CHAPTER 1

INTRODUCTION AND GENERAL DESCRIPTION OF PLANT

1.1 INTRODUCTION

•→7

This report is an update of the Final Safety Analysis Report (FSAR) which was originally submitted on April 22, 1981, in support of the application by Gulf States Utilities Company for a Class 103 license to operate a two-unit nuclear power station designated as River Bend Station - Units 1 and 2. A Class 103 construction permit for these units was issued on March 25, 1977. Unit 1 was completed and went commercial on June 16, 1986. Unit 2 was cancelled on January 5, 1984. The license was applied for under Section 103 of the Atomic Energy Act of 1954, as amended, and the regulations of the U.S. Nuclear Regulatory Commission (NRC) as set forth in Title 10 of the Code of Federal Regulations (CFR).

The applicant, Gulf States Utilities Company (GSU), was acting in behalf of itself and for Cajun Electric Power Cooperative (CEPCO).

GSU was acting as project manager for these owners and was responsible for the design, construction, and (until December 31, 1993) the operation of River Bend Station. Entergy Operations is now responsible for control and operation of River Bend Station.

7←•

River Bend Station (RBS) is on a site in West Feliciana Parish, Louisiana, located approximately 24 mi north-northwest of Baton Rouge, Louisiana. This site is just east of the Mississippi River, which is used as the source of the RBS major water requirements and which receives the RBS liquid discharges.

RBS includes a boiling water reactor nuclear steam supply system (NSSS) and a turbine-generator, both of which are furnished by General Electric Company (GE). The balance of the unit, which is similar in design concept to projects currently under review by the NRC, is designed and constructed by Stone & Webster Engineering Corporation (SWEC).

The containment is a steel structure in the form of a right circular cylinder with torispherical dome and flat bottom. Surrounding the containment is a reinforced concrete shield building. The shield building is a right circular cylinder with constant radius dome. The bottom portion of the annulus between the steel containment and shield building is filled with structural concrete that acts as a connecting element to tie the containment vessel and the shield building wall together to form a composite section. Above the concrete fill, the shield building is separated from the containment. This separation provides annular space between the two structures. The containment internal structures include a reinforced concrete drywell and suppression pool of the GE Mark III concept. The containment, including all internal structures, and the shield building are designed by SWEC.

•→14 •→12 •→10
The reactor for RBS is warranted for a core thermal power of 2,894 MWt. Reactor power output at rated plant operating conditions (Fig. 10.1-3) is 2,887 MWt, which corresponds to a net station electrical output of approximately 936 MWe. The reactor has a design core thermal power of 3,015 MWt (105 percent of reactor warranty steam flow exiting the vessel) for evaluating the design of components, systems, and structures in support of reactor operation. A core thermal power of at least 3,039 MWt (105 percent of reactor warranty power) is used for evaluating radiological consequences of design basis accidents. RBS has been analyzed to support operation at 105% of the original design as outlined in the preceding paragraphs. Subsequently, the Thermal Power Optimization project justified an additional 1.7% increase in licensed thermal power. This equates to full 100% plant operation at an uprated output of 3091 MWt. The heat balance for reactor 3091 MWt power is shown on Fig. 1.1-1.
10←• 12←• 14←•

This report has been organized according to the guidelines established by the regulatory staff of the NRC in their publication, Standard Format and Content of Safety Analysis Reports for Nuclear Power Plants (Revision 3, dated November 1978). This report is intended to be responsive to all existing NRC guides and regulations.

1.2 GENERAL PLANT DESCRIPTION

1.2.1 Principal Design Criteria

•→12

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

12←•

1.2.1.1 General Design Criteria

Some of the criteria are so general that they are applicable, at least in part, to more than one classification or more than one system group. These general criteria are presented below:

•→13

Note: River Bend Station administratively chose not to operate the recirculation flow control system in the master auto or flux auto modes. These modes allow the system to automatically respond and adjust reactor power due to changes in turbine load or neutron flux.

13←•

1. The station is designed, fabricated, erected, and operated to produce electric power in a safe and reliable manner. The station design conforms with applicable codes and regulations as described in this report.

2. The station is designed, fabricated, erected, and operated in such a way that the release of radioactive materials to the environment is limited to less than the limits and guideline values of applicable federal regulations, that pertain to the release of radioactive materials for normal operations and abnormal events.

3. The station employs a General Electric (GE) boiling water reactor to produce steam for direct use in a turbine-generator unit.

4. The station employs a GE nuclear steam supply system (NSSS).

5. Adequate strength and stiffness of components and structures with appropriate safety factors are provided, so that a release of radioactive materials to the environment does not exceed the limits and guideline values of applicable government regulations pertaining to the release of radioactive materials for normal operations and for abnormal transients and accidents.

6. Careful consideration is given to all known environmental conditions, such as earthquakes, floods, and storms, that could result in unplanned releases of radioactive material from the station. Adequate provisions are included in the station design to eliminate unacceptable results of these conditions.

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

   (The following general criteria apply to nuclear safety systems and engineered safeguards:)

8. Design margins for the nuclear safety systems and engineered safety features are conservative.

9. Nuclear safety systems respond to abnormal operational transients to limit fuel damage, so that, if the freed fission products are released to the environs via the designed discharge paths for radioactive material, the limits of 10CFR20 and 10CFR50 would not be exceeded.

10. Nuclear safety systems and engineered safety features act to ensure that no damage to the reactor coolant pressure boundary (RCPB) results from internal pressures caused by abnormal operational transients or accidents.

11. Where positive precise action is immediately required in response to accidents, such action is automatic and requires no decision or manipulation of controls by station operations personnel.

12. Essential safety actions are carried out by equipment of sufficient redundance and independence so that no single failure of active components can prevent the required actions. For systems or components to which IEEE-279 applies, single failures of passive electrical components are considered, as well as single failures of active components, in recognition of the higher anticipated failure rates of passive electrical components relative to passive mechanical components.

13. Provision is made for control of active components of nuclear safety systems and engineered safety features from the main control room.

14. Nuclear safety systems and engineered safety features are designed to permit demonstration of their functional performance requirements.

15. The design of nuclear safety systems and engineered safety features allow for environmental phenomena at the site.

16. Features of the station that are essential to the mitigation of accident consequences are designed, so they can be fabricated and erected to quality standards that reflect the importance of the safety function to be performed.

1.2.1.1.1 Power Generation Design Criteria

1. The station is designed to produce steam for direct use in a turbine-generator unit.

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

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

4. The fuel cladding, in conjunction with other station systems, is designed to retain integrity, such that any failures are within acceptable limits throughout the range of normal operational conditions and abnormal operational transients for the design life of the fuel.

5. The fuel cladding accommodates, without loss of integrity, the pressures generated by fission gases released from fuel material throughout the design life of the fuel.

6. Control equipment is provided to allow the reactor to respond automatically to load changes and abnormal operational transients.

7. Reactor power level is manually controllable.

8. Control of the reactor is possible from a single location.

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

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

11. The station is designed for routine continuous operation, whereby steam activation products, fission products, corrosion products, and coolant dissociation products are processed within acceptable limits.

1.2.1.1.2 Safety Design Criteria

1. The station is designed, fabricated, erected, and operated in such a way that the release of radioactive materials to the environment does not exceed the limits and guideline values of applicable government regulations pertaining to the release of radioactive materials for normal operations and for abnormal transients and accidents.

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

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

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

5. The design provides means by which station operators are alerted when limits on the release of radioactive material are approached.

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

7. Radiation shielding is provided and access control patterns are established to allow a properly trained operating staff to control radiation doses within the limits of applicable regulations in any mode of normal station operations.

8. Those portions of the nuclear system that form part of the RCPB are designed to retain integrity as a radioactive material containment barrier following abnormal operational transients and accidents. For accidents in which one breach in the RCPB is postulated, such breach does not cause additional breaches in the RCPB.

9. Nuclear safety systems and engineering safety features function to assure that no damage to the RCPB results from internal pressures caused by abnormal operational transients and accidents.

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

11. Essential safety actions are provided by equipment of sufficient redundance and independence that no single failure of active components or of passive components in certain cases in the long term prevents the required actions. For systems or components to which IEEE-279, "Criteria for Protection Systems for Nuclear Power Generating Stations," and/or IEEE-308, "Criteria for Class IE Electrical Systems for Nuclear Power Generating Stations," applies, single failures of either active or passive electrical components are considered in recognition of the higher anticipated failure rates of passive electrical components relative to passive mechanical components.

12. Provisions are made for control of active components of nuclear safety systems and engineered safety features from the main control room.

13. Nuclear safety systems and engineered safety features are designed to permit demonstration of their functional performance capabilities. The ability and the extent that systems can be tested during operation is discussed further in each individual system section.

14. The design of nuclear safety systems and engineered safety features includes allowances for natural environmental disturbances such as earthquakes, floods, and storms at the station site.

15. Standby electrical power sources are of sufficient capacity to power all nuclear safety systems and engineered safety features requiring electrical power concurrently.

16. Standby electrical power sources are provided to allow prompt reactor shutdown and removal of decay heat under circumstances where normal auxiliary power is not available.

17. A containment is provided that completely encloses the reactor system, drywell and suppression pool. The containment employs the pressure suppression concept.

18. It is possible to test primary containment integrity and leak tightness at periodic intervals.

19. A shield building is provided that completely encloses the primary containment. This shield building contains a system for controlling the release of radioactive materials from the primary containment, and also includes a capability for filtering radioactive materials collected in the annulus between the primary containment and the shield building.

20. The shield building is designed to act as a radioactive material barrier, if required, when the primary containment is open for expected operational purposes. The primary function of the shield building is, however, to provide missile protection for the primary containment.

21. The primary containment and shield building, in conjunction with other engineered safety features, limit radiological effects of accidents resulting in the release of radioactive material to the containment volumes to less than the prescribed acceptable limits.

22. Provisions are made for removing energy from the primary containment as necessary, to maintain the integrity of the containment system following accidents that release energy to the containment.

•→12

23. Piping that penetrates the primary containment and could serve as a path for the uncontrolled release of radioactive material to the environs is automatically isolated whenever such uncontrolled radioactive material release is imminent. Such isolation is performed in time to limit radiological effects to less than the specified acceptable limits.

12←•

24. Piping that penetrates the drywell and could serve as a path for the release of an amount of steam/water mixture sufficient to overpressurize the containment is automatically isolated whenever such an event might occur, or is permanently enclosed in a structure which prevents release to the containment.

25. Emergency core cooling systems (ECCS) are provided to limit fuel cladding temperature to less than the limits set forth in 10CFR50.46 in the event of a loss-of-coolant accident (LOCA).

    a. The ECCSs provide for continuity of core cooling over the complete range of postulated break sizes in the RCPB.

    b. The ECCSs are diverse, reliable, and redundant.

    c. Operation of the ECCSs is initiated automatically when required, regardless of the availability of offsite power supplies and the normal generating system of the station.

26. The main control room is shielded against radiation so that continued occupancy under accident conditions is possible.

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

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

29. Fuel storage facilities, under dry and flooded conditions, and handling equipment are designed to prevent inadvertent criticality and to maintain shielding and cooling of spent fuel.

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

31. Systems that have redundant or backup safety functions are physically separated and arranged such that any credible events causing damage to any one region of the reactor island complex has minimum prospect for compromising the functional capability of the designated counterpart system.

1.2.1.2 System Criteria

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

1.2.1.2.1 Nuclear System Criteria

1. The fuel cladding is designed to maintain integrity as a radioactive material barrier, such that any failures are within acceptable limits throughout the design power range. The fuel cladding is designed to accommodate, without loss of integrity, the pressures generated by the fission gases released from the fuel material throughout the design life of the fuel.

2. The fuel cladding, in conjunction with other plant systems, is designed to maintain integrity, such that any failures are within acceptable limits throughout any abnormal operational transient.

3. Those portions of the nuclear system that form part of the RCPB are designed to retain integrity as a radioactive material barrier during normal operation and following abnormal operational transients and accidents. For accidents in which one breach in the RCPB is postulated, such breach does not cause additional breaches in the RCPB.

4. Heat removal systems are provided in sufficient capacity and operational adequacy to remove heat generated in the reactor core for the full range of normal operational transients as well as for abnormal operational transients. The capacity of such systems is adequate to prevent fuel cladding damage.

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

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

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

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

1.2.1.2.2 Power Conversion Systems Criteria

Components of the power conversion systems are designed to perform the following basic objectives.

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

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

1.2.1.2.3 Electrical Power Systems Criteria

Sufficient normal auxiliary and standby sources of electrical power are provided to attain prompt shutdown and continued maintenance of the station in a safe condition under all credible circumstances. The power sources are adequate to accomplish all required essential safety actions under all postulated accident conditions.

1.2.1.2.4 Radwaste System Criteria

1. The gaseous and liquid radwaste systems are designed to limit the release of radioactive effluents from the station to the environs to the lowest practical values. Such releases as may be necessary during normal operations are limited to values that meet the requirements of applicable regulations including 10CFR20 and 10CFR50.

2. The solid radwaste disposal system is designed so that processing and offsite shipments are in accordance with all applicable regulations, including 10CFR20, 10CFR71, and 49CFR171 through 49CFR179, and DOT regulations as appropriate.

3. The system's design provides means by which station operations personnel are alerted whenever specified limits on the release of radioactive material may be approached.

1.2.1.2.5 Auxiliary Systems Criteria

1. Fuel storage facilities, under dry and flooded conditions, and handling equipment are designed to prevent criticality and to maintain adequate shielding and cooling for spent fuel. Provisions are made for maintaining the cleanliness of spent fuel cooling and shielding water.

2. Other auxiliary systems such as service water, cooling water, fire protection, heating and ventilating, communications, and lighting which are required for safe shutdown or to mitigate the consequences of an accident are designed to function during normal and/or accident conditions.

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

1.2.1.2.6 Nuclear Safety Systems and Engineered Safety Features Criteria

Principal design criteria for nuclear safety systems and engineered safety features are as follows:

1. These criteria correspond to criteria 1 through 16 in Section 1.2.1.1 and 9 through 16, 25, 27, and 28 in Section 1.2.1.1.2.

2. Standby electrical power sources have sufficient capacity to power all Class 1E and all engineered safety features requiring electrical power concurrently.

3. Standby electrical power sources are provided as necessary for support of all engineered safety feature functions (e.g., decay heat removal) under all circumstances where normal auxiliary power is not available.

4. In the event that the main control room is inaccessible, it is possible to bring the reactor from power range operation to a hot shutdown condition by use of controls and equipment that are available outside the main control room. Furthermore, station design includes the ability, in this event, for operators to bring the reactor to a cold shutdown condition from the hot shutdown condition from outside the main control room.

5. Backup reactor shutdown capability is provided independent of normal reactivity control provisions. This backup system has the capability to shutdown the reactor from any operating condition and subsequently to maintain the shutdown condition.

