed to provide a safe, effective means of transporting and handling fuel from the time it reaches the station in an unirradiated condition until it leaves the station after post-irradiation cooling. The system is designed and constructed to minimize the possibility of mishandling or maloperations that could cause fuel assembly damage and/or potential fission product release. Compliance with Safety Guide 13 is discussed in Appendix 3D.

The reactor is refueled with equipment designed to handle the spent fuel assemblies under borated water from the time they leave the reactor vessel until they are placed in a cask for onsite dry fuel storage or shipment from the site. Underwater transfer of spent fuel assemblies provides an effective, transparent radiation shield, as well as a reliable cooling medium for removal of decay heat. Borated water ensures subcritical conditions during refueling.

1.2.7.2 Operating Characteristics

1.2.7.2.1 Receiving and Storing Fuel

New fuel assemblies are received in shipping containers and stored in the new fuel storage area, which is a separate protected area for the dry storage of new fuel assemblies. The new fuel storage area is sized to accommodate the maximum number of new fuel assemblies required (80 assemblies). However, due to “mist” criticality considerations, rows C and F have been blocked and left vacant. Therefore, the new fuel storage area can effectively store only

60 new fuel assemblies. The new fuel assemblies are stored in racks in parallel rows having a center to center distance of 21 in. in both directions. This spacing, combined with the practice of keeping rows C and F vacant, is sufficient to maintain a k_{eff} of less than 0.95 even if flooded with unborated water, and is sufficient to maintain a k_{eff} of less than 0.98 when immersed in a hydrogenous “mist” that produces optimum moderation. With this geometry, fuel assemblies containing up to 5.00 wt% uranium-235 may be stored in the new fuel storage area.

1.2.7.2.2 Loading and Removing Fuel

New fuel assemblies are transferred from the new fuel storage area into the spent fuel pool area. They are then transferred into the containment vessel by the fuel transfer carriages operating through the fuel transfer tubes. Transfer of new fuel and removal of spent fuel occurs after the reactor is shut down and the refueling canal is filled with borated water.

Once refueling is completed, the refueling canal water is drained and pumped to the borated water storage tank.

1.2.7.2.3 Storage of Spent Fuel

After removal from the reactor and transfer to the fuel storage area in the auxiliary building, spent fuel is stored in the spent fuel storage pool.

The original design capacity of the spent fuel storage was for 260 elements. Prior to the first refueling, racks were installed to accommodate 735 spent fuel elements. In Cycle 13, the existing spent fuel pool storage racks were replaced with high-density fuel storage racks to provide for the plants long-term fuel storage requirements. These racks accommodate 1624 fuel assemblies. After a minimum required storage period in the spent fuel pool storage facility, fuel assemblies may be transferred to the onsite dry fuel storage facility.

A decontamination area is located in the building adjacent to the spent fuel storage pool. The outside surfaces of the casks can be decontaminated there before shipment.

1.2.7.3 Safety Considerations

Safety provisions are designed into the Fuel Handling System to prevent the development of hazardous conditions in the event of component malfunctions, accidental damage, or operational and administrative failures during refueling or transfer operations. Safety considerations are discussed in more detail in Section 9.1.

1.2.8 Cooling Water and Auxiliary Systems

1.2.8.1 Principal Design Criteria

The design, fabrication, and testing of the components and equipment in these systems are in accordance with the applicable codes of ASME, HEI, ASTM, SHI, and ANSI.

1.2.8.2 Operating Characteristics

1.2.8.2.1 Makeup and Purification System

The system supplies the Reactor Coolant System with fill and operational makeup water, circulates seal water for the reactor coolant pumps, and accommodates temporary changes in

the required reactor coolant inventory. It receives, purifies, and recirculates reactor coolant system letdown to provide water quality and reactor coolant boric acid concentration control. This system is not assumed to operate to mitigate USAR Chapter 15 Design Basis accidents.

1.2.8.2.2 Chemical Addition System

Chemical addition operations are required to alter the concentration of various chemicals in the reactor coolant and auxiliary systems. The system is designed to add necessary chemicals to the reactor coolant system for reactivity control, pH control, and oxygen control.

1.2.8.2.3 Cooling Water Systems

The cooling water systems remove heat from the station equipment to permit a sustained operation and safe shutdown of the station.

Condenser Circulating Water System:

The Condenser Circulating Water System is sized to handle the maximum condenser heat loads and consists of a closed system utilizing a hyperbolic natural draft cooling tower and the associated circulating water pumps, piping and valves. Fill and makeup water is taken from Lake Erie through the intake water system and intake structure. There are four circulating water pumps which take their suction from the cooling tower discharge channel, the water is pumped through the condenser and then back to the cooling tower.

Service Water System:

The Service Water System takes Lake Erie water from the intake structure pump suction pit after the traveling screens. It supplies normal and emergency cooling water makeup as described in Section 9.2.1. When the Control Room Emergency Ventilation System (CREVS) is in service, the Service Water System provides a source of cooling via the CREVS water-cooled condensers. The system consists of 3 pumps, heat exchangers and air coolers, associated valves, piping, instrumentation and controls. Additionally, the dilution pump can supply water to the service water system in the event of a fire disabling the service water pumps.

Component Cooling Water System:

This system is a closed loop system which provides cooling water to the nuclear and engineered safety features systems. It also acts as an intermediate barrier between the radioactive system and the Service Water System. The system consists of three circulating pumps, three heat exchangers, a surge tank, associated valves, piping, instrumentation, and controls.

Turbine Plant Cooling Water System:

The recirculated closed loop system furnishes purified and treated cooling water to main turbine and turbine plant pump oil coolers, various pump seals, generator hydrogen equipment auxiliaries including generator hydrogen coolers and stator liquid cooler, isolated phase bus, air compressor jackets and coolers, Auxiliary Boiler Forced Draft Fan bearings, and turbine plant sample coolers. The system consists of three pumps, a low level cooling water tank, three heat exchangers, a high level cooling water tank and associated valves, piping, instrumentation and controls.

The engineered safety features equipment is not dependent on the turbine plant cooling water system.

1.2.8.2.4 Spent Fuel Pool Cooling System

The spent fuel pool cooling system is designed to maintain the borated spent fuel pool water at 125°F or less with a heat load of 12.4 x 10^6 BTU/hr (refer to Section 9.1.3.1).

The unit is, however, designed such that if it becomes necessary at some time to off-load an entire core into the spent fuel pool, the cooling capacity can be provided by the decay heat removal system (refer to Section 9.1.3.3.2).

Based on heat load, both pumps and both heat exchangers may be in continuous operation; as the decay heat emitted by the spent fuel decreases one pump and one heat exchanger can be shut down. The spent fuel water temperature is normally maintained at 125°F or less.

During cold shutdown and refueling conditions, the reactor refueling canal is filled with water from the borated water storage tank.

1.2.8.2.5 Decay Heat Removal System

The normal function of this system is to remove reactor decay heat during the latter stages of cooldown and maintain reactor coolant temperature during refueling.

1.2.8.2.6 Sampling System

This system provides samples for laboratory analyses which serve to guide the operation of the following systems: reactor coolant; makeup and purification; chemical addition; and power conversion steam. These samples flow to central locations in the auxiliary and turbine buildings. Access to the containment vessel for this purpose is not required during power operation. Typical of the analyses performed on such samples are reactor coolant boron concentration, pH, fission product activity levels, dissolved gas content, corrosion product concentration and activity, and condensate activity. Analytical results are used for regulating boron concentration adjustments, evaluating the integrity of fuel rods and the performance of the demineralizers, and regulating chemical addition to the reactor coolant.

1.2.8.2.7 Station Ventilation Systems

The heating, ventilating, and air-conditioning systems are designed to provide a suitable environment for equipment and personnel. Equipment is arranged in zones so that potentially contaminated areas are separated from clean areas. The path of ventilating air in the auxiliary building is from areas of low activity toward areas of progressively higher activity.

The Containment Air Recirculation System and the Emergency Ventilation System are described in detail in Sections 9.4.6 and 6.2.3. The ventilation systems for the auxiliary building (including control room), turbine building, emergency diesel generator rooms, service water pump room, and various other areas are discussed in Section 9.4. The radiological aspects of various auxiliary building ventilation systems are discussed in Section 12.2.

1.2.8.2.8 Station and Instrument Air System

The Station and Instrument Air System provides a reliable continuous supply of dry, oil-free compressed air for pneumatic instrument operation and for control of pneumatic valves. The system also supplies air to service outlets throughout the station for operation of pneumatic tools or other requirements. The compressors supply air at a pressure of 100 psig with pressure reduced as necessary for the various service requirements.

1.2.8.2.9 Auxiliary Feedwater System

This system provides feedwater to the steam generators when the turbine driven main feedwater pumps are not available or following a loss of four reactor coolant pumps, to promote natural circulation of the Reactor Coolant System. A motor driven feedwater pump is installed as a non-safety related backup to the Auxiliary Feedwater pumps and to provide feedwater to the steam generator during plant startup, shutdown and operation as described in Section 10.4.7.2.

1.2.8.2.10 Fire Protection System

The Fire Protection System furnishes water to all points throughout the station area where it may be required for fire fighting.

Fire protection is provided by means of deluge systems, sprinklers, hose lines, portable extinguishers, instrumentation, and controls.

Two full capacity fire pumps (one electrical and one diesel) provide water for the system.

The system also provides back-up service for auxiliary feedwater pump suction and fill water to the emergency water storage tank.

1.2.8.2.11 Emergency Feedwater System

This system provides feedwater to the steam generators via the Auxiliary Feedwater System (AFWS) piping should the AFWS become unavailable. A diesel engine driven feedwater pump, and associated piping and valves, are installed in the Emergency Feedwater Facility and Auxiliary Building to support this function described in section 9.2.9.

1.2.8.3 Safety Considerations

1.2.8.3.1 Makeup and Purification System

This system provides essential functions for the normal operation of the unit. Redundant components and flow paths have been provided to improve system reliability.

The letdown line and the reactor coolant pump seal supply and return lines penetrate the containment vessel. These lines contain power-operated isolation valves which are automatically closed by the containment isolation signal.

