1 INTRODUCTION AND GENERAL DESCRIPTION OF THE PLANT

1.1 INTRODUCTION

Ginna Station is located in Wayne County, near Rochester, New York. The Ginna reactor is a pressurized light water moderated and cooled system designed by Westinghouse. A renewed operating license was issued to R.E. Ginna Nuclear Power Plant by NRC letter dated May 19, 2004. The renewed operating license is effective from the date of issuance through September 18, 2029.

Technical Specification Amendment 115 was issued on April 1, 2014 which approved the transfer of the license for R. E. Ginna Nuclear Power Plant (Ginna) held by R. E. Ginna Nuclear Power Plant, LLC, (Ginna LLC) to Exelon Generation Company, LLC, as approved by Order dated March 24, 2014. The joint venture held between Constellation Energy Nuclear Group, LLC, (CENG) and Électricité de France, S.A., was not modified as part of Amendment 115. The joint venture consists of a 50.01% ownership interest of an ultimate domestic parent Exelon Generation Company, LLC, and a 49.99% ownership interest of an ultimate foreign parent, Électricité de France, S.A., a French corporation (*Reference 1*).

Rochester Gas and Electric filed the application for a construction permit and operating license in October 1965. The construction permit was issued on April 25, 1966. The initial submittal of the Final Facility Description and Safety Analysis Report was filed in March 1969, and the initial provisional operating license was issued on September 19, 1969.

Ginna Station began commercial operation in July 1970, at a licensed output of 1300 MWt and at 420 MW net electrical power. On March l, 1972, the licensed output was increased to 1520 MWt and the net electrical output was increased to 490 MW. In August 1972 RG&E applied for a full-term operating license. The Safety Evaluation Report related to the full-term operating license for the R. E. Ginna Nuclear Power Plant (NUREG 0944) was published in October 1983; Supplement 1 was published in October 1984. The full-term operating license was issued on December 10, 1984. The license was to expire on April 25, 2006. On August 8, 1991, the license was amended to change the expiration date to September 18, 2009, which is 40 years after the date of issuance of the provisional operating license.

During the October 2006 refueling outage, Ginna Station completed the Extended Power Uprate (EPU) Project. The NRC approved the EPU under Technical Specification Amendment No. 97 on July 11, 2006. This license change authorized an approximate 16.8% increase in the steady-state thermal power level from 1520 megawatts thermal to 1775 megawatts thermal. The EPU changed the design electrical rating from 470 MW to 585 MW. Changes to the plant as a result of EPU have been incorporated in the UFSAR.

The R. E. Ginna Nuclear Power Plant was reviewed under Phase II of the Systematic Evaluation Program (SEP). The review began in 1978 and the Integrated Plant Safety Assessment, Final Report, NUREG 082l, was issued by the NRC in December 1982. Supplement 1 to NUREG 0821 was issued in August 1983.

The Ginna Station primary coolant system configuration consists of two hot legs, two U-tube steam generators, a pressurizer, and two cold legs with a reactor coolant pump in each cold leg. The secondary system consists basically of the turbine generator, the condenser, and the feedwater and condensate systems. Auxiliary equipment includes a radioactive waste disposal system, fuel handling system, main transformer, circulating water system, engineered safety features systems, and all auxiliaries, structures, and onsite facilities required to provide for a complete and operable nuclear power plant. A more detailed list of structures, systems, and components is provided in Section 3.2. The turbine and condenser system as well as the nuclear steam supply system were designed and supplied by Westinghouse. The remainder of the plant was designed by either RG&E or Gilbert Associates, Incorporated. The replacement steam generators were designed and supplied by Babcock and Wilcox International (BWI).

The reactor containment structure was designed by Gilbert Associates. It is a reinforced-concrete, vertical right cylinder with a flat base and a hemispherical dome. A welded steel liner is attached to the inside face of the concrete shell to provide for leaktightness. The containment cylinder is founded on rock by post-tensioned rock anchors. The cylinder wall is prestressed vertically by tendons coupled to the rock anchors.

Ginna Station is located on the south shore of Lake Ontario, which is the source of circulating water and the ultimate heat sink. The site initially contained 338 acres. In 1973 the site, including the switchyard, was increased to 488 acres. As a result of the purchase of Ginna Station by Constellation Energy in 2004, the site was reduced to approximately 426 acres.

REFERENCES FOR SECTION 1.1

1. Letter from Nadiyah S. Morgen, NRC, to Mary G. Korsnick and Bryan P. Wright, Constellation Energy Group: R.E. Ginna Nuclear Power Plant – Issuance of Amendment to Conform the Renewed Facility Operating License to Reflect the Direct Transfer of Operating Authority (TAC No. MF2588), dated April 1, 2014.

1.2 GENERAL PLANT DESCRIPTION

1.2.1 SITE AND ENVIRONMENT

The site is on the south shore of Lake Ontario l6 miles east of Rochester, New York, an urban area of about 700,000. The area immediately around the site is sparsely populated and is utilized primarily for farming. The site, in open, rolling terrain, is well ventilated and is not generally subject to severe flooding. Liquids released to the lake from the site will move predominately eastward and diffuse slowly. Hurricanes have not seriously affected the site region and tornadoes and severe ice storms are rare. Onsite measurements indicate that ground water within the site will flow to the lake and will not affect offsite wells.

The site has sound bedrock on which major structures are founded and is in a seismologically quiet region. It is within l50 miles of the St. Lawrence Valley area, where earthquakes of Richter magnitude 7 have been experienced, and 35 miles from the area around Batavia-Attica which has experienced moderate seismological activity of smaller magnitudes.

1.2.2 SUMMARY PLANT DESCRIPTION

The inherent design of the pressurized water reactor ensures that the probability of release of significant quantities of fission products to the atmosphere is low. Four barriers exist between the fission product accumulation and the environment. These are the uranium dioxide fuel matrix, the fuel cladding, the reactor vessel and coolant loops, and the reactor containment. The consequences of a breach of the fuel cladding are greatly reduced by the ability of the uranium dioxide lattice to retain fission products. Escape of fission products through a fuel cladding defect would be contained within the pressure vessel, loops, and auxiliary systems. A breach of these systems or equipment would release the fission products to the reactor containment where they would be retained. The reactor containment is designed to adequately retain these fission products under the most severe accident conditions, the design-basis loss-of-coolant accident. This accident and its consequences are analyzed in Section 15.6.

Several engineered safety features have been incorporated into the plant design to reduce the consequences of a loss-of-coolant accident. These safety features include a safety injection system (Emergency Core Cooling System (ECCS)). This system automatically delivers borated water to the reactor vessel for cooling under high and low reactor coolant pressure conditions. The safety injection system also serves to insert negative reactivity into the core in the form of borated water during an uncontrolled plant cooldown following a steam line break or an accidental steam release. Other safety features which have been included in the reactor containment design are a containment air recirculation, cooling, and filtration system, which would effect a depressurization of the containment following a loss of coolant and provide for iodine filtration if fission products are released from the core; and a containment spray system which would depressurize the containment and remove elemental iodine from the atmosphere by a washing action. The containment spray system and containment air recirculation, cooling, and filtration system are redundant containment heat removal systems. Additional engineered safety features are listed in Section 3.2.

1.2.3 STRUCTURES

1.2.3.1 General

The major structures are a reactor containment, auxiliary building, intermediate building, control building, turbine building, screen house, all volatile-treatment or condensate demineralizer building, standby auxiliary feedwater (SAFW) building, diesel generator buildings, and the service building containing offices, shops, and laboratories. A general plan of the building arrangement is shown in Figure 1.2-1. Several drawings in the 33013-2100 series show the general internal layout of the buildings. Structures containing equipment that is associated with, or required for, operating the plant are part of the power block. Additionally, the old steam generator storage facility is located northwest of the plant outside the security fence.

The reactor containment is a vertical, cylindrical reinforced-concrete type with prestressed tendons in the vertical wall, reinforced-concrete ring anchored to the bedrock and a reinforced hemispherical dome. The containment is designed to withstand the internal pressure accompanying a loss-of-coolant accident or main steam line break and to provide adequate radiation shielding for both MODES 1 and 2 and accident conditions.

1.2.3.2 Containment

The reactor containment structure is a reinforced-concrete vertical right cylinder with a flat base and a hemispherical dome. A welded steel liner is attached to the inside face of the concrete shell to ensure a high degree of leaktightness. The thickness of the liner in the cylinder and dome is 3/8-in. and in the base it is 1/4 in. The cylindrical reinforced-concrete walls are 3 ft 6 in. thick, and the concrete hemispherical dome is 2 ft 6 in. thick. These thicknesses are nominal values. The true relevant engineering values are dependent on the specific location in the structure and the loading condition that is present. The concrete base slab is 2 ft thick with an additional thickness of concrete fill of 2 ft over the bottom liner plate. The containment structure is 99 ft high to the spring line of the dome and has an inside diameter of 105 ft. The containment vessel provides a minimum free volume of approximately 1,000,000 ft^3. Access is provided during operation by means of two personnel airlocks designed with an interlocked single-door-opening feature that is leak testable at containment design pressure between doors. The open and closed status of each door is indicated in the control room.

The major components of the reactor coolant system are located within the containment structure. The containment structure provides a physical barrier to protect the equipment from natural disasters and shielding to protect personnel from radiation emitted from the reactor core while at power.

The reactor vessel is located in the center of the containment structure below ground level. Extending around the reactor vessel is a stainless-steel-lined refueling cavity. During MODE 6 (Refueling) operations, the refueling cavity is flooded with borated water to provide shielding of the irradiated fuel being removed from the reactor vessel.

Thick reinforced-concrete walls are located around the major reactor coolant system components to serve as shielding for plant personnel. These walls also serve as a missile barrier to prevent damage to the containment wall and to components of the safety injection system should a failure occur to one of the reactor coolant system components located inside the walls.

The containment houses the following major equipment (see Drawings 33013-2101, 33013-2102, 33013-2105, 33013-2106, 33013-2107, 33013-2113, 33013-2114, 33013-2115, 33013-2131, and 33013-2132):

1. Reactor coolant loop piping, reactor coolant pumps, and steam generators.
2. Pressurizer.
3. Pressurizer relief tank.
4. Reactor coolant drain tank and pumps.
5. Containment recirculation filtering and cooling units (four).
6. Safety injection system accumulators.
7. Refueling cavity and equipment.

1.2.3.3 Auxiliary Building

The auxiliary building is located just south of the containment. The auxiliary building houses the major support and engineered safety features equipment for plant operation. The auxiliary building is a restricted area and normal exit is from the intermediate building (hot side), as shown in Drawing 33013-2116.

The auxiliary building has three major levels and a subbasement level pit which contains the residual heat removal pumps. The refueling water storage tank (RWST) extends through all three levels. The following is a list of major equipment on each level of the auxiliary building.

Auxiliary Building Basement (See Drawing 33013-2103)

1. Chemical and volume control system holdup tanks.
2. Residual heat removal pumps (subbasement).
3. Residual heat removal heat exchangers.
4. Spent fuel pool pump.
5. Residual heat pump cooling.
6. Boric acid evaporator.
7. Gas stripper.
8. Waste holdup tank.
9. Various operations panels.
10. Waste evaporator (system physically removed in 1999).
11. Blender room.
12. Spent resin tanks.