1.2.1.2.7 Process Control Systems Criteria

The principal design criteria for the process control systems are as follows.

1.2.1.2.7.1 Nuclear System Process Control Criteria

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

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

3. Control of the nuclear system is possible from a central location.

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

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

1.2.1.2.7.2 Power Conversion Systems Process Control Criteria

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

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

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

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

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

1.2.1.2.7.3 Electrical Power System Process Control Criteria

1. The Class 1E power systems are designed as a three Division system, with either Division 1 or 2 being adequate to safely shutdown the unit. Division 3 exclusively serves the high pressure core spray system.

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

3. Voltage relays are used on the emergency equipment buses to isolate these buses from the normal electrical system in the event of loss of offsite power and to initiate starting of the emergency diesel generators.

4. The emergency diesel generators are started and loaded automatically to meet the existing emergency condition.

5. Electrically operated breakers are controllable from the main control room.

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

1.2.1.2.8 Shielding and Access Control Criteria

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

2. The main control room is shielded against radiation so that occupancy is possible under accident conditions and whole body doses are less than that required by 10CFR50.67. Amendment 132 revised the design basis accident main control room dose limit requirements to incorporate the limits of 10CFR50.67. The limits of 10CFR50 Appendix A, General Design Criteria 19, also remain applicable to the RBS design basis.

1.2.1.3 Station Design Criteria

Certain station structures must remain functional and/or protect vital equipment and systems both during and following the most severe natural phenomena. These conditions are considered in the design and are investigated and defined in Chapters 2 and 3. Required combinations of environmental events, normal operating loads, and design accident loads for the structures are given in Section 3.8.

Structures are designed to withstand dead loads, live loads, seismic loads, wind loads, tornado loads, thermal loads, pressures, etc, as applicable in accordance with relevant codes and standards. Loading conditions, and combinations thereof, are determined by the function of the structure and its importance in meeting the station safety and power generation objectives.

1.2.1.4 Station Shielding Classification

The station shielding and radiation zone classifications are based upon personnel occupancy requirements in the various areas of the unit in order to limit personnel exposure to limits specified in 10CFR20, and other guidelines established by the regulatory agencies, and are described in Section 12.1.1. Section 12.1.2 discusses the shielding design basis.

1.2.2 Station Description

1.2.2.1 Site Characteristics

1.2.2.1.1 Location and Size of Site

The site, approximately 3,342 acres in size, is located in West Feliciana Parish on the east bank of the Mississippi River approximately 24 mi north-northwest of Baton Rouge, Louisiana.

1.2.2.1.2 Description of Plant Environs

The site is heavily wooded with several unnamed intermittent streams crossing and draining to either Grants Bayou on the east or Alligator Bayou on the west. There are a few residences along State Highway 965 near the northern property line, but the nearest town is St. Francisville, which had a 1978 population of 1,495 and is located 3 mi northwest of the site. The nearest industrial facility is the Crown Zellerbach Papermill located approximately 2 mi south of the site. The nearest airport offering regular commercial service is the Baton Rouge Metropolitan Airport in Baton Rouge, located approximately 19 mi southeast of the station.

1.2.2.1.3 Statement of Historical Significance

There is nothing of an historic nature which suffers from the construction of River Bend Station. This has been affirmed by the West Feliciana Historical Society and concurred with by the citizenry of West Feliciana Parish as represented by committee action of the Police Jury.

1.2.2.2 General Arrangement of Structures and Equipment

NOTE: Section 1.2 General Arrangement figures are considered "Historical Information." No further attempts will be made to update these figures when details reflected on the figures change. The figures will only be updated if there is a general change in the layout of RBS buildings and structures.

•→8

The principal buildings and structures associated with the unit include the primary containment structure, the shield building, the auxiliary building, the fuel building, the control building, the diesel generator building, auxiliary control building, the radwaste building, the turbine building, the water treatment building, the condensate demineralizer regeneration and offgas building, the makeup water pump structure, the circulating water pump structure, the normal service water cooling towers, the ultimate heat sink, and the instrument air/service air building.

•←8

These buildings and structures are founded upon suitable material for their intended application. Structures essential to the safe operation and shutdown of the plant are designed to withstand more extreme loading conditions than normally considered in conventional nonnuclear design practice. The buildings and internal structures so designated are designed to provide protection as required from tornadoes, earthquakes, and the failure of equipment producing flooding, missiles, and pipe whip. Additional discussions of design consideration are found in Chapter 3.

Location and orientation of the buildings on the site are shown on Fig. 1.2-1 and 1.2-2. The general arrangement of personnel access between buildings is shown on Fig. 1.2-3 through Fig. 1.2-8. The general arrangement of the major structures and equipment is shown on Fig. 1.2-10 through 1.2-44.

The primary containment structure, shown on Fig. 1.2-9 through 1.2-12, is a Seismic Category I structure which encloses the reactor coolant system (RCS), the drywell, the suppression pool, the upper fuel pool and refueling cavity, and some of the engineered safety feature systems and supporting systems. The functional design basis of the primary containment, including its penetrations and isolation valves, is to contain with adequate design margin the energy released from a design basis LOCA and to provide a barrier against the uncontrolled release of radioactivity to the environment.

The shield building, shown on Fig. 1.2-9 through 1.2-12, is a limited leakage Seismic Category I structure that completely encloses the primary containment structure. It is designed to withstand all design basis environmental events, including tornadoes. The primary function of the shield building is to provide missile protection for the primary containment. The shield building provides a boundary for the standby gas treatment system (SGTS) which maintains a negative pressure in the volume between the primary containment and shield building to ensure that leakage of radioactive materials from the primary containment is filtered prior to release to the environment in the unlikely event of a LOCA

The auxiliary building, shown on Fig. 1.2-13 through 1.2-19, is a Seismic Category I structure that contains engineered safety systems, a remote shutdown panel, and necessary auxiliary support systems. Redundant safety trains in the auxiliary building and all other areas of the plant are separated and protected so that a loss of function of one train will not prevent the other train from performing its safety function.

The fuel building, shown on Fig. 1.2-20 through 1.2-23, is a Seismic Category I structure that contains fuel storage and shipping equipment and necessary auxiliary support systems.

The control building, shown on Fig. 1.2-24 through 1.2-27, is a Seismic Category I structure in which many of the control and electrical systems, including required support systems directly related to safety or necessary for plant operations, are located.

The diesel generator building, shown on Fig. 1.2-28, is a Seismic Category I structure enclosing the three diesel generators and their associated equipment. Each diesel generator is in an individual room within the diesel generator building. These rooms are separated by fire walls.

The radwaste building, shown on Fig. 1.2-29 through 1.2-32, contains storage facilities and equipment for the treatment of radioactive liquid waste material. Space is provided for a separately licensed solid radioactive waste equipment contractor. A simplified seismic analysis is performed on the radwaste building; however, the building is not classified as a Seismic Category I structure.

•→8

Additional storage space for radioactive material and low level radwaste is provided in remote facilities located on power station property but outside of the plant protected area. The approximate locations (coordinates) of these facilities are shown in figure 1.2-2.

←•8

The auxiliary control building is classified as a non-Seismic Category I structure. This is a two-story structure located immediately south of the radwaste building and immediately west of the heater bay portion of the turbine building. This structure houses the control panels for water treatment, fire protection, liquid and solid radwaste, sanitary sewage, etc, in the auxiliary control room on the second level. The first floor houses the decontamination area, hot machine shop, and associated storage and office facilities.

The turbine building, shown on Fig 1.2-33 through 1.2-37, houses all equipment associated with the main turbine generator. Other auxiliary equipment is also located in this building.

The water treatment building, shown on Fig. 1.2-38, houses the equipment necessary to provide makeup water of reactor coolant quality and to provide an adequate supply of treated water for all station operating requirements.

The condensate demineralizer regeneration and off gas building shown on Fig. 1.2-39 and 1.2-40 houses the equipment associated with the condensate demineralizer system and the off gas system.

The makeup water pump structure, shown on Fig.1.2-41, houses two full-capacity motor-driven normal cooling tower makeup water pumps and related electrical equipment.

The circulating water pump structure, shown on Fig.1.2-42, houses the normal service water pumps and the circulating water pumps.

•→7

The circulating water system cooling towers, shown on Fig. 1.2-43, consist of four multi-cell cooling towers that provide the heat sink for the circulating water system.

7←• •→6

The service water cooling system cooling tower is a five cell mechanical draft cooling tower that provides the heat sink for the service water cooling system. The service water cooling system cools the normal service water system, and is shown on Figure 1.2-48.

6←•

The ultimate heat sink consists of a Seismic Category I combination mechanical draft standby cooling tower/pumphouse/ basin structure. The tower consists of four cells; each cell has an induced draft fan system. The cells are completely isolated from each other and have separate missile-protected inlet distribution piping systems. The standby service water pumphouse is shown on Fig.1.2-44.

•→8

The instrument air/service air building, shown on Fig. 1.2-2 & 1.2-49, is an open structure with concrete floor and steel roof and houses all equipment associated with instrument and service air systems.

←•8 •→12

A single facility is provided for the storage and supply of hydrogen and oxygen gases in support of the hydrogen water chemistry system. This facility is enclosed in a fenced area approximately 2,000 feet west of the station. The hydrogen supply system consists of a nominal 18,000 gallon cryogenic tank (mechanically restricted to less than 16,500 gallons), cryogenic pumps, gas compressor, atmospheric vaporizers and gas storage tubes to supply high pressure gas to the hydrogen water chemistry and generator cooling systems. The oxygen supply system consists of a 9,000 gallon cryogenic tank and atmospheric vaporizers to supply low pressure gas to the hydrogen water chemistry system. A cryogenic nitrogen tank and atmospheric vaporizer is included at this facility to provide nitrogen gas for purging hydrogen piping and pneumatic controls.

←•12

1.2.2.3 Nuclear System

•→14

The nuclear system includes a direct-cycle, forced circulation, GE boiling water reactor that produces steam for direct use in the steam turbine. A heat balance showing the major parameters of the nuclear system for the power condition is shown on Fig.1.1-1.

14←• •→8

1.2.2.3.1 Reactor Core and Control Rods

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

•←8

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

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

1.2.2.3.2 Reactor Vessel and Internals

The reactor vessel contains the core and supporting structures; the steam separators and dryers; the jet pumps; the control rod guide tubes; the distribution lines for the feedwater, core sprays, and standby liquid control; the in-core instrumentation; and other components. The main connections to the vessel include the steam lines, coolant recirculation lines, feedwater lines, control rod drive and in-core nuclear instrument housings, core spray lines, residual heat removal lines, standby liquid control line, core differential pressure line, jet pump pressure sensing lines, and water level instrumentation.

•→14
The reactor vessel is designed and fabricated in accordance with applicable codes for a pressure of 1,250 psig. The nominal operating pressure in the steam space above the separators is 1070 psia. The vessel is fabricated of low alloy steel and is clad internally with stainless steel (except for the top head nozzles, and nozzle weld zones which are unclad).
14←•
The reactor core is cooled by demineralized water that enters the lower portion of the core and boils as it flows upward around the fuel rods. The steam leaving the core is dried by steam separators and dryers located in the upper portion of the reactor vessel. The steam is then directed to the turbine through the main steam lines. Each steam line is provided with two isolation valves in series, one on either side of the containment barrier.

1.2.2.3.3 Reactor Recirculation System

The reactor recirculation system consists of two recirculation pump loops external to the reactor vessel. These loops provide the piping path for the driving flow of water to the reactor vessel jet pumps. Each external loop contains one high capacity motor-driven recirculation pump, two motor-operated maintenance valves, and one hydraulically operated flow control valve. The variable position hydraulic flow control valve operates in conjunction with a low frequency motor-generator set to control reactor power level through the effects of coolant flow rate on moderator void content.

The jet pumps are reactor vessel internals. The jet pumps provide a continuous internal circulation path for the major portion of the core coolant flow. The jet pumps are located in the annular region between the core shroud and the vessel inner wall. Any recirculation line break still allows core flooding to approximately two-thirds of the core height - the level of the inlet of the jet pumps.

1.2.2.3.4 Residual Heat Removal System

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

1. Removes decay and sensible heat during and after plant shutdown.

2. Injects water into the reactor vessel, following a LOCA, to reflood the core independent of other core cooling systems. This is discussed in Section 1.2.2.4.8, "Emergency Core Cooling Systems."

3. Removes heat from the containment following a LOCA, to limit the increase in containment pressure. This is accomplished by cooling and recirculating the suppression pool water (containment cooling).

1.2.2.3.5 Reactor Water Cleanup System

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

1.2.2.3.6 Nuclear Leak Detection System

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

1. Main steam lines
2. Reactor water cleanup system (RWCU)
3. Residual heat removal (RHR) system
4. Reactor core isolation cooling (RCIC) system
5. Feedwater system
6. ECCS systems
7. Miscellaneous systems.

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

1.2.2.4 Nuclear Safety Systems and Engineered Safety Features

1.2.2.4.1 Reactor Protection System

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

1.2.2.4.2 Neutron Monitoring System

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

1.2.2.4.3 Control Rod Drive System

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

1.2.2.4.4 Control Rod Drive Housing Supports

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

1.2.2.4.5 Control Rod Velocity Limiter

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

1.2.2.4.6 Nuclear System Pressure Relief System

A pressure relief system consisting of safety/relief valves mounted on the main steam lines is provided to prevent excessive pressure inside the nuclear system for operational transients or accidents.

1.2.2.4.7 Reactor Core Isolation Cooling System

The reactor core isolation cooling (RCIC) system provides makeup water to the reactor vessel when the vessel is isolated. The RCIC system uses a steam-driven turbine-pump unit and operates automatically in time and with sufficient coolant flow to maintain adequate water level in the reactor vessel for events defined in Section 5.4.6.1.

1.2.2.4.8 Emergency Core Cooling Systems (ECCS)

Four ECCSs are provided to maintain fuel cladding below the temperature limit in 10CFR50.46 in the event of a breach in the reactor coolant pressure boundary that results in a loss of reactor coolant. The systems are:

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

2. Automatic Depressurization System (ADS) - The automatic depressurization system rapidly reduces reactor vessel pressure in a LOCA situation in which the HPCS system fails to maintain the reactor vessel water level. The depressurization provided by the system enables the low pressure ECCS to deliver cooling water to the reactor vessel. The ADS uses some of the relief valves that are part of the nuclear system pressure relief system. The automatic relief valves are arranged to open on conditions indicating both that a break in the RCPB has occurred and that the HPCS system is not delivering sufficient cooling water to the reactor vessel to maintain the water level above a preselected value. The ADS is not activated unless either the LPCS or LPCI pumps are operating. This is to ensure that adequate coolant is available to maintain reactor water level after the depressurization.