Design and installation of the components and piping to the Makeup and Purification System considers the radioactive service of this system.

Alarms or interlocks are provided to limit variables or conditions of operation that might affect system or station safety.

1.2.8.3.2 Chemical Addition System

No containment isolation is required of this system since its boundaries do not penetrate the containment.

This system delivers additives to the spent fuel storage pool, the makeup tank, and borated water storage tank. Backflow from the tanks to the positive displacement pumps is prevented by check valves and normally closed shutoff valves between them. Additives to the spent fuel pool are delivered above the water level, thereby preventing backflow. Zinc injection may be added continuously within the parameters of the zinc injection program, backflow is prevented by the positive displacement pumps and the check valves when zinc injection is effected.

The boric acid mix and addition tanks are maintained at a sufficient temperature to ensure solubility of the boric acid. The amount of solution maintained in the boric acid addition tanks is sufficient to borate the reactor coolant system for hot shutdown at any time in core life.

1.2.8.3.3 Cooling Water Systems

Service Water System:

Three service water pumps are located in the intake structure to serve this system. Two pumps are used during normal operation. Pumps are sized such that a single pump can handle the essential cooling requirements following an accident.

Following a loss-of-coolant accident or loss of offsite electric power, the supply to the turbine plant cooling water system is automatically isolated, with all service water flow then being routed to the engineered safety features components.

In the event of the loss of offsite power, each of the two emergency diesel generators energizes its associated service water pump.

The intake structure, the service water pumps, and the containment air coolers are designed to Seismic Class I standards.

Adequate redundancy in equipment and piping is provided to guard against any single equipment or pipe failure preventing the mitigation of the accident.

Component Cooling Water System:

One component cooling water pump and one heat exchanger are required during normal station operation as well as following a loss-of-coolant accident.

Two component cooling pumps and two heat exchangers may be utilized during normal reactor shutdown when the Decay Heat Removal System is in service.

All the components in this system serving emergency functions are missile protected and designed for Seismic Class I standards.

In the event of the loss of offsite power, each of the two emergency diesel generators energize its associated pump.

The surge tank allows for expansion of the system and provides sufficient NPSH for the component cooling water pumps. Normal make-up is provided from the demineralized water source to maintain surge tank level. Provisions are made for the addition of chemicals for corrosion and pH control.

The operation of this system is monitored with instruments which measure inlet and outlet temperatures, system pressure, surge tank level, and radioactivity level. The piping is arranged so that following a loss-of-coolant accident, the cooling water to non-critical components can be isolated.

Spent Fuel Pool Cooling System:

The most serious failure of the system would be the complete loss of the spent fuel pool water. To protect against this possibility, all connections to the pool enter near to or above the water level, so the pool could not be gravity drained through leaking valves or piping.

In the event that a leaking fuel assembly is transferred from the refueling pool to the spent fuel pool, a small quantity of fission products may enter the spent fuel pool water. The bypass purification system is provided for removing these fission products and other contaminants from the water.

Decay Heat Removal System:

During reactor operation all equipment of the decay heat removal system is idle. During the accident condition, fission products are recirculated through the exterior piping system. To determine the total radiation dose released from this system, the potential leaks have been evaluated and discussed in Chapter 15.

Sampling System:

Gaseous leakage is collected by placing the sampling stations under hoods provided with off-gas vents to the radwaste area ventilation system. Liquid leakage from the valves in the hoods and sampling effluents are drained to the waste disposal system.

Station Ventilation Systems:

The Containment Purge System maintains a slight negative pressure in the shield building annulus during normal operations. The Emergency Ventilation System is designed to maintain a negative pressure in the shield building annulus following an accident. This minimizes the release of unfiltered radioactive particles from the containment vessel to the environment.

Fire Protection Systems:

This system is designed and constructed to meet all of the potential fire-fighting needs of the station. It is sufficiently redundant to allow a component failure and still provide protection to vital equipment.

1.2.9 Radioactive Waste Management

1.2.9.1 Principal Design Criteria

The radioactive waste system is designed to provide controlled handling and disposal of liquid, gaseous, and solid wastes from the station. The principal design criterion is to ensure that station personnel and the general public are protected against exposure to radioactive material in accordance with the regulations of 10CFR20.

1.2.9.2 Operating Characteristics

The various types of radioactive wastes to be handled are:

a. Liquid Wastes

    1. Clean liquid waste

    2. Miscellaneous liquid waste

    3. Detergent waste

b. Gaseous Waste

    1. Hydrogenated waste gases

    2. Aerated waste gases

c. Solid Wastes

The major sources of clean liquid waste are bleed-off of the reactor coolant during a reduction in reactor coolant boron concentration, an increase in coolant volume due to heat-up of the reactor system, and partial replacement of reactor coolant prior to refueling.

Liquid wastes other than from the reactor coolant system are considered as detergent wastes or as miscellaneous wastes and are collected separately. The detergent wastes may contain oil and detergents.

The sources of gaseous wastes are the reactor system vent, equipment, tank vents, purging from the sampling system, the degasifier (no longer used) the makeup tank, and the boric acid evaporators.

The sources of solid wastes are expected to be spent demineralizer resins, filter elements and/or pre-coat material, contaminated equipment, and paper, rags, plastic sheeting, etc. used in decontamination and contamination control.

Four methods are defined in the treatment of the radioactive wastes:

1. The clean liquid wastes consist of liquids such as reactor coolant, which are relatively low in chemical impurities and suspended solids content. Processing consists of storing, filtering, demineralizing and evaporating. The end products,

concentrated boric acid and demineralized water, are normally stored for later reuse in the reactor cycle.

2. The dirty liquid wastes consist of largely varying types and origins such as radioactive laboratory drains, building sumps, and decontamination drains. These liquids are relatively high in chemical impurities and suspended solids content but low in radioactivity. Normal processing consists of storage and filtration and demineralization. The end products are discharged from the station.

3. The gaseous wastes consist of the gaseous discharges from all potentially radioactive systems connected to the gaseous radwaste system. Processing consists of compression into decay tanks, retention, release through high efficiency filters, and discharge to the atmosphere through the station vent.

4. The solid wastes consist of all potentially radioactive solid wastes such as demineralizer resins, spent filter elements, clothing, and rags. Processing consists of storage and packaging, as appropriate, for later off-site disposal.

1.2.9.3 Safety Considerations

The possibility of an accidental release of activity from the radwaste system is minimized by reuse of much of the liquid wastes. Liquid and stored gaseous wastes are sampled prior to discharge to the environment.

Solid wastes are disposed of by licensed contractors in accordance with Department of Transportation (D.O.T.) regulations.

All liquid radioactive wastes flow to storage tanks prior to discharge to the environment and cannot be discharged to the environment by gravity (i.e., the effluent must be pumped out). All actuator operated valves which control the discharge of radioactive material into the environment fail in the closed position on loss of actuating force or signal.

Radioactive gases are continuously monitored during discharge in compliance with the requirements cited in the Offsite Dose Calculation Manual (ODCM).

Standby units (pumps, ion exchangers, and compressors) permit continuous processing in the event of equipment failures or routine maintenance.

1.2.10 Containment Systems

1.2.10.1 Principal Design Criteria

The containment for the station consists of two structures: A steel containment vessel and a reinforced concrete shield building, and their associated systems.

The Containment System is designed to provide protection for the public from the consequences of any break in the reactor coolant piping up to and including a double-ended break of the largest reactor coolant pipe assuming unobstructed discharge from both ends. Pressure and temperature behavior subsequent to the accident is determined by calculations evaluating the combined influence of the energy sources, heat sinks and engineered safety features.

The Containment System also provides protection for the public from the radiological consequences of a hypothetical accident discussed in Chapter 15. The containment design, along with the engineered safety features provided, ensure that the exposure to the public resulting from a hypothetical accident is below the guidelines established by 10CFR100.

The containment vessel design is in accordance with the ASME Boiler and Pressure Vessel Code 1968 through Summer 1969 Addenda, Section III, Class B. The “maximum internal pressure” as defined in Article N1311 of that code is 40 psig. The coincident temperature is 264°F. The “design internal pressure” as defined in that code is 36 psig. The coincident design temperature is 264°F.

The reinforced concrete shield building was designed in accordance with ACI 307-69, Specification for the Design and Construction of Reinforced Concrete Chimneys, and checked by the Ultimate Strength Design Method in accordance with ACI 318-63. Load combinations specified in ACI 307-69 provide the design basis of the shield building.

1.2.10.2 Operating Characteristics

The containment vessel, including all its penetrations, is a low leakage steel structure designed to withstand a postulated loss-of-coolant accident and to confine a postulated release of radioactive material. Systems directly associated with the containment vessel are the Containment Spray System, the Containment Air Cooling System, and the Containment Isolation System.

The shield building is a concrete structure surrounding the containment vessel. It is designed to provide biological shielding during normal operation and from hypothetical accident conditions. The building provides a means for collection and filtration of fission product leakage from the containment vessel following a hypothetical accident through the Emergency Ventilation System, an engineered safety feature designed for that purpose. In addition, the building provides environmental protection for the containment vessel from adverse atmospheric conditions and external missiles.

The containment vessel is a cylindrical steel pressure vessel with hemispherical dome and ellipsoidal bottom. It houses the reactor vessel, reactor coolant piping, pressurizer, pressurizer quench tank and coolers, reactor coolant pumps, steam generators, core flooding tanks, letdown coolers, and containment air cooling and recirculating systems. It is completely enclosed by a reinforced concrete shield building having a cylindrical shape with a shallow dome roof. An annular space is provided between the wall of the containment vessel and the shield building, and clearance is also provided between the containment vessel and the dome of the shield building.

With the exception of the concrete under the containment vessel there are no structural ties between the containment vessel and the shield building above the foundation slab. Above this there is virtually unlimited freedom for differential movement between the containment vessel and the shield building.

1.2.10.3 Safety Considerations

In addition to the pressure and temperature conditions specified, the containment vessel is designed to safely withstand the following loadings:

a. Structure dead load

b. Operating loads

c. Test pressure loads (containment vessel)

d. Seismic loads

e. Thermal stresses in the steel shell due to temperature gradients

The radiation shielding within the containment vessel is designed to minimize the exposure of station personnel to radiation emanating from the reactor and auxiliary systems. The radiation levels prevalent during the station operations, as well as those experienced upon shutdown, are considered in the determination of the shielding requirements.