13. Safety injection filters.
14. Seal injection filters.
15. Containment spray pumps.
16. Nonregenerative heat exchanger.
17. Seal return filter and cooler.
18. Charging pump rooms and accumulator.
19. Sodium hydroxide tank and leakoff tank.
20. Safety injection pumps (three).
21. Safety injection accumulator makeup pump.

Auxiliary Building - Intermediate Level (See Drawing 33013-2108)

1. Spent fuel pool filter and heat exchanger.
2. Chemical and volume control system holdup tanks.
3. Residual heat removal heat exchangers.
4. Waste gas compressors and gas stripper.
5. Gas decay tanks (four).
6. Reactor coolant filter.
7. Volume control tank.
8. Concentrates holding tank and transfer pump.
9. Demineralizer vault.
10. High efficiency particulate air filters.
11. Nonregenerative heat exchanger.
12. 480-V bus 16 (vital bus).
13. Charcoal filter unit.
14. Motor control center 1D.
15. Motor control center 1M.

Auxiliary - Building Operating Floor (See Drawing 33013-2116)

1. Decontamination pit.
2. Spent fuel storage pool, crane, and transfer canal.
3. New fuel unloading area.
4. New fuel storage racks.
5. Auxiliary building maintenance shop.
6. Crane bay.
7. Refueling water storage tank (RWST) (all levels).

8. Component cooling pumps.
9. Component cooling water heat exchangers and surge tank.
10. Boric acid demineralizers.
11. Monitor tanks and pumps.
12. Waste condensate tanks.
13. Reactor makeup water tank and pumps.
14. Drumming station and drum storage area.
15. 480-V bus 14 (vital bus).
16. Auxiliary building supply fan and filter.
17. Boric acid batching tank.
18. Boric acid storage tank and boric acid transfer pumps.
19. Waste condenser demineralizer.
20. Motor control center 1C.
21. Motor control center 1L.
22. Motor control center 1E.
23. Vendor supplied demineralization system.

1.2.3.4 Intermediate Building (See Drawings 33013-2101, 33013-2102, 33013-2105, 33013-2106, 33013-2107, 33013-2113, 33013-2114, 33013-2115, and 33013-2121)

The intermediate building surrounds the containment building to the west and north and joins the service building and turbine building. It is divided into two sections called the hot side (restricted area access) and the cold side.

<u>Hot Side (Restricted Area Access)</u>

The hot side is west of the containment building and joins the service building, intermediate building cold side, and auxiliary building. Personnel enter and exit the intermediate building hot side, at the access control area.

The intermediate building hot side extends from the access control area to the personnel door to the auxiliary building, spent fuel pool (SFP) area. The intermediate building hot side has four levels, plus a subbasement for access to the containment tendons. In addition, there is a mezzanine level for access to the containment personnel hatch. The following equipment is among that located in the intermediate building cold side:

1. Primary sample room.
2. Post-accident sample panel.
3. Hydrogen recombiner panel.
4. Auxiliary building exhaust fans A, B, and C.
5. Auxiliary building HEPA filter bank.

6. Intermediate building exhaust fans A and B.
7. Access control area exhaust fans A and B.
8. Access control area HEPA and charcoal filter banks.

Cold Side

The intermediate building cold side is a radiologically unrestricted area. The intermediate building cold side provides access to the cable tunnel area. The building is constructed to partially surround the containment structure to the north and west and house its support equipment.

Access to the intermediate building cold side is normally made from the turbine building. Doors from the cold side to the hot side are available but not normally used. The following equipment is among that located in the intermediate building cold side:

1. Turbine-driven auxiliary feedwater pump (TDAFW).
2. Motor-driven auxiliary feed pumps (MDAFW) (two).
3. Rod control power panels.
4. Rod control logic cabinets.
5. Rod drive motor-generator sets and power panels.
6. Reactor trip and bypass breakers.
7. Auxiliary building and containment ventilation units.
8. Safety and relief valves (main steam).
9. Purge exhaust fans.
10. Radiation monitors (e.g., R-11, R-12).
11. Main steam and feedwater lines.

1.2.3.5 Turbine Building

The turbine building is located north of the intermediate building. The turbine building houses the major secondary system equipment and systems, including the main turbine, generator, and condenser (see Drawing 33013-2140 and Drawing 33013-2141). The following equipment is located on each level of the turbine building:

Basement level (See Drawing 33013-2104)

1. Main feedwater pumps (2).
2. Fire service water storage tank.
3. Turbine oil reservoir and purifier.
4. Turbine oil pumps (on top of reservoir).
5. Steam dump valves.
6. Circulating water inlet and outlet headers.

7. Seal-oil unit.
8. Blowdown recovery system.
9. Bus duct cooling fans.
10. Condensate coolers.
11. Condensate pumps (three).
12. Condensate booster pumps (three).
13. Heater drain tank.
14. Heater drain tank pumps.
15. Motor control center 1A.

Intermediate Level-Mezzanine (See Drawing 33013-2112)

1. Low-pressure heaters (inside of condenser).
2. Moisture separator reheater units (four).
3. Main feedwater regulating valves.
4. Hydrazine and NH_4OH addition tanks.
5. Feedwater heaters 1A, 1B, 2A, 2B, 3A, 3B, 4A, 4B, 5A, and 5B.
6. Air ejector and condenser.
7. Gland exhaust condenser.
8. Generator bus ducts.
9. Main power panels and motor control centers: 4l60-V buses 11A, 11B, 12A, 12B; 480-V bus 13, 15; and motor control center 1B.
10. Secondary sampling station.
11. Electro-hydraulic oil system.

Operating Floor (See Drawing 33013-2120)

1. Main turbine and generator.
2. Intercept and low pressure stop valves.
3. Entrance to main control room.

1.2.3.6 Control Building

The control building is adjacent to the turbine building and consists of three floors (see Drawings 33013-2123, 33013-2124, 33013-2125 and 33013-2136). The main control room is on the upper floor. The relay room is directly below the control room and houses relay racks and the multiplexer (MUX) room. The battery rooms and the air handling room are on the lowest level of the control building.

1.2.3.7 All-Volatile-Treatment Building

The all-volatile-treatment building houses demineralizers and other equipment necessary for the condensate polishing system to allow all-volatile-treatment of secondary water (see Drawing 33013-2111).

The technical support center is located on the second floor of the all-volatile-treatment building and houses the computers and equipment, including emergency power supplies (diesel generator and batteries), necessary to provide the staff technical support during an emergency event (see Drawing 33013-2119).

1.2.3.8 Standby Auxiliary Feedwater Pump Building

The standby auxiliary feedwater pump (SAFW) building is located on the southeast corner of the auxiliary building and houses the two standby auxiliary feedwater pumps (SAFW). The building is a Seismic Category I concrete structure supported by caissons (see Drawing D-024-017)

1.2.3.9 Screen House

The screen house is located north of the turbine building on Lake Ontario and houses the main circulating water inlet lines and pumps; the service water (SW) pumps (four); 480-V switchgear buses 17 and 18, the diesel fire pump, the motor-driven fire pump, and motor control center G (MCCG) (see Drawing 33013-2143).

1.2.3.10 Service Building

The service building is located at the west end of the auxiliary building. This building provides the office spaces for the administrative staff at Ginna Station (see Drawings 33013-2109, 33013-2110, 33013-2117, and 33013-2118).

The service building has two levels. The basement level is comprised of storerooms, machine shops, maintenance areas, etc. The basement level also contains a water treatment area, Material and Test Equipment area, and maintenance management offices.

The ground level consists primarily of offices for groups such as Operations, Maintenance Support, Radiation Protection, and Chemistry. The ground level also contains the cafeteria, fire brigade response room, locker rooms, plant management offices, and Instrument and Control office/shop.

1.2.3.11 Diesel Generator Building

The diesel generator building adjoins the turbine building on the east end of the north wall opposite the control building. The building is a one-story reinforced-concrete structure that houses the emergency diesel generators.

1.2.3.12 Old Steam Generator Storage Facility

The old steam generator storage facility (OSGSF) is a reinforced concrete building which will provide long-term storage of the two old steam generators and the attached insulation material. Also stored in the OSGSF are the control rod drive mechanisms (CRDM) and related equipment removed from the plant during the 2003 refueling outage.

The OSGSF is a stand-alone facility located outside the existing security perimeter fence and will have no interface with permanent plant structures.

1.2.3.13 Canister Preparation Building

A Canister Preparation Building (CPB) located south of the Auxiliary Building was constructed at the Ginna Nuclear Generating Station for the general purpose of performing spent fuel Dry Shielded Canister (DSC) and Transfer Cask handling and preparation activities. As of September 2020, the CPB also serves the purpose of performing spent fuel activities involving handling and preparation of the Multi-Purpose Canister (MPC) and Holtec International Transfer Cask (HI-TRAC).

The CPB superstructure is designed to meet the applicable requirements of 10 CFR 50, the Ginna UFSAR, and the Building Code of New York State. The CPB superstructure is a seismic II/I structure such that the building cannot adversely impact the transfer cask, Dry Shielded Canister (DSC), Multi-Purpose Canister (MPC), or Auxiliary Building when fuel is present.

The CPB and large overhead door opening through the south wall of the Auxiliary Building are considered functionally an extension of the Auxiliary Building. The CPB is part of the Auxiliary Building and for NEIL insurance purposes will be considered to act as an Auxiliary Building Truck Bay.

A 30-ton building crane was installed in the CPB. The 30-ton crane is supported on a crane structure mounted to the building columns, which permits operation of the crane in the north-south direction. The 30 ton trolley operates in the east-west direction.

The new 125-ton single failure proof cantilevered gantry crane has a stationary runway mounted to an embedded steel support system. A rolling bridge is mounted on top of the stationary runway. A main trolley is mounted on the rolling bridge with a flying trolley mounted to the main trolley.

1.2.3.14 ISFSI Transfer Path, Storage Pad, and Canister Transfer Pit

The Independent Spent Fuel Storage Installation (ISFSI) pad site is located north and west of the station power block and south of the Meteorological (MET) Tower. The ISFSI pad site was initially constructed to provide storage capacity for 30 loaded Transnuclear (TN) Dry Shielded Canisters (DSCs). In 2018, plans to transition the ISFSI to store both Holtec and TN casks were initiated. The ISFSI pad may provide storage capacity for 10 loaded TN DSCs and up to 24 Holtec International Multi-Purpose Canisters (MPCs). Storage capacity is intended to satisfy spent fuel storage requirements through the end of the extended plant life with the reactor defueled and the SFP full. The ISFSI site was comprised of a reinforced concrete foundation slab (pad) surrounded by a reinforced concrete approach slab (apron) and a concrete haul path to facilitate the transfer of the fuel.

The ISFSI pad was placed on top of the soil mixed elements that stabilized the soil. These soil mixed elements extend approximately 20 feet outside of the ISFSI pad on all sides.