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

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

1.2.2.4.9 Containment Systems

1.2.2.4.9.1 Primary Containment

The primary containment is of the Mark III design which incorporates the drywell/pressure suppression feature of previous BWR containment designs into a dry-containment type of structure.

In fulfilling its design basis as a fission product barrier in case of an accident, the Mark III containment is a low-leakage structure even at the elevated pressures that could follow a main steam line rupture or a recirculation line break.

The main features of the design include the following:

1. A drywell surrounding the reactor pressure vessel (RPV) and a large part of the RCPB.

2. A suppression pool that serves as a heat sink during normal operational transients and accident conditions.

3. A containment upper pool for shielding and refueling operations.

4. A steel containment structure.

1.2.2.4.9.2 Shield Building

A shield building completely encloses the steel primary containment structure and serves as a secondary containment. The bottom portion of the annulus between the steel containment and shield building is filled with structural concrete. The annular space above the concrete fill is normally maintained at slightly below ambient atmospheric pressure. In the event of a design basis LOCA, pressure continues to be subatmospheric, and any leakage from the primary containment into the annulus is collected and passed to the SGTS. In this manner, offsite doses are maintained within the requirements of 10CFR50.67. Amendment 132 revised the design basis accident offsite dose limit requirements from 10CFR100 to 10CFR50.67.

Normal personnel access to the primary containment from outside the shield building is through interlocked doors at either end of an air lock, which passes through and is sealed from the shield building annulus.

The shield building is designed to withstand the safe shutdown earthquake (SSE) and protects the primary containment from the postulated design basis environmental events, such as tornado-generated winds and missiles.

1.2.2.4.9.3 Residual Heat Removal System (Containment Cooling)

The containment cooling subsystem is placed in operation to limit the temperature of the water in the suppression pool and of the atmospheres in the drywell and suppression chamber following a design basis LOCA to control the pool temperature during normal operation of the safety-relief valves and the RCIC system, and to reduce the pool temperature following an isolation transient. In the containment cooling mode of operation, the RHR main system pumps take suction from the suppression pool and pump the water through the RHR heat exchangers where cooling takes place by transferring heat to the service water. The fluid is then discharged back to the suppression pool.

1.2.2.4.9.4 Combustible Gas Control

•→12 •→8

In the unlikely event of a LOCA, hydrogen and oxygen are generated in the drywell and containment. The combustible gas control system ensures that hydrogen concentrations are kept below the limits specified in Regulatory Guide 1.7, Rev. 2. The systems provided include a hydrogen mixing system, a hydrogen recombiner system, a hydrogen ignition system, and a backup primary containment hydrogen purge system.

8←• 12←•

1.2.2.4.10 Containment and Reactor Vessel Isolation Control System

The containment and reactor vessel isolation control system automatically initiates closure of isolation valves to close off all process lines which are potential leakage paths for radioactive material to the environs. This action is taken upon indication of a breach in the RCPB.

1.2.2.4.10.1 Main Steam Isolation Valves

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

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

•→12

2. Deleted

•←12

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

1.2.2.4.10.2 Main Steam Flow Restrictors

A venturi-type flow restrictor is installed in each steam line. These devices limit the loss of coolant from the reactor vessel before the main steam isolation valves are closed in case of a main steam line break outside the containment.

1.2.2.4.11 Radiation Monitoring System

1.2.2.4.11.1 Main Steam Radiation Monitoring System

•→8

The main steam radiation monitoring system consists of two gamma radiation monitors located externally to the main steam lines just outside the containment. The monitors are designed to detect a gross release of fission products from the fuel.

8←•

1.2.2.4.11.2 Ventilation Exhaust Radiation Monitoring System

The ventilation exhaust radiation monitoring systems consist of a number of radiation monitors arranged to monitor the activity level of the air exhaust from the containment and drywell, auxiliary building, fuel handling and pool sweep areas, and main control room.

1.2.2.4.12 Standby Gas Treatment System

The SGTS processes exhaust air from various plant systems to limit the release of radioactivity and keep offsite dose rates below the limits specified in 10CFR50.67. Amendment 132 revised the design basis accident offsite dose limit requirements from 10CFR100 to 10CFR50.67.

•→13 •→12
The system is automatically placed in operation during the design basis accident (DBA), on low annulus pressure control system flow, and on hi-hi gaseous radiation signals from the reactor building annulus ventilation.
12←• 13←•
The SGTS consists of two identical, parallel, physically separated air filtration assemblies. Each assembly is capable of handling the maximum design flow rate.

1.2.2.4.13 Auxiliary and Fuel Building Ventilation Systems

The auxiliary and fuel building ventilation systems automatically initiate closure of isolation valves on selected lines that penetrate the buildings to preserve the integrity of the standby gas treatment boundary. This action is taken upon indication of a breach in the RCPB.

1.2.2.4.14 Safety-Related Electrical Power Systems

Standby ac power for each unit is supplied from three diesel engine generators. Each of the three generators is arranged for connection to one of three independent and segregated 4.16-kV switchgear assemblies which supply ac power required for a safe shutdown.

Power supplies to safety-related equipment are arranged so that alternate or redundant systems are supplied from separate 4.16-kV switchgear assemblies. With this arrangement, failure of any diesel generator or any switchgear assembly does not jeopardize proper operation of redundant systems supplied from other switchgear assemblies. Under this condition, adequate ac power is available for safe shutdown of the unit under all postulated accident conditions.

A 125-V dc system is provided for circuit breaker controls, dc auxiliary motors, normal and standby switchgear controls, diesel generator controls, and other essential control systems. It consists of five independent storage battery systems with associated distribution panels, battery chargers, etc. Three of the five battery systems and their associated chargers supply control power to the three 4.16-kV standby ac power systems. With this arrangement, failure of any battery system does not jeopardize proper operation of the others. The other two of the five battery systems serve the normal switchgear, certain inverters, standby lighting, annunciator service through an inverter, and auxiliary motors. Manual transfer of load between battery systems is possible when required.

1.2.2.4.15 Standby Liquid Control System

Although not intended to provide prompt reactor shutdown, as the control rods are, the standby liquid control system provides a redundant, independent, and alternate way to bring the nuclear fission reaction to subcriticality and to maintain subcriticality as the reactor cools. The system makes possible an orderly and safe shutdown in the event that not enough control rods can be inserted into the reactor core to accomplish shutdown in the normal manner. The system is sized to counteract the positive reactivity effect from rated power to the cold shutdown condition. The Standby Liquid Control System sodium pentaborate solution also functions to control suppression pool pH following a design basis LOCA event with no functioning ECCS injection. This function was added to the Standby Liquid Control System in conjunction with the River Bend implementation of Alternate Source Term (AST) per Regulatory Guide 1.183.

1.2.2.4.16 Safe Shutdown from Outside the Main Control Room

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

1.2.2.4.17 Main Steam Positive Leakage Control System

•→12

The main steam positive leakage control system (MS-PLCS) is designed to minimize the release of fission products which could bypass the SGTS after a LOCA. This is accomplished by pressurizing the piping between the inboard and outboard MSIVs, and by pressurizing the piping between the seals of the outboard MSIV and the main steam shutoff valve. Drain lines from the MSIV bodies are also pressurized from the first isolation valve outside the containment back to the MSIVs. The pressure, which is supplied by an independent Safety Class 2 compressor for each system, is maintained at a level 10 percent higher than the post-LOCA reactor pressure vessel (RPV). This assures that any leakage of the MSIVs is toward the RPV or clean air to the environs (i.e., all leakage is in a direction away from the pressurized area).

12←• •→10 10←•

1.2.2.4.19 Suppression Pool Makeup

The suppression pool makeup is provided by water from the condensate makeup and drawoff system. Makeup to the suppression pool is not required following a LOCA.

1.2.2.4.20 Main Control Room HVAC

The main control room HVAC system provides an environment in the main control room suitable for the operation of equipment necessary for the safe shutdown of the plant and functions in the event of a LOCA. The system protects the plant operators from the results of any accident which could impair their safety and therefore compromise the safety of the plant.

1.2.2.5 Power Conversion System

1.2.2.5.1 Turbine Generator

The turbine is an 1,800 rpm tandem-compound, four-flow, single-stage reheat unit with an electro-hydraulic governor control. The turbine-generator is provided with an emergency trip system for turbine overspeed. The output of the turbine-generator is 990,565 kWe at turbine guarantee conditions with 3.0 in Hg abs back pressure and 0 percent makeup.

The generator is a direct driven, three-phase, 60 Hz, 22,000-V, 1,800 rpm hydrogen inner-cooled, synchronous generator rated at 1,151,100-kVA at 0.90 power factor, 0.58 short-circuit ratio at maximum hydrogen pressure of 75 psig.

1.2.2.5.2 Main Steam System

•→12

The main steam system delivers steam from the nuclear boiler system via four 24-in OD steam lines to the turbine generator, turbine bypass valves, steam jet air ejectors, off gas preheaters, and steam seal evaporator.

12←•

1.2.2.5.3 Main Condenser

The main condenser maintains 3.0 in Hg abs when operating at turbine guarantee conditions with 82.5°F circulating water inlet temperature. The condenser includes provisions for accepting steam bypassed around the turbine-generator. Deaeration of condensate is accomplished in the condenser.

1.2.2.5.4 Main Condenser Air Removal System

The main condenser air removal system using air ejectors for normal operation and vacuum hogging pumps for startup evacuates gases from the main turbine and condenser during plant startup and maintains the condenser essentially free of gases during operation. This system handles all inleakage of noncondensible gases through the turbine seals, condensate, feedwater, and steam systems, and noncondensibles which are generated in the reactor by disassociation of water.

•→12 •→8

1.2.2.5.5 Turbine Gland Sealing System

The turbine gland sealing system provides clean, nonradioactive steam to the seals of the turbine throttle valve stem glands and the turbine shaft glands. The seal steam condenser collects and condenses the air and steam mixture and discharges the air leakage to the turbine building vent, using a motor-driven exhauster. Contaminated gland seal heating steam is condensed in feedwater heater No. 4.

8←• 12←•

1.2.2.5.6 Steam Bypass System and Pressure Control System

A turbine bypass system is provided which passes steam directly to the main condenser under the control of the pressure regulator. Steam is bypassed to the condenser whenever the reactor steaming rate exceeds the load permitted to pass to the turbine generator. The capacity of the turbine bypass system is 10 percent of the reactor rated steam flow. The pressure regulation system provides main turbine control valve and bypass valve flow demands so as to maintain a nearly constant reactor pressure during normal plant operation. It also provides demands to the recirculation system to adjust power level by changing reactor recirculation flow rate.

1.2.2.5.7 Circulating Water System

The circulating water system provides the condenser with a continuous supply of cooling water. The circulating water system is a pumped closed-loop system utilizing air-cooled mechanical draft cooling towers as a heat sink. Four one-quarter capacity circulating water pumps are provided to pump cooling water from the cooling tower basin through the main condenser and back to the top of the cooling towers. Makeup water is provided from the Mississippi River by two 100-percent capacity (one pump for the unit with one as an installed spare) makeup pumps and two upflow intake strainers.

1.2.2.5.8 Condensate and Feedwater Systems

The condensate and feedwater systems supply condensate from the condenser hotwell to the reactor pressure vessel. The condensate is pumped by three condensate pumps through the intercooler of the air ejector, the gland seal condenser, the full flow condensate prefiltration subsystem and the full flow condensate demineralizer system. After leaving the condensate demineralizers, the condensate flows through two drain coolers and five stages of low-pressure heaters. The drain coolers and low-pressure heaters are split into two one-half capacity parallel streams. The last low pressure heaters discharge to the suction of three parallel motor-driven reactor feedwater pumps. The discharge of the reactor feedwater pumps passes through two one-half capacity parallel heaters and into the reactor pressure vessel. The feedwater flow is controlled by varying the feedwater flow control valve position.

1.2.2.5.9 Condensate Prefiltration and Demineralizer System

A full flow condensate prefiltration subsystem complete with bypass capabilities, backwash facilities, instrumentation, and semiautomatic controls, is designed to remove solid particulate before the demineralizers. This will improve water quality and increase demineralizer bed life.

A full flow condensate demineralizer system complete with regeneration facilities, instrumentation, and semiautomatic controls is designed to ensure a constant supply of high quality water to the reactor.

1.2.2.6 Electrical Systems and Instrumentation Control

1.2.2.6.1 Electrical Power Systems

The electrical power systems include the equipment and subsystems necessary to generate electrical power and deliver it to a 230-kV switchyard with a portion of it made available for station service. They further provide power for the control and operation of electrically driven equipment, instrumentation, and for power required during accident conditions.

•→7

The major equipment in the electrical system is one single shaft, turbine driven, main generator designated as Unit 1. The main generator is connected to two main stepup transformers and three normal station service transformers without intervening switchgear. The two 21.45-kV - 230-kV main stepup transformers have both their low voltage and their high voltage windings connected in parallel. The one 22-KV – 4.16-KV (STX-XNS1C) and the two 22-KV – 13.8-KV normal station service transformers (STX-XNS1A and STX-XNS1B) have their primary windings connected in parallel but their secondary windings are connected to separate buses, when the transformers are being used to supply auxiliary plant loads.

7←•

The output of the main stepup transformers is conducted to the 230-kV switchyard by a single circuit via a two-circuit 230-kV tower line. At the switchyard end, the 230-kV input from the generator plant has access to either or both of the two 230-kV buses of the switchyard in a breaker and a half scheme. Fig. 8.1-5 shows the 230-kV switchyard arrangement.

The 230-kV switchyard is located about 4,000 ft southwest of the power plant. The two-circuit tower lines have sufficient separation between them so that the failure of one does not affect the other.

•→7

Preferred plant ac station service power is taken from two physically and electrically separate 230-kV lines originating in the onsite 230-kV switchyard. Their function is to provide all power requirements of the plant when normal power is not being utilized. This includes startup, hot standby, shutdown power, normal power operation and safety-related loads. Each 230-kV line energizes a pair of preferred station service transformers. One pair of transformers is located near the plant in transformer yard 1A, and its 230-kV line is routed from the 230-kV switchyard on the same tower with the generator output. A similar arrangement (installed and operable prior to fuel loading) serves the same function for the pair of transformers serving the plant and located in transformer yard 2A.

7←•

The secondary of one of the preferred transformers in transformer yard 1A is routed to one of the 13.8-kV buses of the plant. Similarly, one of the preferred source transformers in transformer yard 2A is routed to the second 13.8-kV bus of the plant. Similar arrangement and locations are used for the second member of each pair of transformers routed to the two 4.16-kV buses in the plant. The preferred transformers are energized at all times.

Each section of the 13.8-kV bus has operator-controlled access to the assigned normal and assigned preferred source transformers.