The containment vessel is protected from external missiles by the shield building. Protection from pipe whip, jet impingement forces, and internal missiles is provided by the primary shield and other containment internal structures.

Engineered safety features systems are provided to minimize the consequences of postulated accidents by removing heat from the fuel and inserting negative reactivity into the reactor. They also decrease the pressure in the containment vessel by removing thermal energy and remove radioactive material that may leak into the shield building from the containment vessel.

1.2.11 Summary of the Emergency Plan and Facilities

The Emergency Plan and its associated Implementing Procedures have been established for coping with the various types of possible emergencies in an orderly and effective manner, in accordance with the requirements of 10CFR50.54.

The Emergency Plan shall be put into effect whenever a potentially hazardous situation or a radiological emergency as defined in the Emergency Plan is identified. The Emergency Plan establishes the concepts, evaluation and assessment criteria, and protective actions necessary to limit and mitigate the consequences of potential or actual radiological emergencies. The plan provides the necessary prearrangements, directions, and organization so that all station emergencies can be effectively and efficiently resolved. The details of the Emergency Plan Procedures are not included herein; but, the information contained within the plan includes a description of the procedures. This description is sufficient to demonstrate that the Emergency Plan provides assurance that appropriate actions can be taken by the company and other support agencies to protect station personnel and the general public during emergencies.

TABLE 1.2-1

Engineered Safety Features

	Function	Total Equipment Installed	Normal Operation Mode
1.	High Pressure Injection*	2 pumps	Normally Shutdown
2.	Core Flooding System	2 tanks	Self-operating when emergency conditions require it use. No external signal or power source required for operation.
3.	Low Pressure Injection*	2 pumps (decay heat removal) 2 heat exchangers	Normally operates for shutdown cooling as part of the Decay Heat Removal System.
4.	Containment Vessel Spray* System	2 pumps and 2 spray headers	Normally Shutdown
5.	Containment Vessel Cooling System	3 air recirculating and cooling units	2 units normally operating.
6.	Emergency Ventilation System	2 fan-filter assemblies	Normally shutdown
7.	Containment Vessel Isolation System	Discussed in Chapter 6	Operates on test or accident signal
8.	Containment Vessel	1 steel pressure vessel 1 concrete shield building	N/A
9.	Containment Emergency sump	1 sump with 2 suction paths	N/A
10.	Borated Water Storage Tank	1 storage tank	N/A
11.	Safety Features Actuation System	1 four sensing channel system	Operates on test or accident signal
12.	Hydrogen Dilution****	2 blowers	Normally shutdown
13.	Hydrogen Purge****	1 HEPA and charcoal filter assembly	Normally shutdown

TABLE 1.2-1 (Continued)

Engineered Safety Features

	Function	Total Equipment Installed	Normal Operation Mode
14.	Hydrogen Analyzers**	2 analyzers	Normally energized
15.	Hydrogen Recombiner***	See Chapter 6 for discussion	See Chapter 6 for discussion

* These systems receive their water supply from the Borated Water Storage Tank (BWST) which is discussed in Chapter 9.

** The hydrogen analyzers are normally energized to maintain the elements in a standby condition. The pumps are normally secured. 10 CFR 50.44 (Final Rule 68FR54123, effective October 16, 2003) relaxes the requirements for this equipment.

*** The capability to install an external hydrogen recombination system is no longer a requirement of 10 CFR 50.44 (Final Rule 68FR54123, effective October 16, 2003).

**** 10 CFR 50.44 (Final Rule 68FR54123, effective October 16, 2003) no longer requires this combustible control equipment.

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DAVIS BESSE NUCLEAR POWER STATION

NORTH ELEVATION
A-20
FIGURE 1.2-1

NOTES

1. FOR GENERAL NOTES, SEE DWG. A-1.
2. SEE NOTES ON DWG. NO. A-20.
3. HOLLOW METAL DOOR, PRESSED STEEL FRAME AND HARDWARE. DOOR SHALL BE 1 3/4 IN. THICK FULL FLUSH SEAMLESS CONSTRUCTION. 16 GA. STEEL SEPARATED BY 16 GA. CHANNEL STIFFENERS EXTENDING THE FULL HEIGHT OF DOOR, SPACED @ 6 IN. O.C. & WELDED TO BOTH FACES OF COVER SHEETS. THE TOP & BOTTOM EDGES OF DOOR SHALL BE CLOSED FLUSH BY 16 GA. STEEL CHANNEL, WELDED & EXTENDING THE FULL WIDTH OF DOOR. [redacted]

SHOP FACILITY

[redacted] SE NUCLEAR POWER STATION

EAST ELEVATION
A-22
FIGURE 1.2-2

REVISION 30
OCTOBER 2014

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DAVIS-BESSE NUCLEAR POWER STATION

WEST ELEVATION
A-23
FIGURE 1.2-3

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DAVIS BESSE NUCLEAR POWER STATION

GENERAL ARRANGEMENT GRADE PLAN
AT ELEVATION 585
M-104
FIGURE 1.2 4

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[redacted]

[redacted] NUCLEAR POWER STATION

[redacted] ARRANGEMENT PLAN
[redacted]LEVATION 603
M-103
FIGURE 1.2-5

REVISION 33
SEPTEMBER 2020

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"B" DWG. M-107

"D" DWG. M-108

1

11

[redacted]

N

FOR GENERAL NOTES SEE DWG. M-101.

REFERENCE DWG.
1. C-1500 PERSONNEL SHOP FACILITY.
2. C-1700 LOW LEVEL RADWASTE STORAGE FACILITY.

DAVIS-BESSE NUCLEAR POWER STATION

GENERAL ARANGEMENT TURBINE
FLOOR PLAN AT ELEVATION 623
M-102
FIGURE 1.2-6

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[redacted]

TES:

R KEY PLAN AND AREA DWGS.

OCATIONS SEE DWGS.

OULD BE WORKED WITH THE
NGEMENT DWGS.
AN-EL. 623'-0"
AN-EL. 603'-0"
AN-EL. 585'-0"
AN-EL. 565'-0"
AN-EL. 545'-0"
CTIONS 'A-A' AND 'B-B'
CTIONS 'C-C' AND 'D-D'

S-BESSE NUCLEAR POWER STATION
GENERAL ARRANGEMENT PLAN
@ EL. 643'-0"
M-101
FIGURE 1.2-7

REVISION 30

OCTOBER 2014

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DAVIS-BESSE NUCLEAR POWER STATION

GENERAL ARRANGEMENT BELOW GRADE PLAN AT ELEVATION 567

M105

FIGURE 1.2-8

Removed in Accordance with RIS 2015-17

DAVIS-BESSE NUCLEAR POWER STATION
GENERAL ARRANGEMENT BELOW GRADE
PLAN ELEVATION. 545'-0"
M-106
FIGURE 1.2-9

[redacted]

SE NUCLEAR POWER STATION

NERAL ARRANGEMENT
TIONS A-A AND B-B
(M-107)
FIGURE 1.2-10

REVISION 31
OCTOBER 2016

DB 03-21-16 DFN=/J/RASDGN/M107.DGN/TIF

[redacted]