The initial construction of the ISFSI pad and aprons was performed while the area was outside the Protected Area boundary. Upon completion of the initial construction, the Protected Area boundary was extended to include the ISFSI pad and aprons.

Once plans were in place to transition to the Holtec International System, a Canister Transfer Pit (CTP) and additional approach slab were constructed to the northeast of the ISFSI pad to facilitate transfer operations of the MPC between the HI-TRAC transfer cask and the HI-STORM overpack.

1.2.3.15 Administration Building

A new Administration Building was constructed in 2005 to house more personnel onsite. The Administration Building is a two-story structure located on the west side of the Service Building. This building contains conference rooms, an auditorium, and offices for groups such as Scheduling, Planning, Information Technology, Procurement, and Finance. The Administration Building also contains the first aid/Fitness for Duty office. Records Management, which includes a protected vault, is located on the first floor of the building.

The Outage Control Center (OCC) is also located in the Administration Building. The OCC is used at the Operational Support Center (OSC) during plant emergencies.

1.2.4 NUCLEAR STEAM SUPPLY SYSTEM

The nuclear steam supply system consists of a pressurized water reactor, reactor coolant system, and associated auxiliary fluid systems. The reactor coolant system is arranged as two closed reactor coolant loops connected in parallel to the reactor vessel, each containing a reactor coolant pump and a steam generator. An electrically heated pressurizer is connected to one of the loops (loop B).

The reactor core is composed of uranium dioxide pellets enclosed in zircaloy, ZIRLO® High Performance Fuel Clad Material[1], or Optimized ZIRLO™ High Performance Fuel Clad Material[1] tubes with welded end plugs. The tubes are supported in assemblies by a grid

[1] **ZIRLO®** and **Optimized ZIRLO™** fuel rod cladding are trademarks or registered trademarks of Westinghouse

structure. The mechanical control rods consist of clusters of stainless steel clad absorber rods inserted into guide tubes located within the fuel assembly. The core fuel is divided into several regions.

The replacement steam generators are vertical U-tube units containing Inconel tubes. Integral separating equipment reduces the moisture content of the steam at the steam generator outlet nozzle to 0.1% or less.

The reactor coolant pumps are vertical, single-stage, centrifugal pumps equipped with controlled leakage shaft seals.

Auxiliary systems are provided to add makeup water to the reactor coolant system, purify reactor coolant water, provide chemicals for corrosion inhibition and reactor control, cool system components, remove residual heat when the reactor is shut down, cool the spent fuel storage pool, sample reactor coolant water, provide for emergency safety injection, vent and drain the reactor coolant system, and for other purposes.

1.2.5 REACTOR AND PLANT CONTROL

The reactor is controlled by a coordinated combination of chemical shim and mechanical control rods. The control system allows the plant to accept step load increases of 10% and ramp load increases of 5% per minute over the load range of 12.8% to 100%. Similar step and ramp load reductions are possible over the range of 100% to 12.8%.

Complete supervision of both the nuclear and turbine generator plants is accomplished from the central control room. This supervision includes the capability to test periodically the operability of the Reactor Trip System (RTS).

1.2.6 WASTE DISPOSAL SYSTEM

The waste disposal system provides all equipment necessary to collect, process, and prepare for disposal all potentially radioactive liquid, gaseous, and solid wastes produced as a result of reactor operation.

Liquid wastes requiring cleanup before release are collected and processed by a vendor supplied demineralization system. After appropriate cleaning and filtering, the liquid is collected in the chemical and volume control system monitor tank A or B for ultimate release to the circulating water discharge canal at a concentration below 10 CFR 20 limits. The spent demineralizer resin is packaged and shipped from the site for ultimate disposal in an authorized location. Liquid wastes were also processed by the waste evaporator system until 1990 when use of the evaporator was discontinued. The waste evaporator package was physically removed in 1999.

Gaseous wastes are collected and stored until their radioactivity level is low enough so that discharge to the environment does not create radioactivity concentrations above 10 CFR 20 limits.

Solid wastes including evaporator concentrates are packaged and shipped from the site for ultimate disposal in an authorized location. Wet solid wastes are solidified. Dry solid wastes are shipped in bulk form to a vendor for volume reduction and packaging for

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delivery to a disposal site.

Operating procedures generally limit normal effluents to within 10 CFR 50, Appendix I, limits. Sanitary waste from Ginna Station is piped into the Town of Ontario, New York, sewer system.

1.2.7 FUEL HANDLING SYSTEM

The reactor is refueled with equipment designed to handle spent fuel under water from the time it leaves the reactor vessel until it is placed in a cask for shipment from the site. Underwater transfer of spent fuel provides an optically transparent radiation shield, as well as a reliable source of coolant for removal of decay heat.

1.2.8 TURBINE AND AUXILIARIES

The turbine is a tandem-compound, three-cylinder, 1800-rpm unit having 40-in. exhaust blading in the low-pressure elements. Four combination moisture separator reheater units are employed to dry and superheat the steam between the high- and low-pressure turbine cylinders.

A single-pass deaerating, radial flow surface condenser, steam jet air ejectors, three half-capacity condensate pumps, three half-capacity condensate booster pumps, two half-capacity main feedwater pumps, and five stages of feedwater heaters are provided. One preferred auxiliary turbine-driven (TDAFW), two preferred auxiliary motor-driven (MDAFW), and two standby auxiliary motor-driven feedwater pumps (SAFW) are available in case of a complete loss of offsite power.

1.2.9 ELECTRICAL SYSTEM

The main generator is a 1800 rpm, three-phase, 60-cycle, hydrogen inner cooled unit. The main step-up transformer is a conventional two-winding forced oil air cooled unit.

The station service system consists of auxiliary transformers, 4160-V and 480-V switchgear, 480-V motor control centers, and 125-V dc equipment.

Emergency power supplied by one of two diesel-engine-driven generators is capable of operating postaccident safeguards equipment or safe shutdown equipment to ensure an acceptable plant response.

1.2.10 ENGINEERED SAFETY FEATURES PROTECTION SYSTEMS

The engineered safety features protection systems provided for the station have sufficient redundancy of component and power sources such that under the conditions of a design-basis loss-of-coolant accident, the system can, even in the event of a single failure, maintain emergency core cooling, maintain the integrity of the containment, and perform other safeguards functions to ensure that postaccident exposures are maintained below the guidelines of 10 CFR 100.

The systems provided are:

A. The containment system, which provides an essentially leaktight barrier against the escape of fission products. The containment penetrations and liner weld seams are provided with a leak test system, which can be utilized to check the integrity of these two locations that are the most likely sources of containment leakage. Very low leakage requirements are also imposed on the containment isolation valves.

B. The safety injection system, which provides borated water to cool the core by injection into the cold legs of the reactor coolant loops and by injection over the top of the core through nozzles that penetrate the reactor vessel.

C. The containment recirculation fan cooler (CRFC) and filtration system, which provides a dynamic heat sink to cool the containment atmosphere and filtration of the containment atmosphere to remove airborne particulate and halogen fission products that form the source for potential public exposure. The system utilizes the normal containment ventilation and cooling equipment in addition to the charcoal filters.

D. The containment spray system, which provides a spray of cool, chemically treated borated water to the containment atmosphere to provide additional heat sink and iodine removal capability together with the containment air recirculation cooling and filtration system.

E. The hydrogen recombiners, which limit the concentration of hydrogen in containment following a loss-of-coolant accident.

F. Auxiliary systems, which serve to ensure the operability of the above systems.

1.2.11 DESIGN HIGHLIGHTS

The design of Ginna Station was based upon proven concepts which have been developed and successfully applied in the construction of pressurized water reactor systems. In subsequent sections, a few of the design features of Ginna Station are listed that represent slight variations or extrapolations from units such as San Onofre and Connecticut-Yankee, which were licensed to operate before Ginna Station.

1.2.11.1 Power Level

The power level is 1520 MWt. This is greater than the capability of the San Onofre plant, but smaller than the capability of the Connecticut Yankee plant (1825 MWt). Therefore, this power level does not represent any significant variation from the power levels of other pressurized water reactors in operation at the time Ginna Station was licensed. In 2006 the licensed core power level for Ginna Station was increased to 1775 MWt.

1.2.11.2 Reactor Coolant Loops

The reactor coolant system for Ginna Station consists of two loops, as compared with three loops for San Onofre and four loops for Connecticut Yankee, and required an attendant increase in the size and capacity of the reactor coolant system components such as the reactor coolant pumps, piping, and steam generators. These increases represented reasonable engineering extrapolations of existing and proven designs at the time and, as such, the components of the reactor coolant system were designed for conditions exceeding operation at 1520 MWt.

1.2.11.3 Peak Specific Power

Based on the design hot channel factors, operation at 1520 MWt produces a peak specific power of 13.5 kW/ft for a 12 month fuel cycle (with F_Q of 2.32) and 14.2 kW/ft for an 18 month fuel cycle (with F_Q of 2.45). For an 18 month cycle at 1775 MWt core power with a design hot channel F_Q=2.60, the resulting peak specific power is 18.25 kW/ft.

1.2.11.4 Fuel Clad

The initial fuel rod design for Ginna Station utilized zircaloy as a clad material, which has proven successful in other operations. ZIRLO® clad material was also being used , commencing in 1999, which then transitioned to the more advanced Optimized ZIRLO™ clad material in 2018.

1.2.11.5 Fuel Assembly Design

The fuel assembly is a canless type with the basic assembly consisting of the rod cluster control guide thimbles fastened to the grids and the top and bottom nozzles. The fuel rods are held by the grids and grid springs, which provide lateral and axial support.

Ginna Station was initially fueled with Westinghouse fuel. Starting with cycle 8 (1978), Exxon fuel was used. Starting with cycle 14 (1984), Westinghouse (optimized fuel assemblies) fuel is being used. Commencing with cycle 28 (1999), Westinghouse (VANTAGE +) fuel is being used. Commencing with cycle 33 in 2006 as part of the plant uprate to 1775 MWt, Ginna started using Westinghouse 422V+ fuel assemblies.

1.2.11.6 Engineered Safety Features

The engineered safety features provided are of the same types provided for the Connecticut Yankee plant augmented by borated water injection accumulators. There is a safety injection system of the Connecticut Yankee type which can be operated in part (any two of three high-head pumps and any one of two low-head pumps) from emergency onsite diesel power. The system design is such that it can be tested while the plant is at power. There is containment recirculation fan cooler (CRFC) and filtration for post-loss-of-coolant conditions inside the containment that utilize the normal ventilation system flow path so that deterioration is not expected. Provisions are made for periodic testing to determine the condition of the filter material. A containment spray system provides cool, borated water sprayed into the containment atmosphere for additional cooling and iodine removal capacity.

1.2.11.7 Emergency Power

In addition to the multiple ties to outside sources for emergency power, two diesel generator units are provided as backup power supplies in case of a loss of all outside power. Each generator is capable of operating sufficient safeguards equipment to ensure an acceptable post loss-of-coolant containment pressure transient.