•→7

In addition, if operating off of normal auxiliary power, automated throwover from normal to preferred source is provided, as described in Section 8.3.1.1.3, upon loss of normal power. Each of the two 13.8-kV buses supports approximately half of the unit auxiliaries.

7←•

•→7
Each of the two primary 4.16-kV in-station normal buses can be energized via the dual secondaries of a normal station service transformer, or they can be energized from their respective preferred station transformer. A third 4.16-kV in-station normal swing bus is subordinate to one or the other 4.16-kV normal buses. Two of the standby 4.16-kV buses, A and B, are connected to their respective preferred station service transformers. The standby 4.16-kV bus, C, is normally connected to the 4.16-kV in-station normal swing bus. Each of these standby buses has a standby diesel generator capable of supporting it upon loss of normal and preferred power. Switching allows each of the 4.16-kV standby buses to have access to one of the two 4.16-kV in-station buses, while the 4.16-kV standby bus C is subordinate to the 4.16-kV in-station swing bus. The 4.16-kV standby buses serve redundant loads and are not electrically mutually supporting.
7←•

Three onsite diesel generators, electrically and physically independent of each other, are provided to supply electrical ac power upon loss of normal and preferred ac power. These diesel generators supply ac power to safety-related equipment required for a safe shutdown.

Dc power for controls, instrumentation, and dc loads is provided by five 125-V batteries with their associated chargers and distribution panels. Three 125-V battery systems are associated with the three diesel generators, one for each, and the other two 125-V battery systems are associated with the normal switchgear.

Chapter 8 gives more information on electrical power systems.

1.2.2.6.2 Nuclear System Process Control and Instrumentation

1.2.2.6.2.1 Rod Control and Information System

•→12
The rod control and information system provides the means by which control rods are positioned from the main control room for power control. The system operates valves in each hydraulic control unit to change control rod position. One control rod can be manipulated at a time. The system includes the logic that restricts control rod movement (rod block) under certain conditions as a backup to procedural controls.
12←•

1.2.2.6.2.2 Recirculation Flow Control System

During normal power operation, a variable position discharge valve is used to control flow. Adjusting this valve changes the coolant flow rate through the core and thereby changes the core power level. The system can automatically adjust the reactor power output to the load demand. For startup and shutdown flow changes at lower power, the pump speed is changed by adjusting the frequency of the electrical power supply.

1.2.2.6.2.3 Neutron Monitoring System

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

1.2.2.6.2.4 Refueling Interlocks

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

1.2.2.6.2.5 Reactor Vessel Instrumentation

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

1.2.2.6.2.6 Process Computer System

An on-line process computer is provided to monitor and log process variables and to make certain analytical computations.

1.2.2.6.3 Power Conversion Systems Process Control and Instrumentation

1.2.2.6.3.1 Pressure Regulator and Turbine-Generator Control

The pressure regulator maintains control of the turbine control and turbine bypass valves to allow proper generator and reactor response to system load demand changes while maintaining the nuclear system pressure essentially constant.

The turbine-generator speed-load controls act to maintain the turbine speed (generator frequency) constant and respond to load changes by adjusting the reactor recirculation flow control system and pressure regulator setpoint.

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

1.2.2.6.3.2 Feedwater Control System

The feedwater control system automatically controls the flow of feedwater into the reactor pressure vessel to maintain the water within the vessel at predetermined levels. A conventional three-element control system is used to accomplish this function.

1.2.2.7 Fuel Handling and Storage Systems

1.2.2.7.1 New and Spent Fuel Storage

New and spent fuel storage racks are designed to prevent inadvertent criticality and load buckling under dry and flooded conditions. Sufficient coolant and shielding are maintained to prevent overheating and excessive personnel exposure, respectively. The design of the fuel pool provides for corrosion resistance, adherence to Seismic Category I requirements, and prevention of k_{eff} from reaching 0.95 under dry conditions or 0.95 under flooded conditions. This subject is further discussed in Section 9.1.

HOLTEC HI-STORM dry fuel storage systems are designed for the storage of spent fuel outside the spent fuel pool on the Independent Spent Fuel Storage Installation (ISFSI) pad located within the protected area of the plant. This subject is further discussed in Section 9.1.

1.2.2.7.2 Fuel Handling System

The fuel handling equipment includes a 125-ton cask crane, new fuel bridge crane, fuel handling platform, fuel inspection stand, fuel preparation machine, fuel assembly transfer mechanism, containment refueling platform, containment polar crane, and other related tools for reactor servicing.

The principal function of the cask crane is to handle spent fuel casks. The new fuel bridge crane transfers new fuel from the railroad bay to the new fuel storage vault and from the vault to the spent fuel pool. The fuel handling platform transfers the fuel assemblies between the transfer pool, storage pools, and cask. Fuel assemblies are transferred through the transfer tube between the reactor building and the fuel building. The fuel assemblies inside the containment are handled by the refueling platform.

The disassembly and reassembly of the reactor head, removable internals, and drywell head during refueling is accomplished using the containment polar crane.

All tools and servicing equipment necessary to meet the reactor general servicing requirements are designed for efficiency and safe serviceability.

1.2.2.8 Cooling Water and Auxiliary Systems

1.2.2.8.1 Standby Service Water System

The standby service water system removes heat from the various components required to operate during unit upset, emergency, and faulted conditions. The system has four 50 percent capacity pumps, two 100-percent capacity redundant headers, and the necessary associated piping, isolation valves, and instrumentation. The system consists of two independent trains, each capable of cooling the engineered safety features following a LOCA and rejecting this heat to the atmosphere through the standby service water cooling tower. The system is designed to meet Seismic Category I requirements.

1.2.2.8.2 Reactor Plant Component Cooling Water System

The reactor plant component cooling water system, a closed circuit heat transfer system, serves as an isolated intermediate heat sink for cooling reactor plant equipment. Heat is removed from this system by the normal service water system.

1.2.2.8.3 Turbine Plant Component Cooling Water System

The turbine plant component cooling water system, a demineralized water, closed circuit heat transfer system, serves as an isolated intermediate heat sink for cooling turbine plant and radwaste equipment. Heat is removed from the system by the normal service water system.

1.2.2.8.4 Ultimate Heat Sink

The ultimate heat sink consists of a 200-percent cooling tower located atop a 100-percent capacity water storage facility. The tower is capable of dissipating residual heat from the reactor plant undergoing either an orderly shutdown or an accident. The storage basin contains sufficient storage capacity to accommodate evaporative and drift losses over a 30-day period.

1.2.2.8.5 Condensate Makeup and Drawoff System

The condensate makeup and drawoff system consists of a storage tank, piping, and instrumentation. It receives drawoff water from and supplies makeup water to the main condenser and the fuel pool, and provides makeup of reactor coolant inventory for the reactor core isolation cooling system and high-pressure core spray system. Water in the condensate storage tank is replenished from the makeup water treatment system.

1.2.2.8.6 Makeup Water Treatment System

A makeup water treatment system (consisting of two trains, each composed of one cation exchange unit, one vacuum deaerator, one anion exchange unit, and one mixed bed exchange unit) purifies raw well water. It supplies demineralized water for makeup to the power conversion system, the turbine, and reactor plant component cooling systems, plus other unit operating requirements for demineralized water, such as the suppression pool and fuel pools.

1.2.2.8.7 Potable and Sanitary Water System

•→8

Potable water is supplied for drinking and for all other plumbing fixtures from the Consolidated Water District No. 13 system. Potable water meets the standards as promulgated by the U.S. Public Health Service. Raw sanitary waste is processed by the Wastewater Treatment Plant (WWTP) located south-west of the Clarifiers. The WWTP is comprised of aerated lagoons, sedimentation ponds, rock filter basins, gravity sand filter and an ultraviolet disinfection unit. Treatment consist of two parallel systems, one for the sanitary discharge from lift station SLS1 which serves the radiologically active portion of the plant and the other system for all other sanitary discharges outside the Protected Area. The WWTP facility is designed in compliance with the U.S. Environmental Protection Agency (National Pollution Discharge Elimination System) Permit and the Sanitary Code, the State of Louisiana requirements. Radioactive wastes or wastes containing chemicals are routed to the radioactive liquid waste treatment system for separate treatment.

8←•

1.2.2.8.8 Chilled Water Systems

1.2.2.8.8.1 Ventilation Chilled Water System

The ventilation chilled water system provides area cooling by directing plant air through chilled water coils. The system consists of three 50-percent capacity mechanical refrigeration water chillers, chilled water circulation pumps, compression tank, piping, valves, chilled water coils, and accessories.

1.2.2.8.8.2 Control Building Chilled Water System

•→10

The control building chilled water system consists of four 100-percent capacity, mechanical refrigeration water chillers, four chilled water circulation pumps, and associated piping, valves, and instrumentation. The system is designed to provide chilled water to the cooling coils in the air supply ventilation systems for the control building during all modes of plant operation, including DBA conditions.

10←•

1.2.2.8.9 Compressed Air Systems

•→12 •→4 4←•

The plant air system consists of separate Instrument and Service Air headers each supplied with three electric-driven air compressors. Each air compressor is equipped with a trim cooler and a moisture separator. The header is equipped with two parallel (100% capacity) pre-filters, two parallel (100% capacity) air dryers, and two parallel (100% capacity) after-filters. A manual start Diesel Driven Air Compressor is available in case of a loss of offsite power or electric driven compressor maintenance and can be used to supply either or both headers.

The Instrument Air header supplies an air receiver tank which serves as a large buffer tank which compensates for the loading and unloading pressures at the compressor discharge. The Service Air header supplies air to the service air distribution header which supplies Service Air to nearly all areas of the plant.

The Breathing Air sub-system provides clean air from the Service Air header throughout the plant.

The plant air system has the capacity of manually cross-connecting any of the six air compressors to either compressed air supply header. Service Air will automatically cross-connect with Instrument Air on low pressure in the Instrument Air header and will completely isolate the Service Air distribution header is air pressure continues to drop.

12←•

1.2.2.8.10 Process Sampling System

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

The process sampling system consists of three subsystems - turbine plant, reactor plant, and radwaste building sampling subsystems - each composed of instrumentation, coolers, analyzers, pipes, and valves connected to the unit process streams at various locations.

This sampling system takes samples for continuously monitoring the operation of the unit process equipment. It includes onsite chemical/radiochemical laboratory facilities where most liquid and gaseous samples are analyzed for oxygen, hydrogen, copper, iron, silica concentration, conductivity, turbidity, pH measurement, and radionuclide analyses.

1.2.2.8.11 Plant Equipment and Floor Drainage Systems

The plant equipment and floor drainage system collects all of the power station equipment and floor drainage. At the lowest drain point, a sump or a drain receiver tank equipped with pumps and level instrumentation is provided. All potentially radioactive drainage systems are isolated from any system which discharges out of the building and into the storm sewer system. Dependent upon the point of origin, the system directs drainage to the radioactive liquid waste treatment system for processing, to the main condenser hotwell for reuse in the steam generation system, or to the storm sewer system.

1.2.2.8.12 Heating, Ventilation, and Air-Conditioning Systems

Numerous heating, ventilation, and air-conditioning (HVAC) systems are provided throughout the plant in order to satisfy temperature and humidity conditions as required for equipment performance.

1.2.2.8.13 Fire Protection System

The fire protection system consists of: adequate water storage, fire pumps, water distribution system for fire hydrants, hose stations, automatic sprinkler and spray systems, automatic or manually actuated carbon dioxide and halon fixed fire protection equipment, automatic fire detectors in selected areas, and portable fire extinguishing equipment for use by operating personnel at various locations throughout the power station.

1.2.2.8.14 Communications Systems

These systems consist of the South Central Bell Telephone Co. network, radio communication, page-party communication subsystem, portable intercommunication subsystem, and microwave subsystems for communication throughout the Gulf States System and to send and receive signals with Beaumont, Texas, for load dispatching. Dial telephones, loudspeaker stations, and handset stations with muting facilities are provided in selected office and work areas throughout the station for uninterrupted communication. A portable intercommunication system is also provided for instrument calibration and maintenance.

1.2.2.8.15 Lighting Systems

•→12 •→8

Fluorescent, incandescent, mercury vapor type lamps and light emitting diodes (LEDs) are used for lighting the station, roadways, walkways, and parking areas. Lighting transformers are supplied from motor control centers and are located near centers of load. No mercury vapor lamps are used inside the containment, turbine building (except high bay turbine hall), auxiliary building condensate demineralizer area, radwaste building, or inside the fuel building. Mercury vapor lamps are provided for general yard, roadway, high bay turbine hall, and security fence lighting. Dc lighting is supplied from the station battery systems. This dc standby lighting is energized automatically on loss of normal ac power and deenergized automatically when normal ac power is restored.

•←8 12←•

1.2.2.8.16 Diesel Generator Fuel Oil Storage and Transfer System

The diesel generator fuel oil storage and transfer system supplies fuel oil for the operation of the standby diesel generator sets during loss of station power or during a LOCA occurring simultaneously with a loss of offsite power.

1.2.2.8.17 Auxiliary Steam System

An auxiliary steam system is provided to furnish a separate and independent steam supply. Process steam is generated in packaged, high voltage, electrode boilers and distributed through the plant by an auxiliary steam header. Auxiliary steam is required for condensate deaeration/heating, pump testing, and main turbine shaft seal steam during startup.

1.2.2.8.18 Normal Service Water System

•→6

The normal service water system removes heat from the reactor plant component cooling water system, turbine plant component cooling water system, and various equipment located within the unit. Heat is removed from the system by the service water cooling system heat exchangers.

Heat is removed from the service water cooling system by evaporative cooling in the service water cooling system mechanical draft cooling tower. Heat is also removed during mass transfer in the pumpwell through mixing of the service water cooling system water with lower temperature water from the normal makeup to the service water cooling system.

6←•

1.2.2.8.19 Containment Ventilation

The containment ventilation system provides air recirculation and cooling in the containment volume. The system is designed to maintain a bulk air temperature of 90°F during normal operation as a suitable environment for personnel and equipment. The system is an engineered safety system and is designed to be in operation during accident conditions. There are 3 50 percent capacity unit coolers in this system of which 2 are designed to safety grade requirements. Only one of the safety grade unit coolers is required to assist the RHR system during accident conditions.

1.2.2.8.20 Fuel Pool Cooling and Cleanup System

The fuel pool cooling and cleanup system maintains acceptable levels of temperature and clarity, and minimizes radioactivity levels of the water in the upper containment, fuel storage, and cask pools. The system includes two heat exchangers, each capable of removing one-half of the decay heat generated from an average discharge of spent fuel, and two filter/demineralizers, each unit having the capacity to pass the system flow or greater in order to maintain the desired purity level.

One heat exchanger is sufficient to prevent water from boiling in the pools under emergency conditions.