E DWG. M-101

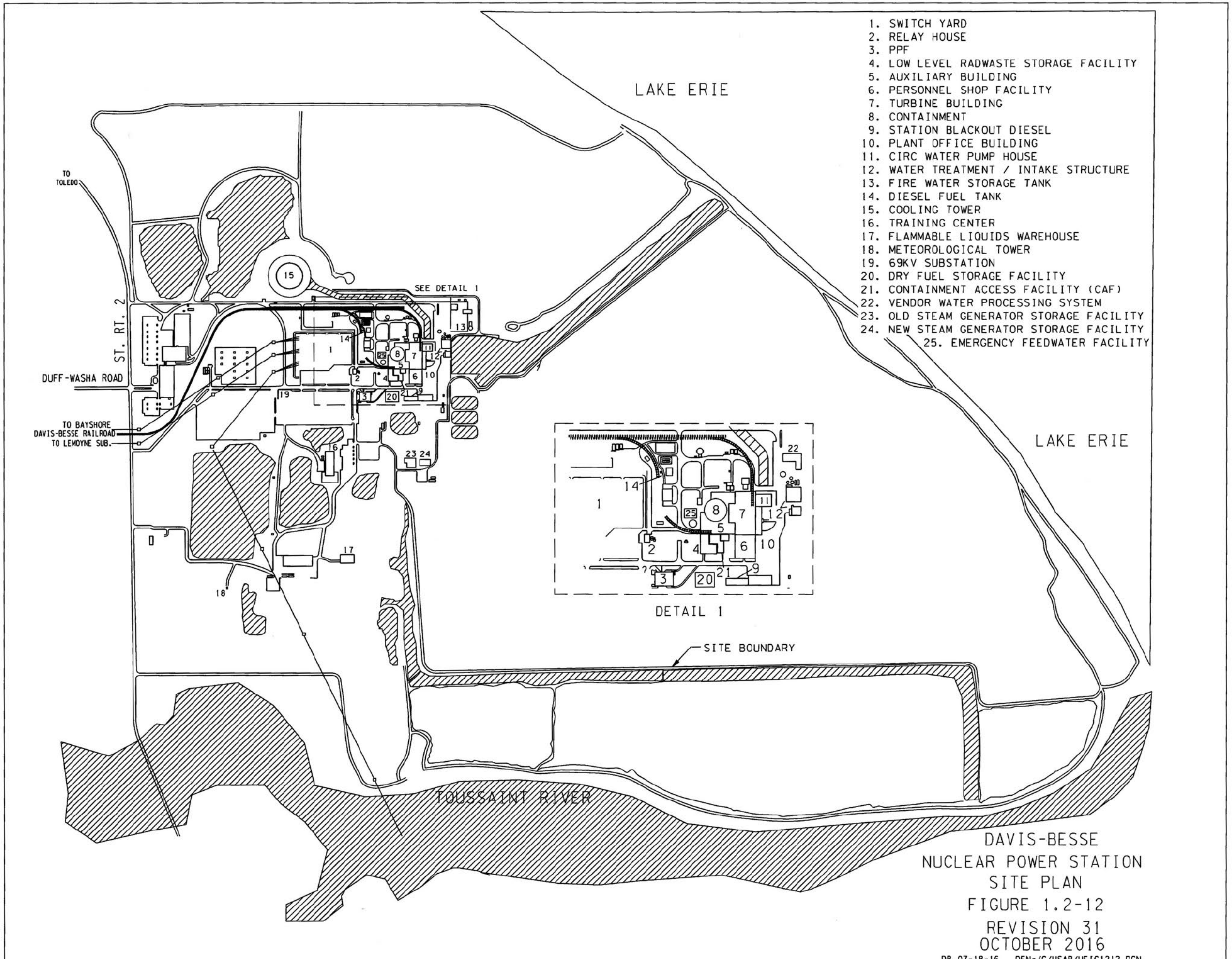
SECTION D-D

BESSE NUCLEAR POWER STATION

RRANGEMENT SECTIONS C-C AND D-D
(M-108)
FIGURE 1.2-11

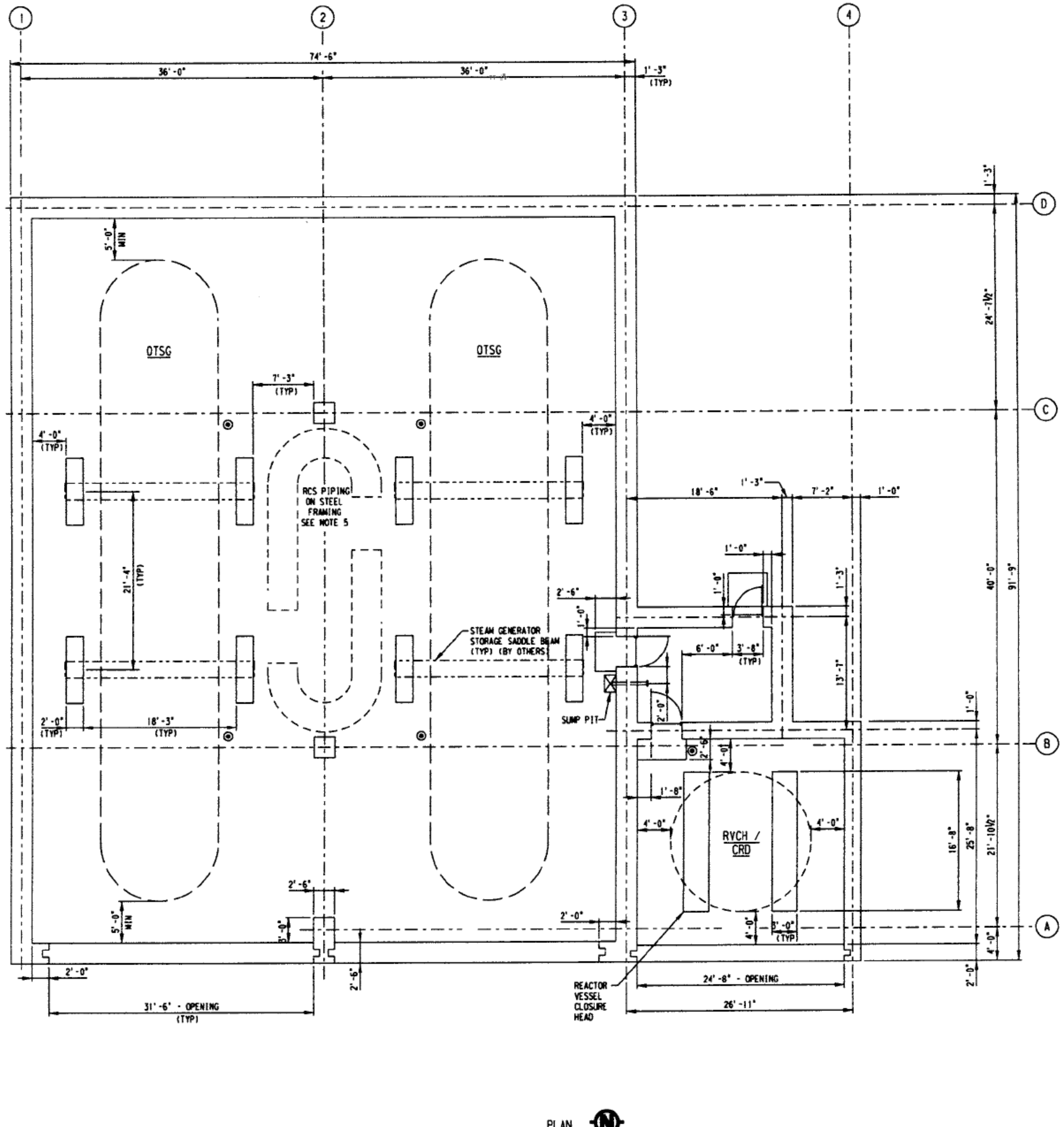
REVISION 31
OCTOBER 2016

DB 10-13-16 DFN-/J/MECH/M108.DGN/TIF



DAVIS-BESSE
NUCLEAR POWER STATION
SITE PLAN
FIGURE 1.2-12

REVISION 31
OCTOBER 2016



DAVIS-BESSE NUCLEAR POWER STATION
OLD STEAM GENERATOR STORAGE FACILITY
STRUCTURE - PLAN
FIGURE 1.2-13
REVISION 30
OCTOBER 2014

DB 07-18-14 DFN:G/USAR/UFIG51213.DGN

1.3 COMPARISON TABLES

1.3.1 Comparison with Similar Facility Design

A comparison of the principal similarities and differences between the initial design of Davis-Besse and Rancho Seco Unit 1 nuclear power station operated by the Sacramento Municipal Utility District (SMUD) is given in FSAR Section 1.3.

1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

1.4.1 General

The Davis-Besse Nuclear Power Station (DBNPS) is owned by Energy Harbor Nuclear Generation LLC. Energy Harbor Nuclear Corp. has exclusive responsibility and control over the operation and maintenance of the DBNPS. Energy Harbor Nuclear Generation LLC and Energy Harbor Nuclear Corp. are wholly owned subsidiaries of Energy Harbor Corp. References to previous owners and operators, the Toledo Edison company (TE), FirstEnergy Nuclear Generation, LLC, and FirstEnergy Nuclear Operating Company (FENOC) have been retained in the UFSAR, where appropriate, for historical purposes.

In the design of the facility, Toledo Edison retained the Bechtel Company to provide engineering services. Bechtel Power Corporation was retained to provide construction management services, and the actual site construction was accomplished through twelve major construction contracts and a number of smaller contracts. On-site materials testing services were done by an independent contractor. Other consultants have been retained by Toledo Edison to provide technical assistance for design, construction, and operation.

Babcock & Wilcox (B&W) furnished the nuclear steam supply system for the station and was responsible for the detailed design of the system and components. The original steam generators have been replaced with replacement steam generators manufactured by Babcock and Wilcox Canada Ltd.

1.4.2 Division of Responsibility

Bechtel was responsible to Toledo Edison for the overall design and arrangement of the station. They acted as Toledo Edison’s agent for quality assurance review of vendors and field construction. Bechtel also provided shop inspection and vendor expediting services for purchased items.

B&W was responsible to Toledo Edison for all contractual matters. In the detailed station design, Bechtel was responsible for integrating the B&W design into the overall station design and arrangement. Toledo Edison retained review responsibility for all aspects of design and arrangement.

The Bechtel field construction management organization administered, for Toledo Edison, all field contracts for construction work at the station and was responsible for proper integration of all contractor work. Bechtel was responsible to Toledo Edison for proper conduct of construction work by the contractors. Consultants retained by Toledo Edison for design aspects of the station were under the general technical direction of Bechtel.

1.4.3 Field Construction Contractors

The following listing sets forth the major organizations who performed field construction and the area of construction involved:

Contractor	Construction Work
The Great Lakes Construction Co.	Site preparation and foundation excavation.
Nicholson Concrete and Supply Co.	Central concrete mix plant.
Pittsburgh Testing Laboratory	Materials testing services.
The A. Bentley & Sons Company	General station structural.
Fegles-Power Service, Inc.	Containment shield building.
Chicago Bridge & Iron Company	Containment vessel.
Nooter Corporation	Fuel pool and canal liners.
ITT-Grinnell Corporation	Piping erection and setting of mechanical equipment.
The Babcock & Wilcox Company	Erection of the nuclear steam supply system.
Fishbach & Moore Inc. and Colgan Electric Company, Inc.	Electrical.
Research-Cottrell, Inc.	Cooling tower.