1.2.12 STATION WATER USE

The total nominal flow of circulating water through the turbine condenser and service water (SW) systems is about 400,000 gpm. Approximately 340,000 gpm is used in the turbine condenser system and the rest is available for the service cooling supply and fire protection systems. In addition, domestic quality water at a flow of about 100,000 gal/day is purchased from the Ontario Water District, Town of Ontario, for drinking, sanitary purposes, auxiliary boiler feed, and condensate makeup and polishing.

Lake Ontario is the source of the circulating water, which is taken through the eight 17.3 ft. wide by 10 ft. high ports of the submerged octagonal intake structure that lies about 3100 ft. offshore in about 33 ft. of water at mean lake level, 244.7 ft. [International Great Lakes Datum, 1955 (IGLD 1955)]. All 24 plant original heater racks have been removed from the faces of the intake structure to mitigate the effects of frazil ice on plant operations.

Refer to Section 10.6.2.1 for a current description of the configuration of the intake structure screens. The water flows by gravity through a 10 ft. diameter concrete lined tunnel into the screen house, where it passes through a fine mesh traveling screen before being pumped through the turbine condenser or service water (SW) system. The water from these two systems is combined and is released to the discharge canal, which opens into Lake Ontario at the shoreline. The discharge canal is protected from large debris by a submarine net placed inside the canal near the shoreline.

1.2.13 FACILITY SAFETY CONCLUSIONS

The safety of the public and station operating personnel and reliability of plant equipment and systems were primary considerations in the plant design. The approach taken in fulfilling the safety consideration was three fold. First, careful attention was given to the design so as to prevent the release of radioactivity to the environment under conditions which could be hazardous to the health and safety of the public. Second, the plant was designed so as to provide adequate protection for plant personnel wherever a potential radiation hazard exists. Third, reactor systems and controls were designed with a great degree of redundancy and fail-safe characteristics.

Based on the overall design of the plant and its engineered safety features and the analysis of the possible incidents and of design basis accidents, it was concluded that Ginna Station can be operated with no undue hazard to the public health and safety.

1.3 COMPARISON TABLES

The information presented in Section 1.3.1 provides a comparison of the R. E. Ginna Nuclear Power Plant as originally licensed at 1300 MWt output and as originally uprated to 1520 MWt output to Point Beach Units 1 and 2 as originally licensed. It also compares Ginna as originally licensed at 1300 MWt to San Onofre Unit 1 and Connecticut Yankee. The information presented in Section 1.3.2 identifies the significant changes made in the Ginna Nuclear Power Plant design between submittal of the PSAR and submittal of the original FSAR. In general, neither of these Sections have been updated. The information contained in them may or may not represent the current design of the Ginna Nuclear Power Plant.

1.3.1 COMPARISONS WITH SIMILAR FACILITY DESIGNS

The design parameters of the Ginna Nuclear Power Plant for 1520 MWt are presented in Table 1.3-1 along with the comparisons of the major parameters from the initial design rating of 1300 MWt for Ginna Nuclear Power Plant and the original Point Beach, Units 1 and 2 design rating. Table 1.3-2 presents a comparison of the Ginna Station steam and power conversion design parameters to those of San Onofre Unit 1 and Connecticut Yankee as presented in the original FSARs of the three plants.

In 2006, Ginna uprated the licensed power level from 1520 MWt to 1775 MWt. Section 1.3.3 compares the Ginna uprated parameter to comparable parameters for another Westinghouse 2 loop plant.

1.3.2 COMPARISON OF FINAL AND PRELIMINARY SAFETY ANALYSIS REPORT INFORMATION (HISTORICAL)

1.3.2.1 Partial Length Rod Cluster Control Assemblies

Previously abandoned in place, partial length rod assemblies were removed and not replaced by PCR 2001-0042, Reactor Vessel Closure Head Replacement.

1.3.2.2 Burnable Shim Rods

Burnable shim rods were added to ensure a zero or negative moderator temperature coefficient of reactivity at all times. (These are no longer used.)

1.3.2.3 Safety Injection System Trip Signal

The actuating signal for the safety injection system was revised to increase the initiation reliability and to increase protection in the case of a steam line rupture.

1.3.2.4 Containment Spray System Signal

The actuating signal for the containment spray system was revised to operate from two sets of two-out-of-three containment high-pressure signal channels.

1.3.2.5 Safety Injection System Accumulators

Two accumulators were added to the safety injection system to provide short-term core cooling before the injection pumps become effective for postulated large area primary system rupture.

1.3.2.6 Spray Additive

The containment spray additive for increasing inorganic iodine removal rate in case of a primary system rupture was changed to sodium hydroxide. (See Chapter 6).

1.3.2.7 Rod Stop and Reactor Trip on Startup

The automatic rod stop signal is actuated by an intermediate range flux level setting, and the reactor trip signal on startup is supplied by a high flux level setting.

1.3.2.8 Miniature Neutron Flux Detectors

Four miniature neutron flux detectors capable of traversing 36 thimbles replace the original three detectors in 25 thimbles to provide more detailed flux mapping during core physics tests.

1.3.2.9 Core Thermocouples

Fewer core thermocouples are provided (39 in place of 45).

1.3.2.10 Initial Leak Rate Test Method

The initial leak rate testing of the containment makes use of the absolute method instead of the reference volume method to provide higher sensitivity at low leak rates.

1.3.2.11 Auxiliary Building Ventilation Filters

Absolute and charcoal filters are added to the auxiliary building ventilation system (ABVS) to reduce air activity levels in case of recirculation system components leakage following a loss-of-coolant accident.

1.3.2.12 Control Center Buses

The 480-V system buses are increased from four to six to provide greater operating flexibility under single component failure or emergency power conditions.

1.3.2.13 Condenser Circulating Water Flow

The condenser circulating water flow was increased to 334,000 gpm.

1.3.2.14 Ramp Loading Range

The ramp loading range is increased from 15% to 95% up to 15% to 100% of full load.

1.3.2.15 Condensate Storage Tanks Capacity

The two condensate storage tanks total capacity is 60,000 gal (decreased from 72,000 gal or 6.5 hr versus 8 hr capacity). A third tank with a 100,000 gal. capacity has been added. It is located outdoors next to the all-volatile-treatment building. See Section 9.2.4.

1.3.2.16 Fuel Transfer System Drive

An air-motor drive replaces the cable drive for the fuel transfer conveyor car. The air-motor was removed by PCR 2005-0033. See Section 9.1.4.3.4.

1.3.2.17 Steam Line Flow Nozzles

Steam line flow nozzles were incorporated to limit the consequences of a steam line rupture.

1.3.3 COMPARISON OF UPRATE PARAMETERS

In 2006 Ginna implemented a power uprate to increase core licensed power from 1520 MWt to 1775 MWt. Prior to the Ginna uprate, Kewaunee which is a Westinghouse 2 loop plant similar to Ginna, had also implemented a power uprate. A comparison of the key NSSS parameters at the uprated power level for both plants is presented in Table 1.3-3.

Table 1.3-1

COMPARISON OF DESIGN PARAMETERS WITH POINT BEACH
[Represents original design parameters for plants listed and may not represent current design of the plants]

	Point Beach Units 1 and 2 1518 MWt	Ginna 1300 MWt	Ginna 1520 MWt
HYDRAULIC AND THERMAL DESIGN PARAMETERS			
Total heat output, MWt	1518.5	1300	1520
Total heat output, Btu/hr	5181×10^6	4437×10^6	5188×10^6
Heat generated in fuel,%	97.4	97.4	97.4
Peak specific power, kW/ft	16	16.5	16.0
System pressure, nominal, psia	2250	2250	2250
System pressure, minimum steady-state, psia	2220	2220	2220
Hot-channel factors			
Heat flux, F_Q	2.80	3.38	2.80
Enthalpy rise, $F_{\Delta H}$	1.60	1.77	1.66
DNBR at nominal conditions	2.11	2.15	2.06
Minimum DNBR for design transients	1.30	1.30	1.30
Coolant flow			
Total flow rate, lb/hr	66.7×10^6	67.3×10^6	68.0×10^6
Effective flow rate for heat transfer, lb/hr	63.6×10^6	64.3×10^6	64.9×10^6
Effective flow area for heat transfer, ft^2	27.0	27.0	27.0
Average velocity along fuel rods, ft/sec	15.0	14.7	14.8
Average mass velocity, lb/hr-ft^2	2.37×10^6	2.38×10^6	2.41×10^6

	Point Beach Units 1 and 2 1518 MWt	Ginna 1300 MWt	Ginna 1520 MWt
Coolant temperature, °F			
Nominal inlet	552.5	551.9	544.5
Maximum inlet due to instrumentation, error, and deadband	556.5	555.9	548.5
Average rise in vessel	57.6	49.5	58.0
Average rise in core	60.0	52	60.5
Average in core	582.5	578.0	575.8
Average in vessel	581.3	577.0	573.5
Nominal outlet of hot channel	642.9	634.0	637.8
Average film coefficient, Btu/hr-ft ^{2}-°F	5600	5590	5690
Average film temperature difference, °F	31.0	26.9	30.9
Heat transfer at 100% power			
Active heat transfer surface area, ft ^{2}	28,715	28,715	28,715
Average heat flux, Btu/hr-ft ^{2}	175,800	150,500	176,700 (Region 4)[a] 176,000 (Region 3)[a]
Maximum heat flux, Btu/hr-ft ^{2}	491,000	508,700	494,800 (Region 4) 492,700 (Region 3)
Average thermal output, kW/ft	5.7	4.88	5.7
Maximum thermal output, kW/ft	16.0	16.5	16.0
Maximum clad surface temperature at nominal pressure, °F	657	657	657

	Point Beach Units 1 and 2 1518 MWt	Ginna 1300 MWt	Ginna 1520 MWt
Fuel central temperature, °F			
Maximum at 100% power	≈3750	3880	3900 (Region 4) 3850 (Region 3)
Maximum at overpower	≈4000	4100	4500 (Region 4) 4500 (Region 3)
Thermal output, kW/ft at maximum overpower	17.9	18.5	21.1
CORE MECHANICAL DESIGN PARAMETERS			
Fuel assemblies			
Design	RCC canless 14 x 14	RCC canless 14 x 14	RCC canless 14 x 14
Rod pitch, in.	0.556	0.556	0.556
Overall dimensions, in.	7.763 x7.763	7.763 x7.763	7.763 x 7.763
Fuel weight (as UO_2), lb	118,729	118,729	118,246[b]
Total weight, lb	154,519	150,750	150,267[b]
Number of grids per assembly	7	9	9
Fuel rods			
Number	21,659	21,659	21,659
Outside diameter, in.	0.422	0.422	0.422
Diametral gap, in.	0.0065	0.0065	0.0085 (Region 4) 0.0065 (Region 3)
Clad thickness, in.	0.0243	0.0243	0.0243
Clad material	Zircaloy	Zircaloy-4	Zircaloy-4