•→10

1.2.2.8.21 Suppression Pool Cleanup, Cooling and Alternate Decay Heat Removal System

The suppression pool cleanup, cooling, and alternate decay heat removal system provides a method to clean and cool the suppression pool during normal plant operation. The system also provides an alternate method of decay heat removal during plant shutdowns. The system includes two 100% capacity pumps, a plate and frame heat exchanger, backwashable filter, demineralizer vessel, and backwash tank. Cooling water to the heat exchanger is provided by service water. Safety related, air operated valves are provided at the interface between residual heat removal system piping and suppression pool cleanup system piping.

10←•

1.2.2.9 Radioactive Waste Management

1.2.2.9.1 Gaseous Radwaste System

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

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

•→12

Continuous radiation monitors are provided which indicate radioactive release from the reactor and from the charcoal adsorbers. The radiation monitors are used to isolate the off gas system on high radioactivity in order to prevent releasing gases of unacceptably high activity.

12←•

1.2.2.9.2 Liquid Radwaste System

The liquid radioactive waste system collects, treats, and stores liquid radioactive waste on a batch basis. Protection against accidental discharge is provided by the design and supplemented by procedural controls. Liquid waste is discharged on a batch basis at a controlled rate after sampling and laboratory analysis. Instrumentation with alarms to detect and record radioactivity concentration in the liquid radioactive waste discharges is provided.

1.2.2.9.3 Solid Radwaste System

A contractor provided solid radioactive waste system collects, treats, and stores solid radioactive wastes for offsite shipment. Solid waste is handled on a batch basis. Radiation levels of the various batches as packaged are determined prior to offsite shipment to ensure conformity with Department of Transportation requirements.

1.2.2.10 Radiation Monitoring and Control

1.2.2.10.1 Radiation Monitoring System

Radiation monitoring systems are provided to monitor and control radioactivity in process and effluent streams and to activate appropriate alarms and controls.

•→9

A radiation monitoring system is provided for indicating and recording radiation levels associated with selected plant process streams and effluent paths leading to the environment. All potentially radioactive gaseous and selected effluent discharge paths are monitored.

9←•

Radiation monitoring is also discussed in Sections 7, 9, and 11.

1.2.2.10.2 Area Radiation Monitoring System

A number of radiation monitors are provided to monitor for abnormal radiation at various locations in the reactor building, control room, auxiliary building, fuel building, condensate demineralizer area, turbine building, and the radwaste building. These monitors annunciate alarms when abnormal radiation levels are detected to alert occupants and the main control room personnel of excessive gamma radiation levels at selected locations within the plant.

1.2.2.10.3 Site Environs Radiation Monitors

Onsite radiation monitors surrounding the proposed location of River Bend Station monitor the environmental radiation level at this site. Airborne particulate matter, gases, and precipitation are sampled.

1.2.2.11 Shielding

Shielding, based on occupancy requirements in the various areas of the unit, is provided to reduce personnel exposure levels not to exceed the limits delineated in 10CFR20 and other appropriate regulations.

1.2.2.12 Particularly Difficult Engineering Problems

In general, particularly difficult engineering problems can be defined as those requiring development work or vendor testing to finalize the design. Such areas are discussed in Section 1.5.

1.3 COMPARISON TABLES

These tables reflect information that was current at the time of FSAR submittal, April 1981.

1.3.1 Comparison with Similar Facility Designs

This section highlights the principal design features of River Bend Station and compares its major features with other boiling water reactor facilities. The Grand Gulf Nuclear Station, the Clinton Power Station, and the Perry Nuclear Power Plant are used for comparison because they utilize the BWR/6 type design and their operating license reviews were initiated by the NRC prior to the River Bend Station operating license submittal. Zimmer is used for comparison because it is the most advanced station of the BWR/5 product line reviewed by the NRC. The design of the River Bend Station is based on proven technology attained during the development, design, construction, and operation of boiling water reactors of similar types. The data, performance characteristics, and other information presented here represent the current design.

Tables 1.3-1 through 1.3-7 compare the River Bend Station with Grand Gulf, Clinton, Perry, and Zimmer, listing design characteristics for the following:

1. Nuclear steam supply system

2. Power conversion systems

3. Engineered safety features

4. Containment

5. Radioactive waste management systems

6. Structural requirements

7. Instrumentation and electrical systems.

1.3.2 Comparison of Final and Preliminary Information

Table 1.3-8 provides a list of significant changes between the final and preliminary designs of River Bend Station. These changes, which occurred since the submission of the RBS-PSAR, were controlled and approved in accordance with administrative procedures and were within the scope of the principal design criteria. In addition to these changes, recently designed equipment was included in the FSAR, whereas only conceptualization of functional descriptions was available for the PSAR.

1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

1.4.1 Applicant

•→7 •→12
The applicant for the facility operating license for the River Bend Station is Gulf States Utilities, an investor-owned company incorporated in 1925 under the laws of the State of Texas. Under the original operating licenses, River Bend Station was owned jointly by Gulf States Utilities Company (70 percent) and Cajun Electric Power Company (30 percent), with Gulf States Utilities acting as the licensing agent and project manager on behalf of the owners and assuming responsibility for the design, construction, and operation of the facility. In December 1997 Cajun Electric Power Cooperative's 30% ownership interest in River Bend Station was transferred to Entergy Gulf States, Inc. by Amendment 101 to the Facility Operating License.
12←• •→10
Gulf States Utilities maintained an engineering and construction staff in support of its generation and transmission facilities. In preparation for the construction and operation of a nuclear generating station, a program of education and training in nuclear power was utilized by both management and engineering personnel. In addition, individuals with backgrounds in the nuclear power industry have joined the staff. With these additions and the training of personnel, Gulf States Utilities (GSU) is qualified to own and operate River Bend Station (RBS). Gulf States Utilities Company (GSU) was renamed Entergy Gulf States, Inc. by Amendment No. 88 to the Facility Operating License (NPF-47) issued July 30, 1996. By way of NRC Order issued on October 26, 2007, the Facility Operating License was transferred to a new Louisiana limited liability company, Entergy Gulf States Louisiana, LLC. By way of Amendment No. 189 issued on October 1, 2015, the Facility Operating License was transferred from Entergy Gulf States Louisiana, LLC, to Entergy Louisiana, LLC. Subsequent references to GSU, contained herein, are retained for historical purposes.
10←•
Entergy Operations, Inc. (EOI or Entergy Operations) now operates RBS. Both GSU and EOI are wholly owned subsidiaries of the Entergy Corporation, a registered public utility holding company. While GSU originally assumed responsibility for design, construction, and operation of the facility, on December 31, 1993, Entergy Operations, Inc. assumed from GSU the responsibility for the control and performance of licensed activities at RBS. As part of the final transfer, Entergy Operations assumed responsibility for commitments originally made by GSU. In those cases in the SAR where Entergy Operations has either present or future responsibility, reference is made to "River Bend Station" (RBS or Company) if the responsibility is site-specific, and "Entergy Operations" if the responsibility is more appropriately placed corporate-wide with no mention of GSU. However, to address certain historical information where a reference to Entergy Operations could cause confusion, "GSU" is used to represent situations where GSU originally had responsibility or made commitments but where Entergy Operations is now responsible.

Entergy Operations, Inc. is qualified to operate River Bend Station based on its quantity of qualify management and engineering support staff that are trained, knowledgeable and experienced in the nuclear power industry.
7←•

•→7
1.4.2 Architect Engineer

Stone & Webster Engineering Corporation (SWEC) is a Massachusetts corporation with offices in Boston, Massachusetts; Cherry Hill, New Jersey (CHOC); Denver, Colorado (DOC); New York, New York (NYOC); and Houston, Texas (HOC). The corporation had approximately 10,500 personnel as a manpower resource pool to support its activities, with about 700 engineers, designers, construction specialists, and clerical and administrative personnel assigned to the River Bend Station project at its peak. In addition to its project-dedicated staff, SWEC had
7←•

THIS PAGE LEFT INTENTIONALLY BLANK

utilized its own staff of specialists in various engineering disciplines to ensure that River Bend Station was designed in accordance with industry codes and standards and meets the requirements of the applicable federal, state, and local regulations for commercial nuclear power plants.

•→7

Preceding, and in addition to, commercial nuclear power projects, SWEC was engaged in engineering, design, and construction of chemical refineries, hydroelectric stations, and fossil fuel power plants. It had participated in the design and construction of fossil fuel plants with a total capacity in excess of 41,000,000 kW. SWEC has been actively engaged in nuclear engineering and construction of nuclear power plants since 1954, with an accumulated experience in excess of 20,000,000 kW reactor thermal power. It had participated in the design and/or construction of the following nuclear power stations, all of which are operating or have operated successfully:

7←•

1. Shipping port Atomic Power Plant of Duquesne Light Company and ERDA
2. Army Package Power Reactor (APPR, also known as SM-1)
3. Yankee Nuclear Power Station of Yankee Atomic Electric Company
4. Carolinas-Virginia Tube Reactor of the Carolinas-Virginia Nuclear Power Associates, Inc.
5. Haddam Neck Plant of Connecticut Yankee Atomic Power Company
6. Nine Mile Point Nuclear Station Unit 1 of Niagara Mohawk Power Corporation
7. Maine Yankee Atomic Power Station of Maine Yankee Atomic Power Company
8. Surry Power Station Units 1 and 2 of Virginia Electric and Power Company
9. James A. FitzPatrick Nuclear Power Plant - Unit 1 of the Power Authority of the State of New York
10. North Anna Power Station - Units 1 and 2 of Virginia Electric and Power Company

11. Beaver Valley Power Station - Unit 1 of Duquesne Light Company.

•→7
In addition to the River Bend Station, SWEC had under design or construction at that time the following nuclear power stations:
7←•

1. North Anna Power Station - Unit 3 of Virginia Electric and Power Company

2. Millstone Nuclear Power Station - Unit 3 of Northeast Utilities Service Company

3. Shoreham Nuclear Power Station - Unit 1 of Long Island Lighting Company

4. Beaver Valley Power Station - Unit 2 of Duquesne Light Company

5. Nine Mile Point - Unit 2 of Niagara Mohawk Power Corporation.

•→7
In addition, SWEC was providing construction management services for the Demonstration Liquid Metal Fast Breeder Reactor Plant (Clinch River Project) and for the Gas Centrifuge Uranium Enrichment Plant by the Department of Energy.

Thus, SWEC was technically qualified to provide the engineering, design, and construction for River Bend Station.

1.4.3 Nuclear Steam Supply System Supplier

The General Electric Company (GE) designed, fabricated, and delivered the direct cycle boiling water nuclear steam supply system, fabricated the first core of nuclear fuel, and provided technical direction for the installation and startup of this equipment. GE has been engaged in the development, design, construction, and operation of boiling water reactors since 1955.

Table 1.4-1 lists over 80 GE reactors completed or under construction. Thus, GE has substantial experience, knowledge, and capability to design, manufacture, and furnish technical assistance for the installation and startup of River Bend Station.
7←•

1.4.4 Turbine-Generator Supplier

GE has supplied the turbine-generators and the technical assistance for installation and startup of this equipment. GE has a long history in the application of turbine-generators in nuclear power stations that goes back to the inception of nuclear facilities for the production of electrical power. GE has firm orders to supply many turbine-generator units for use in nuclear facilities similar to the River Bend Station facility and also many non-nuclear turbine units. The inlet pressures of the nuclear units vary between 750 psig and 1,500 psig, and the inlet temperatures vary from saturation to approximately 40°F super heat. The ratings of these units range from 500,000 kWe to 1,100,000 kWe. Thus, GE is technically qualified to design, fabricate, and deliver the turbine-generator and to provide technical assistance for the installation and startup of the turbine-generator.

1.4.5 Principal Consultants and Outside Service Organizations

•→7

Several consulting groups were employed by Gulf States Utilities for special services relating to the River Bend Station. Engineering related to inservice inspection of the nuclear steam supply system and balance of plant was being provided by Southwest Research Institute. Quality Assurance audits for the nuclear fuel were originally conducted by NUS Corporation. Louisiana State University was working for Gulf States Utilities in the environmental areas of terrestial and acquatic biology. In addition, General Electric Company provided startup services. NUSAC provided expertise and consulting services in the area of operational plant security. Consulting contract engineering firms continue to be used but only on an as-needed basis.

7←•

1.5 REQUIREMENTS FOR FURTHER TECHNICAL INFORMATION

1.5.1 Development of BWR Technology

1.5.1.1 BWR Improvements

Development programs undertaken by General Electric Company (GE) to improve the safety and performance of the BWR product lines are now completed. Detailed discussion of each of these programs is presented below.

1.5.1.2 Current Development Programs

1.5.1.2.1 Instrumentation for Vibration Testing

Vibration testing for reactor internals is performed on all GE-BWR plants. At the time of issue of NRC Reg. Guide 1.20, test programs for compliance were instituted. The first BWR 6 plant of each size is considered a prototype design and is instrumented and subjected to both cold and hot, two-phase flow testing to demonstrate that flow-induced vibrations similar to those expected during operation do not cause damage. Subsequent plants which have internals similar to those of the prototypes are tested in compliance to the requirements of Reg. Guide 1.20 to confirm the adequacy of the design with respect to vibration. Since Kuosheng 1 is the prototype of the Standard 218-size plant, only confirmatory testing was completed at RBS. Further discussion is presented in Section 3.9.2.4B.

1.5.1.2.2 Core Spray Distribution

GE has conducted a program to predict BWR 6 core spray distributions using a combination of single nozzle steam and air tests, single and multiple nozzle analytical models, and full scale air tests. This methodology has been confirmed by a full scale 30-deg sector steam test as discussed in NEDO-24712, "Core Spray Design Methodology Confirmation Test," August 1979. In the letter from Mr. R. L. Tedesco (NRC) to Dr. G. G. Sherwood (GE) dated January 30, 1981, the NRC concluded the tests documented in NEDO-24712 "constitute an adequate confirmation of the GE spray distribution methodology for BWR/6 type spargers."

1.5.1.2.3 Core Spray and Core Flooding Heat Transfer Effectiveness

Due to the incorporation of an 8x8 fuel rod array with unheated "water rods," tests have been conducted to demonstrate the effectiveness of ECCS in the new geometry. These tests are regarded as confirmatory only, since the geometry change is very slight and the "water rods" provide an additional heat sink in the inside of the bundle which improves heat transfer effectiveness.

There are two distinct programs involving the core spray. Testing of the core spray distribution has been accomplished, and the Licensing Topical Report, NEDO-10846, BWR Core Spray Distribution, April 1973, has been submitted. The other program concerns the testing of core spray and core flooding heat transfer effectiveness. The results of testing with stainless steel cladding were reported in the Licensing Topical Report, NEDO-10801, Modeling the BWR/6 Loss-of-Coolant Accident: Core Spray and Bottom Flooding Heat Transfer Effectiveness, March 1973. The results of testing using Zircaloy cladding were reported in the Licensing Topical Report, NEDO-20231, Emergency Core Cooling Tests of an Internally Pressurized Zircaloy Clad, 8x8 Simulated BWR Fuel Bundle, December 1973.