1.5 MATERIAL INCORPORATED BY REFERENCE

1.5.1 Reactor/Fuel Vendor Reports

NUMBER	TITLE	DATE	USAR SECTION
BAW-2303P Rev. 4	OTSG Repair Roll Qualification Report	August 2000	5.5.2.3
BAW-2120P Rev. 0	OTSG 80” Mechanical Sleeve Qualification (Alloy 690)	January, 1991	5.5.2.3
BAW-10001A	Incore Instrumentation Test Program	February, 1973	7.9
BAW-10003A Rev. 4	Qualification Testing of Protection System Instrumentation	January, 1976	7.1 7.2 3.10
BAW-10008 Part 1. Rev. 1	Reactor Internals Stress and Deflection Due to Loss-of-Coolant Accident and Maximum Hypothetical Accident.	June, 1970	4.2.2
BAW-10010 Part 1	Stability Margin for Xenon Oscillations – Modal Analysis	August, 1969	7.9
BAW-10016	Anticipated Transients Without Scram	September, 1972	15.1
BAW-10018	Analysis of the Structural Integrity of a Reactor Vessel Subjected to Thermal Shock	May 1969	3D
BAW-10019	Systematic Failure Study of Reactor Protection Systems	September, 1970	15.1
BAW-10021	TEMP-Thermal Enthalpy Mixing Program	April, 1970	4.4
BAW-10027	Once-Through Steam Generator Research & Development Report (BAW-1002. proprietary version)	April, 1971	5.3 5.5
BAW-10029A Rev. 3	Control Rod Drive Mechanism Test Program Revision 3 (BAW-10007, Rev. 1 proprietary version)	August 1976	4.2
BAW-10030	CRAFT—Description of Model for Equilibrium LOCA Analysis Program (Historical)	October 1971	6.2 15.1

1.5.1 Reactor/Fuel Vendor Reports (Continued)

NUMBER	TITLE	DATE	USAR SECTION
BAW-10034 Rev. 3	Multinode Analysis of B&W's 2568 MWT Nuclear Plants During a Loss-Of-Coolant Accident (Historical)	May 1972	15.1
BAW-10037 Rev. 2	Reactor Vessel Model Flow Tests (Nonproprietary Version of BAW-10012)	November 1972	3.9 4.4 5.3
BAW-10038 Rev. 1	Prototype Vibration Measurement Program for Reactor Internals-177-Fuel-Assembly Plant-	November 1972	3.9 App-3D
BAW-10038A Supp. 1	Prototype Vibration Measurement Program for Reactor Internals-177-Fuel Assembly Plant	June 1979	4.2
BAW-10039 Supp. 1	Prototype Vibration Measurement Results for B&W's 177-Fuel Assembly, 2-Loop Plant	April 1973	4.2
BAW-10039 Supp. 1, Rev. 1	Prototype Vibration Measurement Results for B&W's 177 Fuel Assembly, Two-Loop Plant	August 1977	3.9
BAW-10041	Fuel Assembly Stress Analysis for Seismic Excitation and Loss-of-Coolant Accident for B&W's Nozzle-Supported Vessel.	December 1972	3.9 4.2 4.4
BAW-10043	Overpressure Protection for B&W's Pressurized Water Reactors	May 1972	5.2
BAW-10046A Rev. 2	Methods of Compliance With Fracture Toughness and Operational Requirements of 10 CFR 50, Appendix G	June 1986	5.2
BAW-10051 Rev. 1	Design of Reactor Internals and Incore Instrument Nozzles for Flow Induced Vibration	November 1972	3.9 4.2
BAW-10051 Supp. 1	Structural Analysis of 177-FA Redesigned Surveillance Specimen Holder Tube	June 1979	4.2

1.5.1 Reactor/Fuel Vendor Reports (Continued)

NUMBER	TITLE	DATE	USAR SECTION
BAW-10056A	Radiation Embrittlement Sensitivity of Reactor Pressure Vessel Steels	August 1973	5.1
BAW-10068	KAPP4-Digital Computer Program for Solution of Reactor Kinetics and Primary System Pressure Response	June 1973	15.2 15.1
BAW-10069A Rev. 1	RADAR-Reactor Thermal and Hydraulic Analysis During Reactor Flow Coastdown	October 1974	15.2 15.1
BAW-10070	POWER TRAIN-General Hybrid Simulation for Reactor Coolant and Secondary System Transient Response	July 1973	15.2 15.1
BAW-10071A	SPLIT-Digital Steady State Flow Distributions Code for Various Primary System Combinations	September 1974	15.2 15.1
BAW-10073A Rev. 1	PUMP-Analog-Hybrid Reactor Coolant Hydraulic Transient Model	March 1976	15.2 15.1
BAW-10076P-A Rev. 2	CADD-Computer Application to Direct Digital Simulation of Transients in Water Reactors	December 1974	1.5
BAW-10084P Rev. 1	Program to Determine In-Reactor Performance of B&W Fuels Cladding Creep Collapse	October 1976	4.2
BAW-10084P Rev. 3	Program to Determine In-Reactor Performance of B&W Fuels-Cladding Creep Collapse	July 1995	App-4B
BAW-10091	B&W's ECCS Evaluation Model Report with Specific Application to 177-FA Class Plants with Lowered-Loop Arrangement	August 1974	15.4
BAW-10091 Supp. 1	Supplementary and Supporting Documentation for B&W's ECCS Evaluation Model Report with Specific Application to 177-FA Class Plants with Lowered-Loop Arrangement	December 1974	15.4

1.5.1 Reactor/Fuel Vendor Reports (Continued)

NUMBER	TITLE	DATE	USAR SECTION
BAW-10093	REFLOOD – Description of Model for Multinode Core Reflood Analysis	July 1974	6.2
BAW-10098 Rev. 1	Hsii, Y. H., Watson, C. E., Vasudevan, N., Busby, S. E., and Trent, R. L., "-CADDS-Computer Application to Direct Digital Simulation of Transients in PWRs With or Without Scram – Revision 1,"	January 1978	15.1 15.2
BAW-10099	Babcock & Wilcox – Anticipated Transients Without Scram Analysis	December 1974	15.1 15.2
BAW-10100A	Reactor Vessel Material Surveillance Program Compliance with 10CFR50; Appendix H for Oconee Class Reactors		5.2 5.4
BAW-10104	B&W's ECCS Evaluation Model	May 1975	6.3 6.5
BAW-10104P Rev. 5	B&W's ECCS Evaluation Model	April 1986	App-4B
BAW-10105 Rev. 1	ECCS Evaluation of B&W's 177-FA Raised Loop NSS	July 1975	6.3 15.1 15.4
BAW-10141	TACO2 – Fuel Pin Performance Analysis	August 1979	6.5
BAW-10141P-A Rev. 1	TACO2 – Fuel Pin Performance Analysis - Revision 1	June 1983	4.4 4.5 6.5
BAW-10147P-A	Fuel Rod Bowing in Babcock & Wilcox Fuel Designs	May 1983	4.5
BAW-10156-A	LYNXT – Core Transient Thermal-Hydraulic Program	February 1986	15.1
BAW-10162P-A	TACO3 – Fuel Pin Thermal Analysis Computer Code	November 1989	App-4B

1.5.1 Reactor/Fuel Vendor Reports (Continued)

NUMBER	TITLE	DATE	USAR SECTION
BAW-10164P-A Rev. 3, Rev. 4, and Rev. 6	RELAP5-MOD2-B&W - An Advanced Computer Program for Light Water Reactor LOCA and Non-LOCA Transient Analysis	November 2002	6.2 6.3 6.5 App-4B 15.1 15.4
BAW-10179P-A	Safety Criteria and Methodology for Cycle Reload Analysis	Latest Rev per App. 4B	3.9 4.2 4.3 4.4 App-4B
BAW-10192PA Rev. 0 and Supplement 1	BWNT Loss-of-Coolant Accident Evaluation Model for Once-Through Steam Generator Plants	July 1998	6.2 6.3 6.5 App-4B 15.1 15.4
BAW-10193P-A	RELAP5/MOD2-B&W For Safety Analysis of B&W Designed Pressurized Water Reactors	January 2000	6.2 15.1 15.2
BAW-10227P-A	Evaluation of Advanced Cladding and Structural Material (M5) in PWR Reactor Fuel	February 2000	App-4B 6.3 6.5 15.1 15.4

1.5.2 Bechtel Topical Reports

NUMBER	TITLE	DATE	USAR SECTION
BC-TOP-4-A	Seismic Analysis of Structures and Equipment for Nuclear Power Plants	November 1974	3.7
BN-TOP-2	Design for Pipe Break Effect	June 1974	3.6

1.5.3 Electrical Instrumentation and Control (EI&C) Drawings

1.5.3.1 Single Line Diagrams

E-1	Sh-1	Fig. 8.3-1	AC Electrical System
E-1	Sh-2	Fig. 8.3-2	AC Electrical System
E-4	Sh-1	Fig. 8.3-14	"E" Buses 480V Unit Substations
E-4	Sh-2	Fig. 8.3-15	"F" Buses 480V Unit Substations
E-4	Sh-3	Fig. 8.3-16	Water Treatment & Turbine Bldg. 480V Unit Substations
E-4	Sh-4	Fig. 8.3-17	480V Unit Substation Lighting Distribution
E-4	Sh-5	Fig. 8.3-18	PZR & Annulus Heaters 480V Unit Substation
E-6	Sh-1	Fig. 8.3-23	480V MCC (Essential)
E-6	Sh-2	Fig. 8.3-24	480V MCC (Essential)
E-6	Sh-3	Fig. 8.3-46	125/250 V DC MCC No. 1 (Essential)
E-6	Sh-4	Fig. 8.3-47	125/250 V DC MCC No. 2 (Essential)
E-7	-	Fig. 8.3-25	250/125 V DC & Instrumentation AC
E-9	-		240 VAC & 120 VAC MCC's (Essential)

1.5.3.2 DELETED

1.5.3.3 Safety Features Actuation System (SFAS) Diagrams

E-16	Sh-1	Fig. 7.3-1	SFAS Logic Diagram
E-16	Sh-2	Fig. 7.3-2	SFAS Signal Diagram
E-17B	7 sheets	Fig. 7.3-3 – 7.3-8	SFAS Actuated Equipment Tabulation

1.5.3.4 SFAS – Wiring Diagrams

Consolidated Controls Corporation Drawing No.		Bechtel Drawing No.	Title
S9N16-1	Sht 1	7749-E-30-13	SFAS, Schematic Diagram Channel 1
S9N16-1	Sht 2	7749-E-30-14	SFAS, Schematic Diagram Channel 1
S9N16-1	Sht 3	7749-E-30-15	SFAS, Schematic Diagram Channel 1
S9N16-1	Sht 4	7749-E-30-16	SFAS, Schematic Diagram Channel 1
S9N16-1	Sht 5	7749-E-30-33	SFAS, Schematic Diagram Channel 1
S9N16-2	Sht 1	7749-E-30-23	SFAS, Schematic Diagram Channel 2
S9N16-2	Sht 2	7749-E-30-24	SFAS, Schematic Diagram Channel 2
S9N16-2	Sht 3	7749-E-30-25	SFAS, Schematic Diagram Channel 2
S9N16-2	Sht 4	7749-E-30-26	SFAS, Schematic Diagram Channel 2
S9N16-2	Sht 5	7749-E-30-47	SFAS, Schematic Diagram Channel 2
S9N16-3	Sht 1	7749-E-30-34	SFAS, Schematic Diagram Channel 3
S9N16-3	Sht 2	7749-E-30-35	SFAS, Schematic Diagram Channel 3
S9N16-3	Sht 3	7749-E-30-36	SFAS, Schematic Diagram Channel 3
S9N16-3	Sht 4	7749-E-30-37	SFAS, Schematic Diagram Channel 3