	Point Beach Units 1 and 2 1518 MWt	**Ginna 1300 MWt**	**Ginna 1520 MWt**
Fuel pellets			
Material	UO_2 Sintered	UO Sintered	UO Sintered
Density (% of theoretical)	Unit 1 94-92-91 Unit 2 94-93-92	92-90	92 (Region 4) 90 (Region 3)
Diameter, in	0.3699	0.3699	0.3649 (Region 4) 0.3669 (Region 3)
Length, in.	0.6000	0.6000	0.6000
Rod cluster control assemblies			
Neutron absorber	5% Cd, 15% In, 80% Ag	5% Cd, 15% In, 80% Ag	5% Cd, 15% In, 80% Ag
Cladding material	Type 304 SS-Cold Worked	Type 304 SS-Cold Worked	Type 304 SS-Cold Worked
Clad thickness, in	0.019	0.019	0.019
Number of clusters, full/part-length	37	29/4	29/4
Number of control rods per cluster	16	16	16
Core structure			
Core barrel I.D./O.D., in.	109.0/112.5	109.0/112.5	109.0/112.5
Thermal shield I.D./O.D., in.	115.3/122.5	115.3/122.5	115.3/112.5

	Point Beach Units 1 and 2 1518 MWt	Ginna 1300 MWt	Ginna 1520 MWt
NUCLEAR DESIGN DATA			
Structural characteristics			
Fuel weight (as UO_2), lb	118,729	118,727	118,727
Clad weight, lb	24,260	22,440	22,440
Core diameter, in. (equivalent)	96.5	96.5	96.5
Reflector thickness and composition	144	144	143.4 (Region 4) 144 (Region 3)
Top-water plus steel, in.	10	≈10	≈10
Bottom-water plus steel, in.	10	≈10	≈10
Side-water plus steel, in.	15	≈15	≈15
H_2O/U, unit cell (cold volume ratio)	3.35	3.35	3.35
Number of fuel assemblies	121	121	121
UO_2 rods per assembly	179	179	179
Performance characteristics			
Loading technique	3 region, nonuniform	3 region, nonuniform	3 region, nonuniform
Fuel discharge burnup, MWd/MTU			
Average first cycle	15,100	≈14,126	≈8,000
First core average	33,000	24,400	24,400
Feed enrichments, wt %			
Region 1	2.27	2.44	2.44
Region 2 (first core with burnable poison)	3.03	2.78	2.78
Region 3	3.04	3.48	3.48
Equilibrium	3.40	3.00	3.00

	Point Beach Units 1 and 2 1518 MWt	Ginna 1300 MWt	Ginna 1520 MWt
Control characteristics (beginning-of-life) Effective multiplication (with burnable poison)			
Cold, no power, clean	1.211	1.188	1.188
Hot, no power, clean (T_{mod} = 573 °F)	1.167	1.137	1.137
Hot, full power, xenon and Samarium equilibrium	1.113	1.080	1.080
Rod cluster control assemblies			
Material	5% Cd, 15% In, 80% Ag	5% Cd, 15% In, 80% Ag	5% Cd, 15% In, 80% Ag
Number of rod cluster control assemblies	37	33	33
Number of absorber rods per rod cluster control assembly	16	16	16
Total rod worth	7.1%	6.8%	6.8%
Boron concentrations (first cycle with burnable poison)			
To shut reactor down with no rods inserted, clean,(keff = .99) cold/ hot	1598 ppm/1676 ppm	1630 ppm/1580 ppm	1630 ppm/1580 ppm
To control at power with no rods inserted, clean equilibrium xenon and samarium	1465 ppm/1007 ppm	1470 ppm/1100 ppm	1470 ppm/1100 ppm
Boron worth, hot	1% Δk/k/130 ppm	1% Δk/k/120 ppm	1% Δk/k/120 ppm
Boron worth, cold	1% Δk/k/98 ppm	1% Δk/k/90 ppm	1% Δk/k/90 ppm
Kinetic characteristics			
Moderator temperature coefficient	+0.3 x 10^{-4} to -2.5 x 10^{4} Δk/k/°F	+.3 to -3.5 x 10^{-4} Δk/k/°F	+0.3 to -3.5 x 10^{-4} Δk/k/°F
Moderator pressure coefficient	-0.3 x 10^{-6} to 3.5 x 10^{-6} Δk/k/psi	-0.3 x 10^{-6} to +3.5 x 10^{-6} Δk/k/psi	-0.3 x 10^{-6} to +3.5 x 10^{-6} Δk/k/psi
Moderator void (density coefficient)	-0.10 to -0.30 Δk/k/g/cm^3	-0.10 to +0.30 Δk/k/g/cm^3	-0.10 to +0.30 Δk/k/g/cm^3
Doppler coefficient	-1.0 x 10^{-5} to -1.6 x 10^{-5} Δk/k/°F	-1.0 x 10^{-5} to -1.6 x 10^{-5} Δk/k/°F	-0.93 x 10^{-6} to -2.9 x 10^{-5} Δk/k/°F

	Point Beach Units 1 and 2 1518 MWt	Ginna 1300 MWt	Ginna 1520 MWt
REACTOR COOLANT SYSTEM - CODE REQUIREMENTS			
Component			
Reactor vessel	ASME III, Class A	ASME III, Class A	ASME III, Class A
Steam generator			
Tube side	ASME III, Class A	ASME III, Class A	ASME III, Class A
Shell side	ASME III, Class C[c]	ASME III, Class C	ASME III, Class C
Pressurizer	ASME III, Class A	ASME III, Class A	ASME III, Class A
Pressurizer relief tank	ASME III, Class C	ASME III, Class C	ASME III, Class C
Pressurizer safety valves	ASME III	ASME III	ASME III
Reactor coolant piping	USAS B31.1	USAS B31.1	USAS B31.1
PRINCIPAL DESIGN PARAMETERS OF THE REACTOR COOLANT SYSTEM			
Nuclear steam supply system heat output, MWt	1518.5	1300	1520
Core heat output, Btu/hr	5181×10^6	4437×10^6	5188×10^6
Operating pressure, psig	2235	2235	2235
Reactor inlet temperature, °F	552.5	551.9	551.9
Reactor outlet temperature, °F	610.1	601.4	602.4
Number of loops	2	2	2
Design pressure, psig	2485	2485	2485
Design temperature, °F	650	650	650
Hydrostatic test pressure (cold), psig	3110	3110	3110
Total reactor coolant system volume, ft^3 (hot)	6450	6245	6245

	Point Beach Units 1 and 2 1518 MWt	Ginna 1300 MWt	Ginna 1520 MWt
Total reactor flow, gpm	178,000	180,000	179,400
PRINCIPAL DESIGN PARAMETERS OF THE REACTOR VESSEL			
Material	SA 302 Grade B, low alloy steel, internally clad with Type 304 austenitic stainless steel	SA 302 Grade B, low alloy steel, internally clad with Type 304 austenitic stainless steel	SA 302 Grade B, low alloy steel, internally clad with Type 304 austenitic stainless steel
Design pressure, psig	2485	2485	2485
Design temperature, °F	650	650	650
Operating pressure, psig	2235	2235	2235
Inside diameter of shell, in.	132	132	132
Outside diameter across nozzles, in.	224 1/16	219 5/16	219 5/16
Overall height of vessel and enclosure head, ft-in.	39-0	39-1	39 1-5/16
Minimum clad thickness, in	5/32	5/32	5/32
PRINCIPAL DESIGN PARAMETERS OF THE STEAM GENERATORS			
Number of units	2	2	2
Type	Vertical, U-tube with integral moisture separator	Vertical, U-tube with integral moisture separator	Vertical, U-tube with integral moisture separator
Tube material	Inconel	Inconel	Inconel
Shell material	Carbon steel	Carbon steel	Carbon steel
Tube side design pressure, psig	2485	2485	2485
Tube side design temperature °F	650	650	650
Tube side design flow, lb/hr	33.35×10^6	33.63×10^6	33.63×10^6
Shell side design pressure, psig	1085	1085	1085
Shell side design temperature, °F	556	556	556

	Point Beach Units 1 and 2 1518 MWt	Ginna 1300 MWt	Ginna 1520 MWt
Operating pressure, tube side, nominal, psig	2235	2235	2235
Operating pressure, shell side, maximum, psig	1020	989	989
Maximum moisture at outlet at full load, %	1/4	1/4	1/4
Hydrostatic test pressure, tube side (cold), psig	3110	3110	3110

PRINCIPAL DESIGN PARAMETERS OF THE REACTOR COOLANT PUMPS

	Point Beach Units 1 and 2 1518 MWt	Ginna 1300 MWt	Ginna 1520 MWt
Number of units	2	2	2
Type	Vertical, single stage radial flow with bottom suction and horizontal discharge	Vertical, single stage radial flow with bottom suction and horizontal discharge	Vertical, single stage radial flow with bottom suction and horizontal discharge
Design pressure, psig	2485	2485	2485
Design temperature, °F	650	650	650
Operating pressure, nominal, psig	2235	2235	2235
Suction temperature, °F	551.5	551.9	551.9
Design capacity, gpm	89,000	90,000	90,000
Design head, ft	259	252	252
Hydrostatic test pressure (cold), psig	3110	3110	3110
Motor type	ac induction single speed air cooled	ac induction single speed air cooled	ac induction single speed air cooled
Motor rating	6000 hp	6000 hp	6000 hp
Material	Austenitic SS	Austenitic SS	Austenitic SS
Hot leg - I.D., in.	29	29	29
Cold leg - I.D., in.	27-1/2	27-1/2	27-1/2
Between pump and steam generator - I.D., in.	31	31	31
Design pressure	2485	2485	2485

a. Region 3 was of the non-pressurized rod design; Region 4 was of the pressurized rod design.

b. Assumes reload with pressurized rods.

c. The shell side of the steam generator conforms to the requirements for Class A vessels and is so stamped as permitted under the rules of Section III.

Table 1.3-2
COMPARISON OF DESIGN PARAMETERS WITH SAN ONOFRE AND CONNECTICUT YANKEE[a]

Steam and Power Conversion Design Parameters	San Onofre Final Report	Ginna 1520 MWt	Connecticut Yankee Final Report
Turbine generator			
Turbine type	Three element, tandem compound, four-flow exhaust	Three element, tandem compound, four-flow exhaust	Three element, tandem compound, four-flow exhaust
Turbine capacity, kW			
Maximum guaranteed	450,000	496,322	616,200
Maximum calculated	450,000	516,739	646,135
Turbine speed, rpm	1800	1800	1800
Generator rating, kVa	500,000	608,400	667,000
Condensers			
Type	Single pass, horizontal divided box, deaerating	Radial flow, semicylindrical water boxes, deaerating	Single pass, divided water box, deaerating
Number	2	2	2
Condensing capacity, lb of steam/hr	3,293,000	3,448,805	---
Condensate pumps			
Type	Vertical, wet pit	Multi-stage, vertical pit-type centrifugal	Seven-stage vertical, pit-type
Number	4	3	2
Design capacity each, (gpm)	2900	6600	6200
Motor type	Vertical, induction	Vertical	Vertical, induction
Motor rating, hp	700	1500	1500
Feedwater pumps			

Steam and Power Conversion Design Parameters	San Onofre Final Report	Ginna 1520 MWt	Connecticut Yankee Final Report
Type	Two-stage, horizontal split case, double volute, centrifugal	High-speed, barrel-type, single stage, double-flow, centrifugal	Two-stage, horizontal centrifugal
Number	2	2	2
Design capacity (each), gpm	7000 (10,500 during safety injection)	7400	9600
Motor type	Horizontal, induction	Horizontal	Horizontal, induction
Motor rating, hp	3500	5000	4500
Emergency feedwater			
Source	240,000 gal condensate storage tank	30,000 gal in each of the two condensate storage tanks (CST); Service Water	100,000 gal demineralized storage tank
Emergency feedwater pumps			
Number	2 (1 steam-driven and 1 motor driven)	3 (1 steam-driven and 2 motor driven)	1
Design capacity, gpm	300 (steam-driven), 235 (motor-driven)	400 (steam-driven), 200 (motor-driven)	450

a. The data in this table are not current.