1.5.1.2.4 Verification of Pressure Suppression Design

The General Electric Company has conducted a large scale test program to verify the performance characteristics of the Mark III containment.

The purpose of the Mark III Test Program was to confirm the analytical methods used to predict the drywell and containment pressure response following the postulated LOCA. This test program was also used to obtain information on the hydrodynamic loads that are generated in the vicinity of the suppression pool during a LOCA.

The General Electric Mark III containment pressure suppression testing program was initiated in 1971, with a series of small-scale tests. The test apparatus consisted of small-scale simulations of the reactor pressure vessel, drywell, suppression pool, and horizontal vents. A total of 67 blowdown runs were made. The purpose of these tests was to determine the behavior of the horizontal vents and to obtain data for determining the acceleration of the water in the test section vents during initial clearing. This information was used to establish an analytical model for predicting vent system performance in Mark III and the resulting drywell pressure response.

In November 1973, testing in the Mark III Pressure Suppression Test Facility (PSTF) began. The PSTF consists of an electrically heated steam generator connected to a simulated drywell which can be heated to prevent steam condensation within its volume during the simulated blowdowns. The drywell is modeled as a cylindrical vessel having a 10-ft diameter and 26-ft height. A 6-ft diameter vent duct passes from the drywell into the suppression pool and connects to the simulated vent system. Pool baffles are used to simulate a scaled or full-scale sector of a Mark III suppression pool. The pool arrangement is such that both vent submergence and pool areas can be varied parametrically.

The full-scale PSTF testing performed between November 1973 and February 1974 obtained data for the confirmation of the analytical model. In March 1974, pool swell tests were performed in the PSTF. These full-scale tests involved air blowdown into the drywell and suppression pool to identify bounding pool swell impact loads and breakthrough elevation, i.e., that elevation at which the water ligament begins to break up and impact loads are significantly reduced. Impact load data were obtained on selected targets located above the pool.

In June 1974, after the PSTF vent and pool system was converted to 1/3-scale, four series of tests were performed to provide transient data on the interaction of pool swell with flow restrictions above the suppression pool surface. Other areas where data was obtained included vent clearing, drywell pressurization, and jet forces on pool walls.

The next series of 1/3-scale testing began in January 1975, and was directed at obtaining local impact pressures and loads for typical small structures located over the pressure suppression pool, including I-beams, pipes, and grating. Data from this test series expanded the data base from the full-scale air tests. A further series of 1/3-scale tests was added in June 1975, to obtain comparable data on pool swell velocity and breakthrough elevation to the full-scale air tests.

A series of small-scale flow visualization tests were performed in October 1976, in order to qualitatively investigate the steam condensation phenomena for the Mark III vent configuration. The visual investigation of steam bubble formation and collapse under various bulk pool temperature and vent steam flux conditions provided information for the placing of instrumentation in the vicinity of the PSTF drywell vents for subsequent tests.

The final three phases of the Mark III confirmatory test program began in November 1976, with a series of 1/3-scale tests under various initial suppression pool temperatures and simulated steam and liquid break sizes to obtain data on the localized conditions associated with the steam condensation portion of the LOCA blowdown. In parallel with this data acquisition, other test data was obtained for use in evaluating the loading conditions on submerged structures located in the suppression pool and for evaluating potential vertical thermal stratification of the suppression pool water. The second of the three phases was begun in September 1977. These full-scale tests also provided data on localized steam condensation conditions and thermal stratification.

Phase three consists of a 1/9-scale test series in which a nine-vent array is utilized to evaluate multivent effects. In establishing the LOCA-related conditions within the suppression pool, all of the vent stations are conservatively assumed to be in phase, even though the random nature of the phenomena indicates that some phase separation is expected during the steam condensation process. This final test phase is primarily aimed at confirming that multiple vent loading conditions are not in excess of those identified from single-cell tests.

It should be noted that the emphasis in some testing just described was directed at the evaluation of the pool swell phenomena, while in others the steam condensation phenomena was evaluated. Each test run consisted of a simulation of the postulated blowdown transient. Various postulated break sizes up to two times the Design Basis Accident for the containment were tested. Data were recorded at selected locations around the test facility suppression pool throughout the blowdown, so that the hydrodynamic conditions associated with each phase of the blowdown is available for selecting appropriate design loading conditions. General Electric has utilized this data to develop thermal and hydrodynamic loading conditions in the GE Mark III reference plant pressure suppression containment system during the postulated LOCA. Information on thermal and hydrodynamic loading conditions during the anticipated safety relief valve (SRV) discharge and related dynamic events has also been documented. Separate test data has been utilized to establish the SRV air clearing load prediction model. Information on SRV discharge thermal performance is also provided. The GE reference plant report contains information and guidance to assist the containment designer in evaluating the design conditions for the various structures which form the containment system. Table 1.5-1 identifies all of the GE-conducted LOCA-related tests which form the basis for hydrodynamic loads used. Table 1.5-2 identifies the documents referenced in Table 1.5-1, plus other reports containing test data used for non LOCA-related hydrodynamic load definitions.

1.5.1.2.5 Boiling Transition Testing

Since the formulation of the 1966 Hench-Levy Design Limit Lines for use in BWR thermal design, General Electric has continued to perform extensive steady state and transient boiling transitive test programs. Prior to 1974, over 14,000 data points had been obtained in water and Freon from many test assemblies having various axial heat flux profiles and rod-to-rod power distributions, covering all prototypical aspects of reactor operating conditions. Among those, 2,100 data points were full-scale simulation of 7x7 and 8x8 BWR fuel assemblies performed in the ATLAS test facility. A new boiling transition correlation (GEXL) has been developed and applied to GE-BWR thermal design. Detailed information is provided in the approved Licensing Topical Report, NEDO-10958A, General Electric BWR Thermal Analysis Basis (GETAB): Data, Correlation and Design Application, January 1977.

Since the implementation of GEXL correlation on design in 1974, General Electric has continued to conduct full-scale 8x8 assembly boiling transition tests, accumulating over 1,600 data points after GETAB introduction, to extend the data base and to assure applicability to new 8x8 fuel designs such as the two water-rod design for BWR/2-5 and for BWR/6. It has been shown that the 8x8 GEXL correlation with the appropriate R-factors can predict boiling transition critical power data for the two water-rod assemblies with an accuracy typical of the GEXL correlation predictability for other 8x8 design as described in NEDO-10958-A.

1.6 MATERIAL INCORPORATED BY REFERENCE

Documents which are referenced in this USAR are listed at the end of the sections in which they have been referenced.

1.7 DRAWINGS AND OTHER DETAILED INFORMATION

1.7.1 Electrical, Instrumentation, and Control Drawings

Table 1.7-1 contains a list of safety-related electrical, instrumentation, and control drawings which were submitted as a drawing package at the time of the original FSAR submittal. Updates to these drawings will be provided as specifically requested by the NRC.

1.7.2 Piping and Instrumentation Diagrams

Table 1.7-2 contains a list of the system piping and instrumentation diagrams provided in the USAR.

1.8 CONFORMANCE TO NRC REGULATORY GUIDES

•→12

Regulatory Guides are issued to describe and make available to the public, methods acceptable to the NRC staff for implementing specific parts of the Commission's regulations, to delineate techniques used by them to evaluate specific problems, or postulated accidents, or to provide guidance to applicants. Regulatory Guides are not substitutes for regulations, and compliance with them is not required. Methods and solutions different from those set out in the guides will be acceptable if they provide a basis for the findings requisite to the issuance or continuance of a permit or license by the Commission. Table 1.8-1 indicates the extent of compliance with all applicable NRC Regulatory Guides and the revision number of those guides or the Quality Assurance Program Manual is referenced, which provides the information. Table 1.8-1 also indicates an assessment of those Regulatory Guides which are not applicable due to the implementation provisions of the guides. A reference to the USAR section in which the applicable design features are described is also provided.

Where the design differs from the Regulatory Guides, alternative methods of providing an equivalent level of safety have been utilized. These differences are either discussed in Table 1.8-1, in the Quality Assurance Program Manual, or reference is made to the appropriate USAR section(s) in which they are discussed.

New Regulatory Guides and subsequent revisions to existing Regulatory Guides issued through September 1980 were originally addressed. Later Regulatory Guides and revisions will be addressed as requested by the NRC, and added to the table if it is deemed appropriate.

12←•

APPENDIX 1A

RIVER BEND STATION POSITIONS ON THE NUCLEAR REGULATORY COMMISSION'S POST-TMI REQUIREMENTS, NUREG-0737

This appendix presents the River Bend Station positions on the BWR applicable items from the U.S. Nuclear Regulatory Commission's post-TMI action plan requirements for applicants for an operating license, NUREG-0737, Enclosure 2, revised draft letter from H. R. Denton to the commissioners dated October 22, 1980.

TABLE 1A-1

POSITIONS IN RESPONSE TO POST-TMI REQUIREMENTS

Item and Title	Position	USAR Reference
•→14 •→12 •→7 I.A.1.1 Shift Technical Advisor	Shift Technical Advisors may be used at RBS; however the shift superintendent or Control Room Supervisor, will normally be trained to meet the requirements indicated in the clarification letter of TMI Action Plan, 9/5/80 and NUREG-0737.	13.2.1.1 13.2.7.5
I.A.1.2 Shift Superintendent responsibilities 7←• 12←• 14←•	Policies have been established at RBS to insure that the shift superintendent is not overburdened with administrative duties that distract him from his principal responsibilities.	
I.A.1.3 Shift manning	Shift manning is in accordance with the requirements of the Technical Specifications. Use of overtime will be kept to a minimum in order to meet the intent of this requirement.	13.1.2.3
I.A.2.1 Immediate upgrade of RO and SRO training and qualifications	Training and qualifications for ROs and SROs at RBS meet the requirements stated in NUREG-0737.	13.2
I.A.2.3 Administration of training programs	All instructors permanently employed at RBS who teach systems, integrated responses, transient, and simulator courses are qualified as Senior Reactor Operators (SRO) as specified in NUREG-0737.	13.2.1
I.A.3.1 Revise scope and criteria for licensing exams	Licensing examinations for RBS staff personnel are developed, administered, and graded in such a manner as to comply with NUREG-0737.	13.2.2.1

●→14 ●→10

Item and Title	Position	USAR Reference
I.B.1.2 Independent Safety Engineering Function 10←● 14←●	The staff dedicated to RBS, the On-Site Safety Review Committee, the Safety Review Committee, and the Independent Safety Engineering Function provide engineering and operational expertise throughout the lifetime of the station. This structure is believed to provide adequately experienced and trained people charged with reviewing the safety of the plant. RBS will ensure that the ISE function maintains the equivalent level of independent review that would be accomplished by five dedicated, full-time engineers by distributing the Independent Safety Engineering Function throughout the Quality Programs and the Nuclear Safety and Regulatory Affairs organizations.	13.4.2
●→12 I.C.1 Short-term accident and procedure review 12←●	Small-break LOCA analysis and inadequate core cooling have been conducted by GE and the BWR Owners' Group. Small-break LOCA models and plant-specific small-break LOCA calculations have been provided as indicated in response to Items II.K.3.30 and II.K.3.31. No design changes were made to RBS as a result of these analyses. Symptom oriented Emergency Operating Procedures and Severe Accident Procedures have been developed based on BWR Owners' Group Emergency Procedure and Severe Accident Guidelines and training conducted to provide proper operator response in the event of a small-break LOCA and inadequate core cooling.	13.5.1.2.1.4
I.C.2 Shift and relief turnover procedures	RBS Administrative Procedures contain shift turnover instructions. These procedures include requirements for completion of checksheets, review of logs, and review of plant conditions and system status. Signatures of personnel involved are required on these documents to verify proper completion.	13.5
●→14 ●→12 ●→7 I.C.3 Shift Superintendent responsibilities 7←● 14←●	The shift superintendent responsibilities at RBS are described in the USAR. In addition, the plant administration procedures clearly define the responsibilities and authority of the shift superintendent.	13.1.2.3.1
I.C.4 Main control room access 12←●	RBS procedures address those personnel who are allowed access to the main control room under normal and emergency conditions. RBS procedures also address delineation of authority during plant emergencies.	13.5.1.1.3.1 13.1.2.2.5
I.C.5 Feedback of operating experience	RBS procedures and training programs have been developed. The procedures and training programs have been written in accordance with Section I.C.5 of NUREG-0737.	13.1.2

TABLE 1A-1 (Cont)

Item and Title	Position	USAR Reference
I.C.6 Verify correct performance of operating activities	RBS procedures have been developed that have system status monitoring as a verification system.	13.5.2
•→12 I.C.7 NSSS vendor review of procedures	1. RBS Startup Procedures: The RBS Startup Manual includes the GE Operations Manager in the review, approval, and revision cycle of Startup Test Procedures. 2. RBS Emergency Operating Procedures: NSSS vendor review of the Emergency Operating Procedures (EOPs) is not required since EOPs to be implemented are based on NRC-approved BWR Emergency Procedure Guidelines that were developed by GE and the BWR Owners' Group (see RBS SER, Section 13.5.2.3).	14.2.9 14.2.2.4.1 13.5.1.2.1.4
I.C.8 Pilot monitoring of selected emergency procedures for NTOLs	N/A to RBS	
I.D.1 Main control room design reviews	The RBS main control room is designed with consideration given to human engineering factors. Current industry activities have led to a BWR Owners' Group Control Room Survey Program. A preliminary panel layout review has been conducted using a BWR Owners' Group survey checklist.	13.1
I.D.2 Plant safety parameter display console	The RBS design includes a safety parameter display system to satisfy the requirement of this item.	14.2.12.3.28
I.G.1 Training during low-power testing 12←•	Training during low-power testing has been accomplished at RBS in accordance with NUREG-0737. Information on low-power testing is submitted to the NRC as required by the Technical Specifications.	13.2.1.2

TABLE 1A-1 (Cont)