1.5.3.4 SFAS – Wiring Diagrams (Continued)

Consolidated Controls Corporation Drawing No.		Bechtel Drawing No.	Titles
S9N16-3	Sht 5	7749-E-30-38	SFAS, Schematic Diagram Channel 3
S9N16-4	Sht 1	7749-E-30-27	SFAS, Schematic Diagram Channel 4
S9N16-4	Sht 2	7749-E-30-28	SFAS, Schematic Diagram Channel 4
S9N16-4	Sht 3	7749-E-30-29	SFAS, Schematic Diagram Channel 4
S9N16-4	Sht 4	7749-E-30-30	SFAS, Schematic Diagram Channel 4
S9N16-4	Sht 5	7749-E-30-48	SFAS, Schematic Diagram Channel 4
6N81		7749-E-30-17	Bistable #1
6N82		7749-E-30-19	Bistable #2
6N83		7749-E-30-6	Output Module Assembly
6N84		7749-E-30-4	Sam Logic Module Assembly
6N86		7749-E-30-5	Analog Module Assembly
6N87	Sht 1	7749-E-30-3	Sequencer Assembly
6N87	Sht 2	7749-E-30-3	Sequencer Assembly
9N16	Sht 1	7749-E-30-41	Safety Features Actuation System Cabinets
9N16	Sht 2	7749-E-30-42	Safety Features Actuation System Cabinets
9N16	Sht 3	7749-E-30-43	Safety Features Actuation System Cabinets
9N16	Sht 4	7749-E-30-44	Safety Features Actuation System Cabinets
1209786D		E-30-340	SFAS MOV Insert Type 1X Electrical Schematic
1209787D		E-30-341	SFAS SOV Insert Type 2F Electrical Schematic
1209788D		E-30-342	SFAS SOV Insert Type 2I Electrical Schematic
1209789D		E-30-343	SFAS Pump Insert Type 3X Electrical Schematic
1209790D		E-30-344	SFAS SAM Relay Board type 4I Outline
1209791D		E-30-345	SFAS Pump Jumper Board type 5X Outline
1209792D		E-30-346	SFAS Surveillance Insert Outline

1.5.3.5 SFAS – Location Drawings

Bechtel Drawing No.	Title
7749-M-122	Equipment Locations, Containment & Auxiliary Building Plan El. 603’ -0”
7749-M-123	Equipment Locations, Containment & Auxiliary Building Plan El. 585’ -0”
7749-M-124	Equipment Locations, Containment & Auxiliary Building Plan El. 565’ -0”
7749-M-567	Instrument Locations, Containment & Auxiliary Building Plan El. 603’ -0”
7749-M-568	Instrument Locations, Containment & Auxiliary Building Plan El. 585’ -0”
7749-M-581B	Control Room Panels, Engineered Safety Features Panels C-16 & C-17

1.5.3.6 Three Line Diagrams

E-27	Emergency Diesel Generator Relaying and Control

1.5.3.7 Elementary Diagrams

(Almost all sheets consist of multiple lettered sheets i.e., 1A, 1B, 2A, 2B, 2C, 3A, 3B, etc.)

E-30B	Index, Sh-1, Sh-3 – 11, 16	General Guides
E-31B	Index, 3, 4, 5, 8, 9, 12	Generator & Transformer Protection. Tripping Lock-Out Relay Circuits
E-34B	Index, 3-6, 9-14, 16, 17	4.16 kV FD Breakers
E-37B	Index, 1-5	Essential Unit Substations
E-44B	Index, 1-9, 12, 14, 15	Feedwater System
E-45B	Index, 1-11, 17	AFPT & MFPT Control & Auxiliaries
E-46B	Index, 1, 3, 4, 23, 32, 46	Steam & Condensate
E-48B	Index, 6-9, 11-15, 26-28, 30, 31, 33, 36, 44	Lake Water System
E-49B	Index, 18, 19, 19C, 20, 22, 50	Treated Water Systems
E-50B	Index, 3, 4, 7-12, 13, 15,17, 21, 23, 24, 25	Cooling Water Systems
E-52B	Index, 1, 2, 5, 6, 7, 14-19, 21-30, 32-36, 39, 40, 43, 45, 46, 49, 53, 57	Reactor Cooling Systems
E-56B	Index, 19, 24, 25	Radioactive Waste Systems
E-58B	Index, 1-8, 10-16, 18	Containment Ventilation System
E-60B	Index, 1-5, 14, 17, 18, 23, 25, 26, 33	Station Heating, Ventilation and Cooling System
E-62B	Index, 4, 5	Air & Nitrogen Supply Systems
E-64B	Index, 1, 2, 5, 7, 8, 9, 13-16, 22	Diesel Generators

1.5.3.8 Reactor Protection System (RPS) Diagrams

Fig. 7.2-1 - RPS Logic Diagram

1.5.3.9 RPS Schematics

Bailey Meter Co. Drawing No.	Bechtel Drawing No.	Title
E8047534	7749-M-536-7	Analog Logic Drawing Nuclear Instrumentation
D8047556	7749-M-536-29	Power Range Channel (Subassembly A), Part 1 of 2
D8047557	7749-M-536-30	Power Range Channel (Subassembly A), Part 2 of 2
D8047559	7749-M-536-32	Reactor Coolant Pump Monitor Channel (Subassembly A)
D8047560	7749-M-536-33	Reactor Coolant Flow Channel (Subassembly A)
D8047561	7749-M-536-34	Reactor Coolant Pressure Channel (Subassembly A)
D8047562	7749-M-536-35	Reactor Coolant Outlet Temperature Channel (Subassembly A)
C8047600	7749-M-536-71	Reactor Building High Pressure; (Subassembly A)
D8047563	7749-M-536-36	Reactor Trip, Module Interlock, Test Trip (Subassembly A) (Sheet 1 represents original Bailey Reactor Trip Module, and Sheet 2 represents an equivalent replacement Framatome Reactor Trip Module. Reference MOD 01-0031)
D8047558	7749-M-536-31	Source Range Channel (Subassembly A)
D8047564	7749-M-536-37	Power Distribution (Subassembly A)
D8047565	7749-M-536-38	Bus Bar Wiring (Subassembly A)
C8047598	7749-M-536-69	Selection Panel Outputs (Subassembly A)
D8047539	7749-M-536-12	External Connection Diagram (Subassembly A) Cabinet 1
D8047540	7749-M-536-13	External Connection Diagram (Subassembly A) Cabinet 2
D8047566	7749-M-536-39	Power Range Channel (Subassembly B), Part 1 of 2
D8047567	7749-M-536-40	Power Range Channel (Subassembly B), Part 2 of 2
D8047569	7749-M-536-42	Reactor Coolant Pump Monitor Channel (Subassembly B)

1.5.3.9 RPS Schematics (Continued)

Bailey Meter Co. Drawing No.	Bechtel Drawing No.	Title
D8047570	7749-M-536-43	Reactor Coolant Flow Channel (Subassembly B)
D8047571	7749-M-536-44	Reactor Coolant Pressure Channel (Subassembly B)
D8047572	7749-M-536-45	Reactor Coolant Outlet Temperature Channel (Subassembly B)
C8047601	7749-M-536-72	Reactor Building High Pressure (Subassembly B)
D8047573	7749-M-536-46	Reactor Trip, Module Interlock Test Trip (Subassembly B) (Sheet 1 represents original Bailey Reactor Trip Module, and Sheet 2 represents an equivalent replacement Framatome Reactor Trip Module. Reference MOD 01-0031)
D8047568	7749-M-536-41	Source Range Channel (Subassembly B)
D8047574	7749-M-536-47	Power Distribution (Subassembly B)
D8047575	7749-M-536-48	Bus Bar Wiring (Subassembly B)
D8047541	7749-M-536-14	External Connection Diagram (Subassembly B) Cabinet 1
D8047542	7749-M-536-27	External Connection Diagram (Subassembly B) Cabinet 2
D8047576	7749-M-536-49	Power Range Channel (Subassembly C), Part 1 of 2
D8047577	7749-M-536-50	Power Range Channel (Subassembly C), Part 2 of 2
D8047579	7749-M-536-52	Reactor Coolant Pump Monitor Channel (Subassembly C)
D8047580	7749-M-536-53	Reactor Coolant Flow Channel (Subassembly C)
D8047581	7749-M-536-54	Reactor Coolant Pressure Channel (Subassembly C)
D8047582	7749-M-536-55	Reactor Coolant Outlet Temperature Channel (Subassembly C)
C8047602	7749-M-536-73	Reactor Building High Pressure (Subassembly C)

1.5.3.9 RPS Schematics (Continued)

Bailey Meter Co. Drawing No.	Bechtel Drawing No.	Title
D8047583	7749-M-536-56	Reactor Trip, Module Interlock, Test Trip (Subassembly C) (Sheet 1 represents original Bailey Reactor Trip Module, and Sheet 2 represents an equivalent replacement Framatome Reactor Trip Module. Reference MOD 01-0031)
D8047578	7749-M-536-51	Intermediate Range Channel (Subassembly C)
D8047584	7749-M-536-57	Power Distribution (Subassembly C)
D8047585	7749-M-536-58	Bus Bar Wiring (Subassembly C)
D8047543	7749-M-536-15	External Connection Diagram (Subassembly C) Cabinet 1
D8047544	7749-M-536-16	External Connection Diagram (Subassembly C) Cabinet 2
D8047586	7749-M-536-59	Power Range Channel (Subassembly D) Part 1 of 2
D8047587	7749-M-536-60	Power Range Channel (Subassembly D) Part 2 of 2
D8047589	7749-M-536-62	Reactor Coolant Pump Monitor Channel (Subassembly D)
D8047590	7749-M-536-63	Reactor Coolant Flow Channel (Subassembly D)
D8047591	7749-M-536-64	Reactor Coolant Pressure Channel (Subassembly D)
D8047592	7749-M-536-75	Reactor Coolant Outlet Temperature Channel (Subassembly D)
C8047603	7749-M-536-74	Reactor Building High Pressure (Subassembly D)
D8047593	7749-M-536-65	Reactor Trip, Module Interlock, Test Trip (Subassembly D) (Sheet 1 represents original Bailey Reactor Trip Module, and Sheet 2 represents an equivalent replacement Framatome Reactor Trip Module. Reference MOD 01-0031)
D8047588	7749-M-536-61	Intermediate Range Channel (Subassembly D)