Table 1.3-3
COMPARISON OF GINNA AND KEWAUNEE UPRATE NSSS DESIGN PARAMETERS

Parameter	GINNA	KEWAUNEE
Total Core Power	1775 MWt	1772 MWt
System Pressure	2250 psia	2250 psia
Minimum Reactor Flow	85,200 gpm/loop	89,000 gpm/loop
Coolant Volume with Pressurizer	6084 ft^3	6435 ft^3
Pressurizer Volume	800 ft^3	1000 ft^3
Maximum Inlet Temperature	540.2°F	539.2°F
Maximum Average Temperature	576.0°F	573.0°F
Maximum Outlet Temperature	611.8°F	606.8°F

1.4 IDENTIFICATION OF AGENTS AND CONTRACTORS

The Rochester Gas and Electric Corporation (RG&E), as owner, engaged or approved the engagement of the contractors and consultants identified below in connection with the design and construction of the R. E. Ginna Nuclear Power Plant. However, regardless of the explanation of contractual arrangements offered below, Rochester Gas and Electric Corporation was the sole applicant for the construction permit and operating license and as owner and applicant was responsible for the design, construction, and operation of the plant.

The R. E. Ginna Nuclear Power Plant was designed and built by the Westinghouse Electric Corporation as prime contractor for RG&E. The project was directed by Westinghouse from the offices of its Atomic Power Division in Pittsburgh, Pennsylvania, and by Westinghouse representatives at the plant site during construction and plant startup. Westinghouse engaged the engineering firm of Gilbert Associates, Inc., of Reading, Pennsylvania, to provide the design of the structures and non-nuclear portions of the plant and to prepare specifications for the purchase and construction thereof. Rochester Gas and Electric Corporation reviewed the designs and specifications prepared by Westinghouse and Gilbert Associates to ensure that the general plant arrangements, equipment, and operating provisions were satisfactory to them. Rochester Gas and Electric Corporation inspected the construction work to ensure that the plant was built in accordance with the approved plans and specifications.

The plant was constructed under the general direction of Westinghouse through a general contractor, Bechtel Corporation, who was responsible for the management of all site construction activities and who either performed the work or subcontracted the work of construction and equipment erection. Preoperational testing of equipment and systems and initial plant operation was performed by RG&E personnel under the technical direction of Westinghouse.

Rochester Gas and Electric Corporation engaged the firm of Dames and Moore of New York, New York, as consultants on studies of plant site geology, hydrology, and seismology.

Rochester Gas and Electric Corporation engaged Dr. George Sutton of La Mont Geological Observatory, Palisades, New York, as an additional consultant on seismology.

Rochester Gas and Electric Corporation engaged the firm of Pickard, Lowe, and Associates, Washington, D.C., as consultants on reactor and plant engineering, site meteorology, and general site studies. In addition, specialists in environmental sciences participated in developing information concerning the site. These included: Dr. Ben Davidson, meteorologist and Director, Geophysical Science Laboratory, New York University College of Engineering; Drs. Donald Pritchard and James Carpenter, hydrologists, and Professor and assistant Professor, respectively, Department of Oceanography, Johns Hopkins University; Dr. G. Hoyt Whipple, health physicist, Professor of Radiological Health, School of Public Health, University of Michigan; and Dr. Robert Sutton, geologist, University of Rochester.

Westinghouse engaged the firm of Praeger-Kavanagh-Waterbury of New York, New York, as consultants on the structural design of the containment and other important structures.

The firm of Hansen, Holby, and Biggs, Massachusetts Institute of Technology, was engaged for structural engineering analyses. The Southwest Research Institute, San Antonio, Texas, was engaged as a consultant for quality control and for the establishment of an operating surveillance program.

Contractual support available during operations is discussed in Section 13.1.1.3.5.

1.5 REQUIREMENTS FOR FURTHER TECHNICAL INFORMATION

This section is provided for historical purposes and has not been updated. It includes a discussion of research and development completed and the requirement for further research and development perceived to be necessary at the time of submission of the original FSAR.

1.5.1 INTRODUCTION

Research and development to the level necessary to ensure safe operation of the R. E. Ginna Nuclear Power Plant was conducted in the following areas:

1. Development of the final core design and final thermal, hydraulic, and physics parameters.
2. Core stability including adequacy of out-of-core instrumentation.
3. Development of long ion chambers.
4. Control rod ejection accident analyses.
5. Charcoal filters for the removal of organic forms of iodine from the containment atmosphere following an accident.
6. Reactor coolant pump controlled leakage seal testing.
7. Safety injection system both design and analytical methods.
8. Development of design, inspection, and acceptance criteria for prestressed reinforced concrete pressure vessels.
9. Development of containment hydrogen recombiner.

The term "research and development" as used in this section is the same as that used by the NRC in Section 50.2 of its regulations as follows:

(n) "Research and development" means (1) theoretical analysis, exploration or experimentation; or (2) the extension of investigative findings and theories of a scientific nature into practical application for experimental and demonstration purposes including the experimental production and testing of models, devices, equipment, materials, and processes.

The research and development done for the R. E. Ginna Nuclear Power Plant confirms the engineering and design values used to complete the equipment and systems designs. It did not, in general, involve the creation of new concepts or ideas.

The technical information generated demonstrates the safety of the design and more sharply defines margins of conservatism.

1.5.2 DEVELOPMENT OF THE FINAL CORE DESIGN AND FINAL THERMAL-HYDRAULIC AND PHYSICS PARAMETERS

The detailed final core design and thermal-hydraulics and physics parameters have been completed. The nuclear design, including fuel configuration and enrichments, control rod pattern and worths, reactivity coefficients, and boron requirements are described in the original FSAR. The final thermal-hydraulics design parameters, as well as the final fuel, fuel rod, fuel assembly, and control rod mechanical design are also discussed in detail in the original FSAR. The core design incorporates fixed burnable poison rods (*Reference 1* and *2*) in the initial loading to ensure a negative moderator temperature coefficient of reactivity at operating temperature. This improves reactor stability and lessens the consequences of a rod ejection or a loss-of-coolant accident.

1.5.3 CORE STABILITY

1.5.3.1 Core Power Distribution

In the transition to 12 ft. long, zircaloy-clad fuel cores, a potential for core power distribution oscillations due to spatial oscillation in xenon concentration was created. Analytical methods have been developed to examine this problem, and their use has resulted in the development of suitable control hardware and a control strategy.

Nuclear calculation codes have been modified to simulate these power oscillations and the operator actions necessary to damp out these oscillations. The effect of power redistribution in the core on total power capability has been calculated and the control system is designed to automatically cut back turbine power, and therefore core power, if limits on power distribution are exceeded. The protection system is designed to automatically reset thermal trip levels if these limits on power distribution are exceeded.

The core of the R. E. Ginna Nuclear Power Plant contains burnable poison rods, which eliminate the positive moderator coefficient that was expected at operating temperatures early in the first fuel cycle in the original core design. The burnable poison rods will be borosilicate glass. Critical experiments have been conducted at the Westinghouse Reactor Evaluation Center using rods containing 12.8 wt % boron and zircaloy-clad uranium dioxide fuel rods, 2.27% enriched. These values are typical of this plant also. These experiments showed that standard analytical methods can be used to calculate the reactivity worth of the burnable poison rods. The design basis and critical experiments are described in *References 1* and *2*. (Note: burnable poison rods are no longer included in the core.)

In-core testing completed in the Saxton reactor has shown satisfactory performance. The tests are continuing and the research and development effort on these burnable poison rods is described in more detail in the R&D topical report presented at the Salem Public Hearing, August 15, 1968.

1.5.3.2 Out-of-Core Ion Chambers

The control system input from the nuclear instrumentation is the signals from four 10-ft long, two-section ion chambers (described in Section 1.5.4), mounted outside the reactor vessel. Calculations have shown that the response of these ion chambers should accurately indicate gross power redistribution in the core, both axial and transverse, and this has been confirmed by experimental measurements made on the SENA, San Onofre, and Haddam Neck reactors. Tests performed to date include forcing various axial and transverse power shapes with full-length control rods, and comparing the measured out-of-core readings with detailed in-core measurements. Excellent correlation has been obtained. The calculations and results are detailed and discussed in *Reference 3*.

1.5.3.3 In-Core Control Equipment

Calculations performed for this plant demonstrate that power oscillations across the core will be inherently highly damped and no control applied damping is either provided or necessary. In any case, there is no mode of normal operation (MODES 1 and 2) which could cause a transverse power tilt or, if one occurred, would make it worse.

There is, in a zircaloy core of this length, the possibility that xenon-induced axial power distribution oscillations may occur. Detailed calculations have shown that these oscillations can be simply and effectively controlled, and suitable equipment has been developed for this plant.

The in-core control equipment consists of four part-length rods, symmetrically placed about the core axial center line, and moved in unison. Each rod has absorber in the bottom quarter only, and is raised and lowered by a mechanism that holds the rod in a fixed position following a reactor trip or loss of power to the mechanism. Since the xenon oscillation period is about 1 day, the part-length rods are under operator control. The control strategy is based on maintaining the difference in output between the top and bottom sections of the long ion chambers within a specified range. If the operator allows axial power imbalance to exceed operating limits, automatic protection occurs (*Reference 3*). The operating band is well inside core thermal limits.

The part-length control rods permit axial power shaping as well as axial power oscillation control. (Note: The part-length rods have been removed from the in-core control equipment.)

The hardware, out-of-core instrumentation adequacy, control strategy, and rod insertion limits are described in *Reference 3*. The performance of the system will be verified and the calculated performance checked during the thorough startup test program, which is described below and in Chapter 14.

1.5.3.4 Startup Test Program

Experimental verification that spatial power redistribution transients can be monitored and controlled is to be obtained in four consecutive stages of power testing in the overall plant startup program. These states of power testing are described in the following.