Item and Title	Position	USAR Reference
•→7 II.B.1 Reactor coolant system vents 7←•	Entergy Operations supports the BWR Owners' Group position submitted on October 17, 1979. Specifically for RBS, primary venting capability is provided by the 16 power-operated safety relief valves. Each of the safety relief valves is seismically and Class 1E qualified, and the air supply to the seven valves which comprise the automatic depressurization system is seismically qualified. These valves can be manually operated from the main control room to vent the reactor coolant system. Emergency procedures provided to assure core cooling under accident conditions result in system venting and, hence, no specific venting procedures have been provided. Positive position indication for each valve is provided in the main control room. Additional venting capability is provided via a reactor vessel head vent valve and through operation of the turbine-driven reactor core isolation cooling system. No additional accident analyses have been provided as a result of a break in any of these vent lines because a more bounding complete steam line break is part of the RBS design basis.	5.2
II.B.2 Plant shielding	A radiation and shielding-design review has been performed on spaces around systems that may require access during and/or after an accident and may contain high radiation levels. In the evaluation of plant shielding and vital area access, post-accident radiation releases equivalent to the source terms described in NUREG-0737, Item II.B.2, are assumed. The radiation source terms used in the evaluation are described in Section 12.3.2.4. The results of the evaluation provide for adequate access to vital areas and protection of safety equipment by design changes, increased permanent shielding, and post-accident procedural or administrative controls. The personnel exposure in a vital area will be maintained in accordance with the guidelines in GDC 19 during the course of an accident. This position complies with NUREG-0737, TMI Action Plan Item II.B.2.	
II.B.3 Post-accident sampling	The RBS design incorporates a post-accident sampling system in accordance with NUREG-0737 and related clarification letters.	9.3.2

Item and Title	Position	USAR Reference
II.B.4 Training for mitigating core damage	Training for mitigating core damage has been incorporated into the operator training program. Currently, the training program addresses Enclosure 3 to H. R. Denton's March 28, 1980, letter as referenced in NUREG-0737. All other related personnel will receive training commensurate with their responsibilities.	13.2
•→7 II.D.1 Relief and safety valve requirements 7←•	River Bend personnel participated in a generic test program to satisfy the requirements of TMI Action Plan Item II.D.1. The Crosby SRV used at RBS was included in the test program. The testing requirements were determined by the BWR Owners' Group through systematic analysis of design accidents and operational transients. The conclusion from that analysis was "there is no design-basis accident or transient which requires safety, relief, or dual function SRVs to pass two phase or liquid flow at high pressure." A generic test program which addressed the alternate shutdown mode (two phase and liquid under low pressure conditions) of cooling has been completed. The final test report for the operability test program was submitted in a letter from T.J. Dente (BWR Owners' Group) to D.G. Eisenhut (NRC), dated September 25, 1981. This report, which includes final test data and analyses, demonstrates the operational adequacy of the SRVs and SRV discharge piping and supports. These final test results are contained in the General Electric Co. document NEDE-24988-P, "Analysis of Generic BWR Safety/Relief Valve Operability Test Results," which was included in the September 25, 1981, letter. A review of the test report shows the operational adequacy of the SRVs, discharge piping and supports has been demonstrated for the conditions defined in this TMI Action Plan item. This position is consistent with LRG-II Issue 6-RSB.	3.9.6B
II.D.3 Valve position indication	The RBS design includes acoustic sensors which provide a reliable indication of flow in each SRV discharge pipe. An individual indicating light for each SRV and a common annunciator are provided in the main control room to indicate when any one of the 16 SRVs is not fully closed. Individual indicating lights are also provided in the main control room for each SRV from the digital signals that actuate the pilot valves that open the associated SRV.	5.2.2.12

TABLE 1A-1 (Cont)

Item and Title	Position	USAR Reference
•→12 II.E.4.1 Dedicated hydrogen Penetrations 12←•	The RBS design includes two redundant hydrogen recombiners inside the containment. Additionally, the containment purge system is designed to satisfy the requirements of this item.	6.2.5
II.E.4.2 Containment isolation dependability	A containment isolation dependability study has been performed for RBS. Each paragraph of Item II.E.4.2 has been specifically addressed. The results of this study are provided in Section 6.2.4.3.7.	6.2.4
•→1 II.F.1 Accident-monitoring Instrumentation 1←•	A review of accident monitoring instrumentation in comparison with the guidance of Regulatory Guide 1.97, has been conducted. The results of this review are provided in revised Section 7.5. Effluent radiological monitoring and sampling is further discussed in Section 11.5.	7.5, 11.5
II.F.2 Instrumentation for detection of inadequate core cooling	RBS has participated in a BWR Owners' Group study analyzing inadequate core cooling (ICC) in boiling water reactors. In conjunction with the submittal of Sol Levy Report Nos. SLI-8211 and SLI-8218 to the NRC, RBS has reviewed and evaluated the reports against its plant design and has concluded the following: 1. The results of Sol Levy Report No. SLI-8218 affirm that reactor pressure vessel (RPV) water level is a reliable, responsive indicator of ICC. RPV water level instrumentation measures the trend toward ICC, indicates its existence, and indicates the return of adequate core cooling. 2. The RPV water level measurement enhancements, as modified by specific changes identified in SLI-8211 and discussed in the response to NRC Question 421.014, provide suitable means for detection of adequacy of core cooling.	7.5
•→7 II.K.1.5 Review ESF valves 7←•	River Bend personnel reviewed all safety-related valve positions, positioning requirements, and positive controls to assure that valves remain positioned (open or closed) in a manner that ensures the proper operation of engineered safety features. This review covered procedures for control of maintenance and testing of safety-related systems, system operating and general plant startup procedures, and shift turnover and general rounds procedures.	13.5

TABLE 1A-1 (Cont)

Item and Title	Position	USAR Reference
•→14 •→7	Manually operated valves in the main flow paths for safety-related systems will be locked in position and verified by valve lineup. Valve lineups for safety-related systems will be performed by two qualified operations personnel, one person doing the initial positioning and a second person verifying the position. The maintenance work request procedure requires the shift superintendent or Control Operations Foreman to ensure that appropriate functional tests are assigned and/or the system has been properly restored to its normal configuration after maintenance has been performed. Surveillance test procedures have data sheets requiring operator signoff to verify that each system is returned to its normal configuration after performing a test. Shift turnover procedures are as described in GSU response to Item I.C.2 and include review of system status.	
II.K.1.10 Operability status 7←• 14←•	a. The RBS maintenance work request procedure requires the shift superintendent or Control Operations Foreman to initially determine if a deficiency (maintenance problem) affects a safety-related system and if it has rendered a piece of equipment inoperable. The shift superintendent or Control Operations Foreman then initiates technical specification actions if necessary. In order for maintenance to begin, proper unit and/or system conditions must be established and formal permission must be granted by the shift superintendent or Control Operations Foreman, who determines if allowing the work to begin would render any system inoperable and initiates proper actions. b. The RBS maintenance work request procedure requires the shift superintendent or Control Operations Foreman to ensure that the requested maintenance has been performed, that an appropriate functional test has been assigned, and/or that the system has been properly restored to its normal configuration upon completion of maintenance work on a safety-related system. Surveillance test procedures have data sheets requiring operator signoff to verify that each system is returned to its normal configuration after performing a test. c. In order for maintenance to begin on a safety-related system, formal permission must be granted by the shift superintendent or Control Operations Foreman. The shift	13.5

Item and Title	Position	USAR Reference
•→14 •→7 7←• 14←•	superintendent or Control Operations Foreman coordinates the conduct of retests and restoration of systems after maintenance. In order for a surveillance test to be performed on a safety-related system, the shift superintendent must give permission and the Control Room Operator must be notified. Upon completion of the surveillance test, the shift superintendent and Control Room Operator must be informed.	
II.K.1.22 Auxiliary heat removal system •→12 12←• •→4 4←•	When the RBS feedwater system is inoperable, the reactor automatically scrams at water level 3. If the RCIC system is not manually initiated from the main control room, the RCIC and HPCS systems automatically initiate at level 2 after the automatic isolation of the main steam line. These systems supply makeup water to the reactor pressure vessel until water level 8 is reached. At level 8, the RCIC and HPCS systems are tripped. Both systems automatically restart once the high level trip signal clears and a level 2 signal is received. The main steam relief valves will automatically or manually blow down to the suppression pool, in the event the vessel is isolated. In this event, the RBS suppression pool cooling mode of the residual heat removal system is used to transfer heat to the ultimate heat sink (USAR Section 7.3). This requires remote manual alignment of the residual heat removal system valves and, if normal service water is lost, startup of two standby service water pumps. For accident situations with the reactor vessel at high pressure, the high pressure core spray system is used to automatically provide the required makeup flow. Manual operations are not required since the high pressure core spray system cycles on and off automatically as water level reaches level 2 and level 8, respectively. If the high pressure core spray system fails under these conditions,	7.4.1

TABLE 1A-1 (Cont)

Item and Title	Position	USAR Reference
•→12 12←•	the operator can manually depressurize the reactor vessel using the automatic depressurization system to permit the low pressure emergency core cooling systems to provide makeup coolant. Automatic depressurization occurs if all of the following signals are present: high drywell pressure, level 3 water level permissive, level 1 water level, pressure in at least one low pressure injection system, and the runout of a 105-sec timer which starts with the coincidence of the other four signals.	
II.K.1.23 Reactor vessel level procedures	The BWR water level instrumentation provides multiple level indications displayed on the reactor control console or nearby panels. These indications include three narrow range (normal operating range) level indicators and one narrow range level recorder, two wide range level recorders and one wide range level indicator, one fuel zone level recorder, one upset range level recorder, and one shutdown range (vessel flooding) level indicator. In addition, multiple indicating trip units provide wide range and narrow range reactor level safety-related trip signals and related alarms. GE has described this in greater detail in NEDO-24708.	7.5
II.K.3.3 Reporting safety valve and relief valve failures and challenges	Failures of reactor system relief valves are reported in the appropriate manner to the necessary NRC organizations.	16
II.K.3.11 Justification use of certain PORVs	N/A to RBS	
•→12 II.K.3.13 HPCI & RCIC Initiating levels 12←•	The evaluation performed by General Electric (GE) on behalf of the Owners' Group in a letter transmitted on October 1, 1980, from R. H. Buchholz (GE) to D. G. Eisenhut (NRC) concerning NUREG-0737, II.K.3.13, HPCI and RCIC Initiating Levels, is applicable to RBS. The report presented the analyses, conclusions, and recommendations regarding separation of the initiating levels of the High Pressure Coolant Injection (HPCI) and Reactor Core Isolation Cooling (RCIC)	7.4

Item and Title	Position	USAR Reference
	systems. RBS does not employ an HPCI system in its GE NSSS design, but instead has a High Pressure Core Spray (HPCS). The report identifies the differences in the HPCI/HPCS thermal fatigue analyses where appropriate. In summary, the study concluded that the HPCI and RCIC initiations at the current low water level setpoints are within the design basis thermal fatigue analysis of the reactor vessel and its internals. Separating HPCI and RCIC setpoints as a means of reducing thermal cycles is of negligible benefit. In addition, raising the RCIC setpoint or lowering the HPCI setpoint have undesirable consequences which outweigh the benefit of the limited reduction in thermal cycles. Therefore, when evaluated on this basis, no change in the RCIC or HPCI/HPCS setpoints is needed. In a letter dated December 29, 1980, from D. B. Waters, Chairman-BWR Owners' Group, to D. G. Eisenhut (NRC), the BWR Owners' Group transmitted the results of an evaluation performed by GE. The report recommends modifying the RCIC system to automatically restart following a trip of the system at high reactor vessel water level. This will be accomplished by relocating the existing high level trip from the RCIC turbine trip valve to the steam supply valve. Once the level reaches a predetermined high level the steam supply valve will close automatically. Entergy Operations endorses the modification to automatically restart the RCIC system on low water level and has incorporated the change in the RBS design. This position is consistent with LRG-II Issue 2-RBS(a).	
•→7		
II.K.3.15 Isolation of HPCI and RCIC	The NUREG-0737 position recommends a modification to the HPCI/RCIC steam supply pipe-break-detection circuitry to reduce inadvertent system isolation due to the pressure spike which accompanies startup of the systems. RBS has modified the existing isolation relay in each detection circuit with a Class 1E time delay relay having a setpoint range from 3 to 13 seconds. The existing circuitry is based on continuous high-steam flow closure (trip) of the isolation valves when the flow in that line exceeds approximately 300% of rated flow. The timer starts when the flow meters exceed the trip setpoint. System isolation only occurs if the flow meters still read at or above the trip setpoint at the end of the timer period. It has been determined that the addition of the 3- to 13-second time delay does not result in any change in the total reactor fluid mass release when considering design basis	7.4
7←•		

Item and Title	Position	USAR Reference
	conditions. Therefore, no effect is seen on the design basis analysis. This position is consistent with LRG-II Issue 2-RSB(b).	
II.K.3.16 Challenges to and failure of relief valves	The BWR Owners' Group position submitted to the NRC in May 1981 concludes that BWR/6 plants have design features which reduce the occurrence of stuck open relief valve (SORV) events such that no further modifications are required. RBS has already incorporated three design features which should reduce its SORV frequency by at least a factor of 17 compared to the standard BWR/4 plant design. The three features are: 1. Use of Crosby SRVs. 2. Use of a lower reactor pressure vessel water level isolation setpoint for main steam isolation valve closure. 3. Use of LOW-LOW SET SRV control logic.	5.2
II.K.3.17 ECCS outages •→16 16←•	A plan for data collection relating to outage dates and duration for all ECC systems has been developed. These data will be reviewed for availability information on these systems. All ECCS outages will be reported to the NRC via Licensing Event Reports (LER). The report will contain the following: 1. Outage dates and duration of outages. 2. Cause of outage. 3. ECC systems or components involved in the outage. 4. Corrective action taken. The LERs will provide the staff with the capability to accumulate, on a yearly basis, reliability data due to test and maintenance outages.	16
•→12 •→10 II.K.3.18 ADS actuation 10←• 12←•	GSU has modified its automatic depressurization trip system with a 5-minute bypass timer on the drywell pressure signal which provides an automatic backup to operator action to ensure adequate core cooling. This modification conforms to Option 4 of the BWR Owners' Group position submitted March 31, 1981, to the NRC.	7.3.1.1.1.2

Item and Title	Position	USAR Reference
	In addition, the BWROG symptom-oriented EPGs provide explicit instructions on when to manually depressurize the vessel if the high pressure systems cannot maintain inventory. Implementation of these improved procedures and operator training provides adequate assurance that the vessel is depressurized, if required.	
•→7 II.K.3.21 Restart of LPCS and LPCI •7←	River Bend personnel have concluded from their review of the RBS design that, in light of the BWR Owners' Group position submitted to the NRC on December 29, 1980 (April 20, 1982, letter from D. B. Waters, BWROG Chairman, to D. G. Eisenhut, NRC), no modifications should be made to the control logic of the existing LPCI and LPCS systems. It has been determined that modifications to the RBS HPCS system to automate restart on low level following manual trip are not required for safe operation. LRG-II Position 1-RSB, which supports this conclusion, has been accepted (February 26, 1982, letter from J. R. Miller, NRC, to D. L. Holtzscher, LRG-II Chairman).	7.3
II.K.3.22 RCIC suction	RBS design provides automatic switchover of the RCIC system suction from the condensate storage tank to the suppression pool when condensate storage tank level is low. Therefore, RBS design satisfies the intent of this item.	7.4
II.K.3.24 Space cooling for HPCI/RCIC, modifications	The RBS RCIC system is designed to withstand a complete loss of offsite ac power. The RCIC system turbine room space coolers are provided with a backup emergency power supply to ensure that pump room temperatures are maintained below equipment qualification limits during periods when offsite power is unavailable. This position is consistent with LRG-II Issue 4-ASB/2-RSB(c).	5.4.6, 9.4.3.2.1.3
•→7 II.K.3.25 Power on pump Seals 7←•	Gulf States Utilities endorsed the BWR Owners' Group position submitted to the NRC Office of Nuclear Reactor Regulation, in letters dated May 26, 1981, September 21, 1981, and September 2, 1982. RBS design employs recirculation	5.4.1