1.5.3.9 RPS Schematics (Continued)

Bailey Meter Co. Drawing No.	Bechtel Drawing No.	Title
D8047594	7749-M-536-66	Power Distribution (Subassembly D)
D8047595	7749-M-536-67	Bus Bar Wiring (Subassembly D)
D8047545	7749-M-536-17	External Connection Diagram (Subassembly D) Cabinet 1
D8047546	7749-M-536-18	External Connection Diagram (Subassembly D) Cabinet 2
D8047599	7749-M-536-70	Auctioneered Average Power
D8047596	7749-M-536-68	Rod Hold and H. V. Cut-Off
D8047538	7749-M-536-11	External Connection Diagram Inter Subassembly Wiring
30752F	7749-M-536-1	Reactor Protection System

1.5.3.10 Emergency Diesel Generator Schematic Diagrams

Bruce Gm. Dwg. No.		Bechtel Dwg. No.	Title
B157F02501	Sh. 2	12501-M-180Q-13	Engine Control Panel
B157F02501	Sh. 3	12501-M-180Q-14	Engine Control Panel
B157F02501	Sh. 4	7749-M-180-4 Sh.1 & 2	Alarm Panel & Contactors
B157F02501	Sh. 5	7749-M-180-10	Governor & Voltage Control

EMMCO Dwg. No.	Bechtel Dwg. No.	Title
368D262	7749-M-180-31	CVT Static Exciter Regulator

1.5.3.11 DELETED

1.5.3.12 Steam and Feedwater Rupture Control System (SFRCS) Drawings

Consolidated Controls Corporation Drawing No.	Toledo Drawing No.	Title
S9N124	E-30AQ-006	Schematic Logic Cabinet
S9N125	E-30AQ-012	Schematic Termination Cabinet
S6N341	E-30AQ-021	Schematic Field Buffer
S6N370-2	E-30AQ-024	Schematic Relay Driver Module
S6N314	E-30AQ-027	Schematic Alarm Output Module
KJH7316	E-30AQ-036	Logic Cabinet, Power Distrib. Diagram

1.5.4 DELETED

1.5.5 Davis-Besse Controlled Documents

1.5.5.1 Fire Hazards Analysis Report (FHAR)

1.5.5.2 Technical Requirements Manual (TRM)

1.5.5.3 Quality Assurance Program Manual

1.5.6 USAR System Functional Drawings

USAR Figure Number	Title
3.6-24	Turbine Plant Cooling Water System
5.1-2	Reactor Coolant System
6.3-1	Containment Spray System
6.3-1A	Core Flooding System
6.3-2	High Pressure Injection System
6.3-2A	Decay Heat Removal/Low Pressure Injection System
7.3-9	Nitrogen Supply System
7.4-1	CRDM Reactor Trip and Power Supply Configuration
7.8-1	Nuclear Instrumentation System (NI)
9.1-5	Spent Fuel Pool Cooling System
9.2-1	Service Water System
9.2-2	Component Cooling Water System
9.2-3	Deleted
9.2-4A	Demineralized Water System
9.2-5	Domestic Water System
9.2-7	Emergency Feedwater System
9.3-1	Station and Instrument Air System
9.3-3A	Primary and Post Accident Sampling System

1.5.6 USAR System Functional Drawings (Continued)

USAR Figure Number	Title
9.3-4	Station Drainage System
9.3-16	Makeup and Purification System
9.3-18	Chemical Addition System
9.4-1	Control Room Normal and Emergency H&V System
9.4-2	Auxiliary Building Non-Radioactive H&V Systems
9.4-7	Station Heating System
9.4-9	Auxiliary Building Radioactive H&V Systems
9.4-10	Turbine Building H&V System
9.4-11	Emergency Ventilation and Containment Hydrogen Dilution Systems
9.4-11A	Containment Gas Analyzing & Radiation Monitoring Systems
9.4-12	Containment Air Cooling, Purge and Recirculation Systems
9.5-1	Station Fire Protection System
9.5-8	Emergency Diesel Generator Auxiliary Systems
9.5-8A	Diesel Generator Air Start System
10.1-2	Auxiliary Steam System
10.1-2A	Auxiliary Steam System
10.3-1	Main Steam and Reheat System
10.4-1	Vacuum Systems
10.4-2	HP & LP Turbine Drains and Seal System
10.4-2A	Main Feedwater Pump Turbine Drains and Seal System
10.4-4	Circulating Water, Cooling Tower Makeup and Screen Wash Systems

1.5.6 USAR System Functional Drawings (Continued)

1.6 CONTROLLED DRAWING/USAR FIGURE CROSS-REFERENCE

The following tabulation lists controlled drawings which have been incorporated into the USAR.

Control Drawing Number	USAR Figure Number	Title
A11	12.1-1	Radiation Zones for Normal Operation – Elevation 545
A12	12.1-2	Radiation Zones for Normal Operation – Elevation 565
A13	12.1-3	Radiation Zones for Normal Operation – Elevation 585
A14	12.1-4	Radiation Zones for Normal Operation – Elevation 603
A15	12.1-5	Radiation Zones for Normal Operation – Elevation 623
A16	12.1-6	Radiation Zones for Normal Operation – Elevation 643
A17	12.1-7	Radiation Zones Tabulation Sheet 1
A18	12.1-8	Radiation Zones Tabulation Sheet 2
A19	12.1-9	Radiation Zones Tabulation Sheet 3
A20	1.2-1	North Elevation
A22	1.2-2	East Elevation
A23	1.2-3	West Elevation
C253	9.1-2	New Fuel Storage Rack
E1. Sh. 1	8.3-1	AC Electrical System One-Line Diagram

1.6 CONTROLLED DRAWING/USAR FIGURE CROSS-REFERENCE (Continued)

Control Drawing Number	USAR Figure Number	Title
E1. Sh. 2	8.3-2	AC Electrical System One-Line Diagram
E4 Sh. 1	8.3-14	“E” Buses 480V Unit Substations One-Line Diagram
E4 Sh. 2	8.3-15	“F” Buses 480V Unit Substations One-Line Diagram
E4 Sh. 3	8.3-16	Water Treatment & Turbine Bldg. 480V Unit Substations One-Line Diagram
E4 Sh. 4	8.3-17	480V Unit Substation Lighting Distribution One-Line Diagram
E4 Sh. 5	8.3-18	PZR & Annulus Heaters 480V Unit Substation One-Line Diagram
E6 Sh. 1	8.3-23	480V A.C MCC (Essential) One-Line Diagram
E6 Sh. 2	8.3-24	480V A.C MCC (Essential) One-Line Diagram
E6 Sh. 3	8.3-46	125/250V DC MCC No. 1 (Essential) One-Line Diagram
E6 Sh. 4	8.3-47	125/250V DC MCC No. 2 (Essential) Single-Line Diagram
E6 Sh. 5	8.3-48	480V AC MCC (Essential) One-Line Diagram
E7	8.3-25	250/125V DC and Instrumentation AC One-Line Diagram
E16 Sh. 1	7.3-1	Safety Features Actuation System – Logic Diagram
E16 Sh. 2	7.3-2	Safety Features Actuation System – Signal Diagram
E17B Sh. 1	7.3-3A	Safety Features Actuation System – Actuated Equipment Tabulation, Sheet 1 of 7
E17B Sh. 2	7.3-4	Safety Features Actuation System – Actuated Equipment Tabulation, Sheet 2 of 7

1.6 CONTROLLED DRAWING/USAR FIGURE CROSS-REFERENCE (Continued)

Control Drawing Number	USAR Figure Number	Title
E17B Sh. 3	7.3-5	Safety Features Actuation System – Actuated Equipment Tabulation, Sheet 3 of 7
E17B Sh. 4	7.3-6	Safety Features Actuation System – Actuated Equipment Tabulation, Sheet 4 of 7
E17B Sh. 5	7.3-7	Safety Features Actuation System – Actuated Equipment Tabulation, Sheet 5 of 7
E17B Sh. 6	7.3-8	Safety Features Actuation System – Actuated Equipment Tabulation, Sheet 6 of 7
E17B Sh. 7	7.3-3	Safety Features Actuation System – Actuated Equipment Tabulation, Sheet 7 of 7
E18 Sh. 1. 2 & 3	7.4-4	Steam and Feedwater Line Rupture Control System Logic Diagram
E19B Sh. 1	7.4-5	SFRCS – Actuated Equipment Tabulation Sheet 1
E19B Sh. 2	7.4-6	SFRCS – Actuated Equipment Tabulation Sheet 2
E28	7.4-8	Anticipatory Reactor Trip System Logic Diagram
E34B Sh. 3	8.3-5	4.16kV Feeder Breaker Bus Tie XFMR BD Brkr. ABDC1 Control
E34B Sh. 4	8.3-6	4.16kV Feeder Breaker Bus Tie XMFR Brkr. ABDC1 Control
E34B Sh. 5	8.3-7	4.16kV Feeder Breaker Buses C1, C2 Tie Brkr. AC 110 Control
E34B Sh. 6	8.3-8	4.16kV Feeder Breaker Buses C1, C2 Tie Brkr AC 110 Control
E34B Sh. 13	8.3-9	4.16kV Feeder Breakers – Bus C1 (D1) Tripping and Lockout Relays and Synchro Check Relays