A. Steady-state calibration of power range instrumentation in which the out-of-core power range nuclear channels (using the long ion chambers), in-core core exit thermocouples, and primary loop resistance temperature detectors are calibrated on the bases of measured secondary heat balances and detailed in-core power distributions measured with the movable detector system. These instrumentation intercalibrations are repeated at several power levels of interest between 30% and 100% of full power in typical operating control rod configurations. The results of these steady-state measurements are analyzed and correlations developed between out-of-core detector response and in-core detector measurements of power peaking. Design operational curves are verified or appropriate adjustments made to ensure that design limits on power peaking are not exceeded. Instrumentation accuracies are evaluated in these tests.

B. Follow of spatial power redistribution transients in which spatial transients are initiated at a reduced constant power level by prescribed control rod maneuvers and the resultant changes in core power distribution are monitored in terms of axial and azimuthal power offsets (*Reference 3*) as indicated by the out-of-core power range nuclear detectors and of assembly-wise power sharing factors and gross power tilts as indicated by the in-core thermocouple system. Concurrent periodic measurements of the core power distribution made with the in-core movable detectors allow verification of the inter-calibrations of the out-of-core power range instrumentation under transient conditions and direct evaluation of nuclear hot-channel factors. Transient reactivity changes are met by adjustment of the reactor coolant boron concentration.

C. Controlled follow of spatial power redistribution transients in which spatial transients are initiated, as before, by control rod maneuvering at constant power and the resultant power peaking transients are suppressed by subsequent maneuvering of the part-length control rods by the operator. The maneuvering scheme for limiting local power peaking during the induced transients is to be the normal procedure prescribed for plant operation where successive control rod maneuvers are dictated by the current values of axial offset ratios derived from the out-of-core power range nuclear detector responses (for example see *Reference 3*). Concurrent periodic power distribution measurements made with the in-core movable detector system allow verification both of the values of limiting power distribution parameters as deduced from the out-of-core instrumentation responses and of the adequacy of the prescribed operating procedure for limiting power peaking during spatial power distribution transients.

D. Controlled follow of dynamic power redistribution transients in which the operation of the plant reproduces a typical load variation cycle, but at a reduced power level. Spatial power redistribution transients resulting from the associated power level changes and the attendant control rod maneuvers are monitored with the out-of-core nuclear detectors and core exit thermocouples and power peaking is by part-length control rod manipulation according to standard operating procedures. Concurrent detailed core power distribution measurements with the movable detector system are made to evaluate nuclear hot-channel factors and verify correlations with out-of-core instrumentation.

The results of the several stages of measurement and verification are reviewed for adequacy, before the next stage of testing is undertaken.

As burnup of the core progresses, test 1 will be repeated at regular intervals under typical operating conditions in accord with normal operating practice. At less frequent intervals test 2 and test 4 during a normal load variation cycle, including in both cases comprehensive detailed power distribution measurements made with the moveable detector system, will be repeated to allow assessment of the effects of core depletion.

1.5.4 DEVELOPMENT OF LONG ION CHAMBERS

This plant uses four long ion chambers, mounted vertically outside the reactor pressure vessel for power range nuclear instrumentation. The chambers are 90 degrees apart in plan; each chamber has an active length of 10 ft with its center level with the core horizontal midplane.

Each chamber is split into an upper and lower section to effectively form two uncompensated ion chambers of equal size.

One purpose of these long ion chambers in this plant is to detect axial power redistributions when they occur, and any transverse power tilts that could arise if control rods become malpositioned. The efficiency of these out-of-core long ion chambers in accurately reflecting in-core power distribution is shown in *Reference 3*. Also, their long total active length minimizes differences in indicated core average power for the same actual power but different control rod positions.

This is the first U.S. plant to use uncompensated long ion chambers as standard instrumentation, but the design is similar in both size and configuration to chambers that have now been successfully tested over extended periods in similar reactor service. Four two-section (one section compensated, the other uncompensated) 8 ft. long ion chambers have been used on the SENA reactor as their standard instrumentation for about four months. An 8 ft. long two section ion chamber, similar to the Ginna design, was tested on the Carolinas Virginia Tube Reactor for about 12 months. This chamber was then transferred to the San Onofre reactor where it has had about 15 months operation. In addition, a long ion chamber, identical to those to be fitted on Ginna, was installed for testing at San Onofre in September 1968.

From this design, manufacturing, and test experience of long ion chambers, it is expected that the long ion chambers for this plant will perform satisfactorily.

1.5.5 CONTROL ROD EJECTION AND DROPPED CONTROL ROD ACCIDENT ANALYSES

The ejection of a control rod from the core would require the failure of its control rod mechanism housing. Although such a failure is not considered credible, single control rod ejection analyses using the final core design parameters, including abnormal conditions that could occur during plant operation and tolerances for instrumentation error and reactivity coefficient, have been completed. The four cases analyzed are zero and full power; beginning and end of core life. These show that no consequential damage to the reactor coolant system will occur under these adverse conditions.

This plant core was initially designed to use only movable absorber rods and chemical shim to control reactivity, but will now, in addition, have burnable poison rods installed. The consequences of a rod ejection accident are inherently limited in a core with chemical shim control since the amount of rod insertion is limited to that necessary to change load, while the chemical shim concentration is adjusted to compensate for fuel burnup. The addition of the burnable poison rods now also ensures that the moderator coefficient of reactivity is negative throughout core life at operating temperature, further reducing the consequences of an ejection accident. The research and development program on the burnable poison rods is discussed in Section 1.5.3.

The consequences of dropping single full-length control rods have been analyzed. Either the actual rod drop or its resultant effects on local power and flux distribution will be detected, and action to protect the core and coolant system against damage is automatic. This protection includes blocking control rod withdrawal.

1.5.6 CHARCOAL FILTERS

At the time the plant was proposed, it appeared that further development work would be required to prove the effectiveness of impregnated activated charcoal filters in removing radioactive iodine in both organic (methyl iodine) and inorganic (elemental) forms.

Tests on the extraction of methyl iodide by full-size charcoal filters were made in cooperation with the Connecticut Yankee Atomic Power Company for their Haddam Neck plant. These demonstrated the suitability of using iodized activated charcoal filters to remove radioactive methyl iodide from a containment environment under the most extreme conditions anticipated following a loss-of-coolant accident. The results of these tests (*Reference 4*) filed with the AEC under Docket No. 50-213 are applicable to the charcoal filter system employed in this plant.

Before any testing was started on the extraction of elemental iodine by the charcoal filters, a literature survey was made. This showed that sufficient experimental data was already available from other sources (*References 5 through 8*) to confirm that activated charcoal filters were even more efficient in extracting elemental iodine than methyl iodide under any typical post loss-of-coolant accident environmental conditions. It was therefore decided that tests for elemental iodine extraction were no longer necessary, and no further experiments were conducted. This conclusion that further research and development on elemental iodine extraction by charcoal filters was unnecessary was also expressed by the AEC staff at the Public Hearing in the matter of the Salem Nuclear Plant for the Public Service Electric and Gas Company, August 15, 1968, Docket Numbers 50-272 and 50-311.

The effectiveness of the charcoal filter units during plant use will be demonstrated by periodic tests at Haddam Neck and in this plant, as required by the Technical Specifications. These tests will determine if there is any need for filter replacement because of deterioration with time.

1.5.7 REACTOR COOLANT PUMP CONTROLLED LEAKAGE SEALS

The reactor coolant pump controlled leakage seal design for this plant has been fully developed. A full scale mock-up of this seal was operated for over 100 hr to confirm that seal deflection under load and leak rate are acceptable. These tests also showed that erosion and corrosion of the seal materials were not adversely affected by the slight increase in water velocity through the seal due to the increased seal size necessary to fit the larger shafts used in these pumps. A full-scale mock-up was used during the development of the controlled leakage seal to provide information on long-term performance and this life testing will continue.

One of the seals used in this plant was operated about 300 hr and the other about 100 hr, each in its pump motor unit. During hot functional testing in the plant, before the core is loaded, additional operation will bring the total operating time for each seal to well over 500 hours.

Successful operation of similar seals has been demonstrated with over 5000 hours total running time in San Onofre and over 3000 hours in Haddam Neck. More than 10 pumps have already been built for later plants and tested successfully for at least 100 hours each. The seals in these latter pumps are the same size as those used in this plant.

1.5.8 SAFETY INJECTION SYSTEM

1.5.8.l Development of Safety Injection System Design

The development effort on the Emergency Core Cooling System (ECCS) design has resulted in the modification of the system to include nitrogen pressurized accumulator tanks for rapid core reflooding with borated water. The accumulators are passive devices, and the only valves between them and their injection nozzles are swing check valves which open entirely automatically once the reactor coolant system pressure falls. The increased flooding capability limits the clad temperature after a loss-of-coolant accident to well below the melting temperature of Zircaloy-4, minimizes metal-water reaction, and ensures that the core remains in place and intact, thereby ensuring preservation of essential heat transfer geometry. The system design incorporates redundancy of components such that the minimum required water addition rates can be met assuming any active component to fail concurrent with the loss-of-coolant accident or, over the long-term period of post-accident core decay heat removal, a passive or active component failure in either the safety injection or service water systems, or an active failure in the component cooling water (CCW) system.

1.5.8.2 Development of Core Cooling Analysis

The loss-of-coolant analysis presented in the PSAR was based on a one-element code (LOCO) for the blowdown and reflooding portions of the transients. For the FSAR a more detailed blowdown code (FLASH) was used. The FLASH code divides the reactor coolant system into three regions. This division provides for a more precise description of the blowdown process, and in particular for the input to the reactor kinetics and core cooling analysis.

The FLASH code has been compared to many blowdown experiments primarily those performed at LOFT. It has been demonstrated that the code is conservative in two principal areas: rate of depressurization and mass of water left after blowdown. The FLASH code was required to analyze the performance of the improved Emergency Core Cooling System (ECCS) for large area ruptures.

The LOCTA-R2 transient digital computer program was developed during the final design of the Ginna reactor for evaluating fuel pellet and cladding temperatures during a loss-of-coolant accident.

The code is able to stack axial sections and thereby describe the behavior of a full-length region as a function of time. A mass and energy balance is used in evaluating the temperature rise in the steam as it flows through the core.

The present code is a more sophisticated version of LOCTA-R which was used in the loss-of-coolant accident analyses reported in the PSAR. LOCTA-R was able to describe the behavior of only one axial location on the rod while holding the environmental sink temperature constant throughout the accident.

The SLAP code has replaced LOCO for predicting the entire blowdown and reflooding characteristics of the smaller ruptures. The SLAP code is essentially an extension of the LOCO code, but it provides a better description of the transient on the steam generator shell side and the heat transferred between the reactor and steam generator during blowdown.

For the smaller breaks it is important to determine if departure from nucleate boiling occurs during blowdown. The SATAN-R and THINC codes were used for this purpose. Core parameters obtained from SATAN-R, such as pressure, power, and flow, were used as input to the THINC code. The THINC code is used to calculate coolant density, mass velocity, enthalpy, vapor voids, and static pressure distribution along parallel flow channels in the core.

Extensive work on the development of these new models was completed during the final design of the Ginna reactor.