Item and Title	Position	USAR Reference
	pumps manufactured by the Bingham Pump Company. The test simulated a loss of cooling water to the recirculation pump seal coolers which were exposed to a temperature in excess of 270 F. After 5 hours of visually monitoring pump seal leakage, no leakages were detected above 5 gpm. The test results confirmed that a loss of cooling to the Bingham pump seal for 5 hours does not lead to unacceptable seal leakage. Consequently, no change in the RBS design is necessary.	
II.K.3.27 Common reference level	The RBS reactor water level instrumentation provides operators with a common reference point for reactor vessel level measurement, thereby eliminating operator confusion when reading various water level meters. The common reference point for RBS reactor water level instrumentation is located at the bottom of the steam dryer skirt plus 15 inches. This is consistent with LRG-II Position 3-HFS.	4.4.6
II.K.3.28 Qualification of ADS accumulators	The RBS air supply system for the automatic depressurization system (ADS) valves consists of two ASME III Division I, Class 2, air compressors and two non-nuclear safety compressors which feed two separate charging systems for the accumulators. Both ASME III compressors are powered from the preferred ac power supply systems and can be powered by onsite power. The system is sufficient to supply enough air capacity to cycle the valves open 4 to 5 times at atmospheric pressure. The safety grade ASME III compressors, ADS valves, accumulators, and associated equipment and instrumentation are designed to withstand its environment following an accident and perform its function. This position is consistent with LRG-II Issue 8-RSB.	5.2.2.4.1
•→15 II.K.3.30 SB LOCA methods 15←•	SBA models used for RBS are found in Chapter 15.6 of the USAR. Qualification of these models was performed by General Electric and submitted to the NRC Office of Nuclear Reactor Regulation on June 26, 1981, from R. H. Bucholz to D. G. Eisenhut. Qualification of the models has been verbally accepted by the NRC. The Framatome-ANP small break LOCA methodology was accepted by the NRC in the licensing topical: XN-NF-80-19 (P)(A) Volumes 2, 2A, 2B and 2C, Exxon Nuclear Methodolgy for Boiling Water Reactors: EXEM BWR ECCS Evaluation Model, Exxon Nuclear Company, September 1982.	15.6

TABLE 1A-1 (Cont)

Item and Title	Position	USAR Reference
II.K.3.31 Plant-specific analysis	Plant-specific small-break LOCA calculations are provided in Sections 6.3.3 and 15.6 of the USAR.	15.6
•→7 II.K.3.44 Evaluate transients with single failure	GE and the BWR Owners' Group have concluded, based on a representative BWR/6 plant study, that for all anticipated transients in Regulatory Guide 1.70, Revision 3, combined with the worst single failure, the reactor core remains covered with water until stable conditions are achieved. Furthermore, even with more degraded conditions involving a stuck-open relief valve in addition to the worst transient (loss of feed) and worst single failure (failure of high pressure core spray), studies show (NEDO-24708, March 31, 1980) that the core remains covered and adequate core cooling is available during the whole course of the transient. River Bend submitted a response to Generic Letter 81-32 on November 24, 1981, to D.G. Eisenhut, which confirmed that the assumptions and initial conditions used in the BWR Owners' Group generic analyses are applicable to RBS.	15
II.K.3.45 Manual Depressurization 7←•	GE and the BWR Owners' Group have performed analysis of alternate depressurization rates other than full actuation of the ADS. The results of these analyses were submitted to the NRC on December 29, 1980, from D.B. Waters, BWR Owners' Group Chairman, to D.G. Eisenhut. It was concluded from this analysis that: 1) there is little impact on vessel fatigue usage from slower depressurization rates relative to full ADS blowdown, and 2) slower depressurization rates have an adverse impact on core cooling capability. River Bend maintains that the small improvement in vessel fatigue usage resulting from a lower depressurization rate is not sufficient, in light of the corresponding reduction in core cooling capability, to justify a change in depressurization rate. Therefore, the current RBS full ADS blowdown scheme to rapidly depressurize the pressure vessel should not be altered.	5.2.2

TABLE 1A-1 (Cont)

Item and Title	Position	USAR Reference
•→7 II.K.3.46 Michelson Concerns 7←•	GE (R.H. Buchholz) has responded to questions asked by C. Michelson in a letter to the NRC (D. F. Ross) dated February 21, 1980. The GE response applied to BWR/2-6 plants on a generic basis. River Bend personnel reviewed this response for applicability to RBS. This response has been accepted by the NRC via a letter dated June 12, 1981, to D. B. Walters (Chairman-BWROG) from D. G. Eisenhut (NRC).	
III.A.1.1 Emergency preparedness, short term	In accordance with the TMI Action Plan as stated in NUREG-0737, the RBS Emergency Plan has been prepared to meet the criteria established in NUREG-0654 (January 1980); 10CFR50, Appendix E; and Regulatory Guide 1.23.	13.3
III.A.1.2 Upgrade emergency support facilities	The RBS design has been revised to incorporate a Technical Support Center (TSC) and an Emergency Operations Facility (EOF). These facilities are designed in accordance with NUREG-0737 criteria.	13.3
III.D.1.1 Primary coolant outside containment	RBS complies with NUREG-0737, Section III.D.1.1, with the following clarification: The RBS program to reduce leakage to low-as-practical levels for all required post-accident systems outside the containment that could contain highly radioactive fluid will consist of: 1. Monitoring drain sumps to ascertain gross leakage occurring from the systems included in this program. 2. Leak rates for containment isolation valves will be determined by type "C" leak rate tests as defined in 10CFR50, Appendix J, paragraph II.H. Type "C" leak rate tests were performed during the Preoperational Test Program and are periodically performed after initial fuel load in accordance with 10CFR50, Appendix J, paragraph III.D.3.	5.2.5

Item and Title	Position	USAR Reference
	3. Miscellaneous components (i.e., vents, drains, valve packing, valve packing leakoffs, pump packing, pump gland seal leakoffs, etc) were inspected for leakage, and any detected leakage reduced to as-low-as-practical levels, during initial system operations as part of the system preoperational test. After fuel load these components are monitored as part of the RBS surveillance test program. 4. Where it is not possible, practical, or permissible ALARA) to make direct inspections (e.g., high radiation areas, no provision for testing the component, etc), indirect inspections or a suitable substitute are performed. Indirect inspections may consist of, but are not limited to: a. Inspecting floor areas and equipment drain cups for wetting which would occur if leakage were present. b. Monitoring the associated equipment or floor drain sump for excessive flow or fill rates.	
•→7 III.D.3.3 Inplant I_2 radiation Monitoring 7←•	River Bend provided the equipment, procedures, and associated training required to accurately determine airborne iodine concentration in areas where plant personnel may be present during an accident. Where stationary monitoring instrumentation is restricted due to its size, or ALARA considerations are present, portable monitoring instrumentation will be used. Under accident conditions, an area will be available to analyze the sample for iodine concentrations. A sufficient number of samplers will be available to sample the vital areas. Additional information is provided in revised Sections 7.5 and 12.5.	

TABLE 1A-1 (Cont)

Item and Title	Position	USAR Reference
III.D.3.4 Control room habitability	Control room habitability requirements are met by the current RBS design.	6.4

APPENDIX 1B

RIVER BEND STATION POSITIONS IN RESPONSE TO LICENSING REVIEW GROUP (LRG)-II ISSUES

TABLE 1B-1

POSITIONS IN RESPONSE TO LICENSING REVIEW GROUP (LRG)-II ISSUES

Item	Title	Endorsed	USAR Section
1-RSB	Auto-Restart of HPCS	Yes	Appendix 1A, Item II.K.3.21
2-RSB	Design Adequacy of RCIC		
	a. Provide Automatic Restart Capability	Yes	Appendix 1A, Item II.K.3.13
	b. Preventing Inadvertent RCIC System Isolation	Yes	Appendix 1A, Item II.K.3.15
	c. Design Adequacy of RCIC Room Space Cooling	Yes	See 4-RSB
	d. Water Hammer Protection	Yes	5.4.6.1
3RSB	SRV Surveillance	Yes	5.2.2.11
4RSB	Operator Action (10 min vs 20 min)	Yes	Appendix 1A, Item II.K.3.18
5RSB	Control of Post-LOCA Leakage	No	6.3.1.1.3
6RSB	Liquid Flow Through SRV	Yes	Appendix 1A, Item II.D.1 5.4.7.1.5
7RSB	Preclude Vortex Formation	Yes	6.3.2.2
8RSB	Long-Term Operability of ADS	Yes	Appendix 1A, Item II.K.3.28 5.2.2.4.1
9RSB	Deep Draft Pump Operability	Yes	9.2.7.4
10RSB	Flow Control Valve Closure	Yes	6.3
11RSB	Shaft Seizure Event	Yes	15.3.3.3.3
12RSB	Proper Classification of Transients	Yes	15.2.2.1.2.2 15.2.3.1.2.2
13RSB	Removal of High Drywell Pressure Interlock	Yes	6.3.2.2.1 7.3.1.1.1.1
1CPB	Clad Ballooning and Rupture	Yes	NA

TABLE 1B-1 (Cont)

2CPB	Seismic and LOCA Loads on Fuel	Yes	4.2.3.2.15
•→12			
3CPB	Channel Box Deflection	No	NA
4CPB	High Burn-Up Fission Gas Release	Yes	NA
5CPB	Cladding Water-Side Corrosion	Yes	10.4.6.2
6CPB	Inadequate Core Cooling Instrumentation	Yes	Appendix 1A, Items II.F.1/ II.F.2
7CPB	Rod Withdrawal Transient Analysis	No	7.6.1.7 ITS 3.1.3 ITS 3.3.2.1
8CPB	Mislocated or Misoriented Fuel Bundles	Yes	NA
9CPB	Void Coefficient Calculation	Yes	4.3.2.4.2
10CPB	Bounding Rod Worth Analysis	Yes	15.4.9.3.1
11CPB	Core Thermal-Hydraulic Stability	Yes	4.4.4.6
1CSB	Containment Dynamic Loads	Yes	Appendix 6A
2CSB	Hydrogen Control Capability	Yes	6.2.5
3CSB	Periodic Low Pressure Leakage Testing of the Drywell	Yes	6.2.1.1.3.4
1AEB	MSIV Leakage Rate	Yes	6.7.3.5
1ASB	Scram Discharge Volume Modification	Yes	4.6.1.1.2.4.2 ITS 3.1.8
2ASB	Safe Shutdown For Fires	Yes	9.5.1.3
12←•			
3ASB	Line Break in Main Steam Tunnel	Yes	Appendix 3B 3.11

TABLE 1B-1 (Cont)

4ASB	RCIC Pump Room Cooling System	Yes	9.4.3.2.1.3
5ASB	Control Rod Drive System Vessel Inventory Make-Up Test	Yes	NA
1RAB	Exposure from SRV Activation	Yes	12.4.1
2RAB	Routine Exposures Inside Containment	Yes	12.3.2.2.2
3RAB	Radioactivity During Dryer/ Separator Transfers	Yes	12.5.3.2.1
4RAB	Shielding of Transfer Tube and Canal During Refueling	Yes	12.3.2.2.2
1ICSB	Vessel Level Sensing Line Failure	Yes	7.1.5
2ICSB	Redundancy of High/Low Pressure Interlocks	Yes	7.3.1.1.1.3 7.3.1.1.1.4
3ICSB	Failure of Lowest Low/Low Set Point Valves	Yes	7.6.1.8 (B.2)
4ICSB	IE Bulletin 80-06, ESF Reset	Yes	7.3.2.1
5ICSB	Control Systems Failure	Yes	RBS SER Section 7.7.2.2
6ICSB	Procedures Following Bus Failure	Yes	7.5.3
7ICSB	Harsh Environment for Electrical Equipment	Yes	RBS SER Section 7.7.2.1
1PSB	Diesel Generator Reliability	Yes	8.3.1.1.3.9
1GIB	Unresolved Safety Issues	Yes	RBS SER Appendix C
	A-1 Waterhammer A-9 Anticipated Transients Without Scram		See RBS SER, Appendix C

TABLE 1B-1 (Cont)

	A-11 Reactor Vessel Materials Toughness A-17 Systems Interaction in Nuclear Plants A-39 Safety Relief Valve Hydrodynamic Loads A-40 Seismic Design Criteria Short-Term Program A-43 Containment Emergency Sump Reliability A-44 Station Blackout A-45 Shutdown Decay Heat Removal Requirement A-46 Seismic Qualification of Equipment in Operating Plants A-47 Safety Implications of Control Systems A-48 Hydrogen Control Measures and Effects of Hydrogen Burns on Safety Equipment		
1HFS	Special Low Power Testing Program	No	14.2.12.3.28 Appendix 1A, Item I.G.1
2HFS	Reactivity Emergency Procedures	Yes	13.5.2 Appendix 1A, Item I.C.1
3HFS	Common Vessel Level Reference	Yes	7.5.1.1.2 Appendix 1A, Item II.K.3.27
1CHEB	Reactor Coolant Sampling	Yes	9.3.2.6 Appendix 1A, Item II.B.3
2CHEB	Suppression Pool Sampling	Yes	9.3.2.6 Appendix 1A, Item II.B.3

•→12 12←•

TABLE 1B-1 (Cont)

•→12 12←•

3CHEB	Post-Accident Sampling Core Damage Estimates	Yes	9.3.2.6 Appendix 1A, Item II.B.3
1MEB	SRSS for Mechanical Equipment	Yes	3.9.3.4.1.2A
2MEB	RPV Internals Vibration	Yes	3.9.2.4B
3MEB	OBE Stress Cycles	Yes	3.7.3.2 3.9.1.1.5B 3.9.1.1.6B
4MEB	Kuo Sheng Incore Instrumentation Tube Break	Yes	5.3.3.1.4.5
1MTEB	Flued Head Inspectability	NA	Appendix 3D.1
1SEB	Combination of Loads	Yes	3.8.2.4.1
2SEB	Fluid-Structure Interaction	Yes	NA

•→12 12←•