1.6 CONTROLLED DRAWING/USAR FIGURE CROSS-REFERENCE (Continued)

Control Drawing Number	USAR Figure Number	Title
E34B Sh. 13A	8.3-4	4.16kV Feeder Breakers Bus C1 Tripping and Lockout and Sychro Check Relays
E34B Sh. 14	8.3-10	4.16kV Feeder Breaker – Bus C1 (D1) Voltage and Aux Relays
E34B Sh. 16	8.3-11	4.16kV Feeder Breaker – Essential Unit Substations E1 and F1 Control, Sheet 1
E34B Sh. 17	8.3-12	4.16kV Feeder Breaker – Essential Unit Substations E1 and F1 Control, Sheet 2
E37B Sh. 1	8.3-19	Essential Unit Substations E1 and F1 Incoming Feeder Circuit Breakers, Sheet 1
E37B Sh. 2	8.3-20	Essential Unit Substations E1 and F1 Incoming Feeder Circuit Breakers, Sheet 2
E37B Sh. 3	8.3-21	Essential Unit Substations Incoming Feeder Circuit Breakers, Sheet 3
E37B Sh. 4	8.3-22	Essential Unit Substations MCC Feeder Circuit Breakers, Sheet 4
E52B Sh. 24A	7.6-1	Decay Heat Normal Suction Valve, Sheet 1 of 4
E52B Sh. 24B	7.6-2	Decay Heat Normal Suction Valve, Sheet 2 of 4
E52B Sh. 24C	7.6-3	Decay Heat Normal Suction Valve, Sheet 3 of 4
E52B Sh. 24D	7.6-4	Decay Heat Normal Suction Valve, Sheet 4 of 4
E356 Sh. 12	8.3-35	Cable Spreading Room Fire Barriers
E356 Sh. 13	8.3-36	Cable Spreading Room Fire Barriers
E356 Sh. 14	8.3-37	Cable Spreading Room Fire Barriers
E362	8.3-31	Raceways and Grounding, Containment Details

1.6 CONTROLLED DRAWING/USAR FIGURE CROSS-REFERENCE (Continued)

Control Drawing Number	USAR Figure Number	Title
E366	8.3-32	Raceways, Containment El. 4 (603)
E370	8.3-33	Electrical Penetration Arrangement, Shield Bldg.
E378B Sh. 1, 2 & 3	8.3-34	Electrical Penetration Details
E-1042 Sh. 1 & 2	Table 8.3-1	Emergency Diesel Generator 1-1 Loading Table
E-1043 Sh. 1 & 2	Table 8.3-1	Emergency Diesel Generator 1-2 Loading Table
FSKM-EBB5-1H	3.6-40	Secondary Side Drain Piping From Steam Generator E24-1 Nozzle (A) to Drain Header
FSKM-EBB5-2H Sh. 1	3.6-41	Secondary Side Drain Piping From Steam Generator E24-1 Nozzle (B) to Drain Header
FSKM-EBB5-3H	3.6-42	Secondary Side Drain Piping From Steam Generator E24-1 Nozzle (C) to Drain Header
FSKM-EBB5-4H	3.6-43	Secondary Side Drain Piping From Steam Generator E24-1 Nozzle (D) to Drain Header
FSKM-EBB5-7H Sh. 1	3.6-44	Secondary Side Drain Piping From Steam Generator E24-1 Nozzle (G) to Drain Header
FSKM-EBB5-8H	3.6-45	Secondary Side Drain Piping From Steam Generator E24-1 Nozzle (H) to Drain Header
FSKM-EBB5-9H	3.6-46	Secondary Side Drain Piping From Steam Generator E24-1 Nozzle (J) to Drain Header
FSKM-EBB5-10H	3.6-47	Secondary Side Drain Piping From Steam Generator E24-1 Nozzle (K) to Drain Header
FSKM-EBB5-11H	3.6-48	Secondary Side Drain Piping From Steam Generator E24-1 Nozzle (L) to Drain Header
FSKM-EBB5-12H	3.6-49	Secondary Side Drain Piping From Steam Generator E24-1 Nozzle (M) to Drain Header

1.6 CONTROLLED DRAWING/USAR FIGURE CROSS-REFERENCE (Continued)

Control Drawing Number	USAR Figure Number	Title
FSKM-EBB5-15H Sh. 1	3.6-50	Secondary Side Drain Piping From Steam Generator E24-1 Nozzle (R) to Drain Header
FSKM-EBB5-16H Sh. 1	3.5-51	Secondary Side Drain Piping From Steam Generator E24-1 Nozzle (S) to Drain Header
FSKM-EBD61-3H	3.6-39	Secondary Side Drain Piping From 4" EBD-61 to Steam Generator Wet Layup Pump (P182-2)
FSK-M-EBD-83-1	3.6-12A	AFPT Main Steam Minimum Flow Line
HL-203A	3.6-10	Main Steam System – Containment Building Steam Generator 1-1
HL-203B	3.6-11	Main Steam System – Containment Building Steam Generator 1-2
HL-203F	3.6-12	Main Steam System – Supply to Auxiliary Feedwater Pump Turbine 1-1
HL-203H	3.6-13	Main Steam System – Supply to Auxiliary Feedwater Pump Turbine 1-2 and Exhaust
HL-203J	3.6-14	Main Steam System – Main Steam Crossovers
HL-207A	3.6-35	Steam Generator Drain System – Containment Building
HL-207C	3.6-15	Main Feedwater System Auxiliary and Containment Building
HL-230A	3.6-32	Pressurizer Relief System – Containment Building
HL-231A	3.6-27	Purification R.C. Letdown System – Containment Building
HL-233C	3.6-31	Decay Heat Removal System
HL-233E	3.6-28	High Pressure Injection System – Containment Building

1.6 CONTROLLED DRAWING/USAR FIGURE CROSS-REFERENCE (Continued)

Control Drawing Number	USAR Figure Number	Title
HL-234A	3.6-26	Low Pressure Injection – Core Flooding System
HL-234B	3.6-25	Containment Spray System – Containment Building
HL-1147 Sh. 1	3.6-36	Steam Generator Blowdown System – Containment Building Steam Generator 1-1 Drain System
HL-1147 Sh. 2	3.6-37	Steam Generator Blowdown System – Auxiliary and Turbine Building Steam Generator 1-1 Drain System
HL-1147 Sh. 3	3.6-38	Steam Generator Blowdown System – Auxiliary and Turbine Building Steam Generator 1-2 Drain System
HL-1155	3.6-17	Auxiliary Feedwater System – Auxiliary Feedwater Pump Discharge to Steam Generators
HL-1170	3.6-56A	Hanger Location Isometric R.V. Head to Hot Leg Vent Line – Containment Building
M050A	7.4-3	Main Steam Line and Main Feedwater Line Rupture Control System Logic
M050B	7.4-3A	Main Steam Line and Main Feedwater Line Rupture Control System Logic
M051	7.4-2	Auxiliary Feedwater Pump Turbine Start Control System Logic
M090	9.3-5	Schematic Diagram – Station Plumbing and Drains
M101	1.2-7	General Arrangement Plan at Elevation 643
M102	1.2-6	General Arrangement Turbine Floor Plan at Elevation 623
M103	1.2-5	General Arrangement Grade Plan at Elevation 603
M104	1.2-4	General Arrangement Grade Plan at Elevation 585
M105	1.2-8	General Arrangement Below Grade Plan at Elevation 567

1.6 CONTROLLED DRAWING/USAR FIGURE CROSS-REFERENCE (Continued)

Control Drawing Number	USAR Figure Number	Title
M106	1.2-9	General Arrangement Below Grade Plan at Elevation 545
M107	1.2-10	General Arrangement Sections A-A and B-B
M108	1.2-11	General Arrangement Sections C-C and D-D
M112	3.6-23	Turbine Building Plan El. 585’ -0”
M113	3.6-20	Turbine Building Partial Plans
M114	3.6-21	Turbine Building Section B-B
M115	3.6-22	Turbine Building Section C-C
M120	3.6-2	Containment and Auxiliary Buildings – Plan El. 643’-0”
M121	3.6-6	Containment and Auxiliary Buildings – Plan El. 623’-0”
M122	3.6-7	Containment and Auxiliary Buildings – Plan El. 603’-0”
M123	3.6-3	Containment and Auxiliary Buildings – Plan El. 585’-0”
M124	3.6-4	Containment and Auxiliary Buildings – Plan El. 565’-0”
M125	3.6-5	Containment and Auxiliary Buildings – Plan El. 545’-0”
M132	12.1-10	Equipment Location – Control Room and Computer Room
M133	12.1-11	Isometric of Control Room
M160	9.3-6	Symbols, Schedules and Details
M163	9.3-15	Drainage Systems – Turbine Building
M165	9.3-14	Drainage Systems – Piping Tunnel and Intake Structure
M170	9.3-8	Drainage Systems – Auxiliary Building
M171	9.3-9	Drainage Systems – Auxiliary Building

1.6 CONTROLLED DRAWING/USAR FIGURE CROSS-REFERENCE (Continued)

Control Drawing Number	USAR Figure Number	Title
M172	9.3-10	Drainage Systems – Auxiliary Building
M173	9.3-11	Drainage Systems – Auxiliary Building
M174	9.3-12	Drainage Systems – Auxiliary Building
M230B	3.6-33	ASME Section III Class I
M230C	3.6-34	ASME Section III Class I
M231B	3.6-16	Make-up and Purification System – Auxiliary Building
M246B	9.3-13	Sump Pump Discharge System
M410	6.2-34	Auxiliary Building Ventilation Equipment and Duct Layout, Plan at El. 623
M411	6.2-35	Auxiliary Building Ventilation Equipment and Duct Layout, Plan at El. 603
M412	6.2-36	Auxiliary Building Ventilation Equipment and Duct Layout, Plan at El. 585
M431	6.2-27	Containment Vessel Cooling System Plan, Sections and Details
M450	9.4-3	Heating, Ventilating and Air Conditioning – Control Room – Equipment Room Plan at Elevation 638
M451	9.4-4	Heating, Ventilating and Air Conditioning – Control Room Plan at Elevation 623
M453	9.4-5	Heating, Ventilating and Air Conditioning – Control Room Sections and Details
M519-9	9.1-8	Fuel Handling System (Section)
M536-1	7.2-1	Reactor Protection System
M592	7.4-9	Auxiliary Shutdown Panel (ASP)