1.5.9 DEVELOPMENT OF DESIGN, INSPECTION, AND ACCEPTANCE CRITERIA FOR PRESTRESSED REINFORCED-CONCRETE PRESSURE VESSELS

At the time Ginna Station was proposed, the unusual feature of the steel-lined reinforced-concrete reactor containment vessel was the use of post-tensioned prestressing tendons, although their use in construction is well proven. The developments and tests discussed below are therefore confined to those elements directly applicable to the prestressing of the containment vessel. These are:

- Rock anchor design criteria and test results.
- Rock anchor grout.
- Tendon inspection and acceptance criteria.
- Tendon corrosion protection system.

These topics are discussed in more detail below.

1.5.9.1 Rock Anchors

1.5.9.1.1 Design Criteria and Assumptions

The basic criterion in determining the length of rock anchors necessary to develop adequate hold-down capacity, is that the pull of the anchor is resisted only by the submerged weight of rock. The assumptions are made that (1) the rock has no tensile strength, (2) it breaks out at an angle of 45 degrees to the vertical, with the depth taken to the midpoint of the bond development length, and (3) the bond-stress between rock and grout is 170 psi.

1.5.9.1.2 Test Verification and Results

These assumptions and their historical justification are discussed in Section 3.8.1.4.2. In order to determine the factors of safety represented by these assumptions for the conditions pertaining to this plant site, a series of tests were carried out on three scaled-down test anchors, to demonstrate rock hold-down capacity and bond strength between grout and rock.

These tests and results are described in Section 3.8.1.7.

1.5.9.2 Rock Anchor Grout

Grouting techniques used followed closely those developed by the Swiss parent company of the BBRV system. The grout used is a mix of 5 gallons of water to one bag of cement, with 1 lb of a special BBRV additive. The latter, designed to reduce the water requirements of the cement (and so retard the setting time), also provides a controlled expansion of the grout of about 8%, accomplished by the reaction of an aluminum powder with the alkalies of the cement. The additive is free from chlorides, sulfides, and other salts whose presence could possibly create a corrosion problem. The cement used is non-air entraining, Type II.

A test was carried out at the site to verify the grout application procedure and to ensure cohesion and hardening of the grout, even when pumped under water.

1.5.9.3 Tendon Inspection and Acceptance Criteria

Buttonhead dimensional accuracy and symmetry are important to ensure maximum development of both the rock anchor and wall tendon strength. Consistency of length of tendon wires is necessary to ensure uniform load distribution on individual wire elements. Uniformity of material properties is important in obtaining correct tendon characteristics compatible with those assumed for analysis, i.e., ductility and ultimate and yield strengths.

The acceptance criteria and the program to ensure conformity with these were developed after inspection of the fabricator’s initial production runs and are outlined in Section 3.8.1.6.7.

1.5.9.4 Wall Tendons

1.5.9.4.1 Corrosion Protection

The use of unbonded tendons gives, in addition to other advantages, accessibility for inspection or replacement. However, because the tendons are not in intimate and integral contact with surrounding concrete, the advantage of the high alkaline environment generally considered to promote adequate corrosion protection is lost. Therefore, these tendons must be provided with a corrosion preventive medium that gives protection equivalent to concrete, but still enables withdrawal of a tendon for inspection or replacement.

Consequently, one of the more important programs in connection with the tendons has been the selection of a complete corrosion protection system. The various elements involved are (1) a cathodic protection system in which all tendons are connected to the liner and then to a copper grounding system which is completed by the addition of reference cells and anodes, from which a protective potential can be generated if the need for cathodic protection is indicated by the reference cells, (2) a steel conduit surrounding each tendon providing shielding against stray electrical currents, (3) temporary shipping and erection protection of all wires in each tendon, by the application of a coating, followed by complete filling of each tendon conduit with a petroleum base wax, NO-OX-ID "CM," that provides a permanent, chemically stable environment for protection from corrosion, while still giving flexibility of withdrawal for inspection. The selection, testing, and application of the coating and wax was an important program in the development of the overall corrosion protection system. Tests at the W. R. Grace & Company Dearborn Division Research Center are outlined below.

Two tendon mock-up test rigs were set up for evaluation of individual wire coverage by the wax and for determination of pumping characteristics. One test rig consisted of a transparent pyrex glass tube test section containing a tendon section through which the wax could be circulated. Tests showed that as the wax moved through the test section it completely immersed all the wires, even though some were tightly bunched together. Subsequent inspection of individual wires showed complete coverage. In a second test, a quantity of water was introduced into the test section and pumping started. The water "plug" was driven ahead of the wax, which preferentially wetted all the wires. There appeared to be no diffusion or mixing of the water into the wax.

A second test rig consisted of a 20 ft. high conduit section containing a short-length tendon, complete with all anchor heads and hardware. This was used to determine pump pressures for circulation under ambient conditions, flow rates, and friction losses.

Specimens coated with both the initial coating and the wax were compared to uncoated control plates under extreme conditions of continuous exposure to salt water, steam, relative humidity, and temperature in environmental testing cabinets. Results obtained after many hundreds of hours showed no deterioration of the coated specimens.

1.5.9.4.2 Inspection and Acceptance

Preoperational testing on the complete containment is discussed in Section 3.8.1.7.1.

1.5.10 DEVELOPMENT OF CONTAINMENT HYDROGEN RECOMBINER

Following a major loss-of-coolant accident in the Ginna Station reactor, hydrogen may be generated inside the containment by the mechanisms of radiolysis, zirconium-water reaction, and the reaction of alkaline spray solution with aluminum. Because of the high level of radioactivity in the containment which may also result from the accident, the containment must be sealed for an extended period to prevent the spread of contamination to the environment.

Under these circumstances, if the containment isolation is sufficiently long, the possibility of hydrogen reaching a flammable concentration of 4.1 volume percent in air must be considered. Equipment was therefore provided for the controlled recombination of hydrogen at a concentration. The system selected is a flame combustor using containment atmosphere (containing a low concentration of hydrogen) as primary oxidant and supplemental hydrogen as a fuel. The product of combustion, water vapor, is cooled and condensed from the atmosphere by the vital cooling systems of the containment. Operation of the system will control buildup of hydrogen to less than 2 volume percent or one-half of the lower flammable limit.

Inside the containment are two complete combustor systems, one a spare. Each system consists of a blower to circulate containment air to the combustor, a combustion chamber complete with main burner, two igniters (one a spare), pilot burner, and a dilution chamber downstream of the flame zone where products of combustion are mixed with a large excess of containment air to reduce the temperature of gas leaving the system.

Testing of a recombiner system will be used to:

- Demonstrate that the design is sound (proof testing).
- Determine certain limits for the combustor in performance.

A description of the recombiner and the research, development, and test program is discussed in more detail in *Reference 9*.

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1.7 DRAWINGS AND OTHER DETAILED INFORMATION

1.7.1 ELECTRICAL, INSTRUMENTATION, AND CONTROL DRAWINGS

Updated electrical drawings, schematics, logic diagrams, and elementary wiring diagrams were submitted to the NRC during the Systematic Evaluation Program (SEP) as necessary to permit the staff to review the safety-related aspects of Ginna Station.

Drawings representing the electrical, instrumentation, and control systems are referenced throughout the UFSAR.

A list of electrical, instrumentation, and control drawings, which previously were included as figures in earlier revisions of the UFSAR, is given in Table 1.7-1.

1.7.2 PIPING AND INSTRUMENTATION DIAGRAMS (P&ID)

Updated piping and instrumentation diagrams were submitted to the NRC during the SEP as necessary to permit the staff to review the safety-related aspects of Ginna Station.

Drawings representing the piping and instrumentation diagrams are referenced throughout the UFSAR. A list of piping and instrumentation diagrams, which previously were included as figures in earlier revisions of the UFSAR, is given in Table 1.7-2. The legend for symbols used in these diagrams is included in Drawing 33013-2242, Sheets 1-4.

1.7.3 OTHER DETAILED INFORMATION

References to detailed information submitted to the NRC are incorporated in the appropriate sections throughout the UFSAR and are not duplicated in this section.

Table 1.7-1
ELECTRICAL, INSTRUMENTATION, AND CONTROL DRAWINGS

Drawing Number	Title	Historical Link to UFSAR Figure Number
03201-0102	120-Volt AC Instrument Bus One-Line Diagram	8.3-4
03202-0102	One-Line Diagram, 125-Volt DC System	8.3-6
33013-623		
Sheet 1	Main One-Line Diagram	8.3-1, Sheet 1
Sheet 2	Main One-Line Diagram	8.3-1, Sheet 2
33013-652	480-Volt One-Line Diagram	8.3-3
33013-653	4160-Volt One-Line Diagram	8.3-2
33013-1353		
Sheet 1	Logic Diagram, Index and Symbols	7.2-3
Sheet 2	Logic Diagram, Reactor Trip Signals	7.2-4
Sheet 3	Logic Diagram, Turbine Trip Signals	7.2-9
Sheet 4	Logic Diagram, Electrical Protection Logic	7.2-8
Sheet 5	Logic Diagram, Emergency Diesel Generator Startup Logic	8.3-5
Sheet 6	Logic Diagram, Safeguards Actuation Signals	7.3-1, Sheet 1
Sheet 7	Logic Diagram, Safeguards Actuation Signals	7.3-1, Sheet 2
Sheet 8	Logic Diagram, Safeguards Sequence	7.3-3
Sheet 9	Logic Diagram, Feedwater Isolation and Auxiliary Feedwater Pump Actuation Signals	7.3-2
Sheet 10	Logic Diagram, Nuclear Instrumentation Trip Signals	7.2-6
Sheet 11	Logic Diagram, Nuclear Instrumentation Permissives, and Blocks	7.2-11
Sheet 12	Logic Diagram, Pressurizer Trip Signals	7.2-7
Sheet 13	Logic Diagram, Steam Generator Trip Signals	7.2-10
Sheet 14	Logic Diagram, Reactor Coolant System Trip Signals	7.2-5
Sheet 15	Logic Diagram, Rod Stops and Turbine Runbacks	7.7-5

Table 1.7-2
PIPING AND INSTRUMENTATION DIAGRAMS (P &ID)

Drawing Number	Title	Historical Link to UFSAR Figure Number
33013-1231	Main Steam System (Safety Related) - P&ID	10.3-1
33013-1232	Main Steam System (Non Safety Related) - P&ID	10.3-2
33013-1233	Condensate Low Pressure Feedwater Heaters - P&ID	10.4-3
33013-1234	Condensate Storage System - P&ID	10.7-5
33013-1235	Condensate System (Condensate Booster Pumps to Hydrogen Coolers and Blowdown Recovery System) - P&ID	10.4-2
33013-1236		
Sheet 1	Feedwater System - P&ID	10.4-4, Sheet 1
Sheet 2	Feedwater System - P&ID	10.4-4, Sheet 2
33013-1237	Auxiliary Feedwater System - P&ID	10.5-1
33013-1238	Standby Auxiliary Feedwater System - P&ID	10.5-2
33013-1239		
Sheet 1	Diesel Generator "A" Supporting Systems - P&ID	9.5-5, Sheet 1
Sheet 2	Diesel Generator "B" Supporting Systems - P&ID	9.5-5, Sheet 2
33013-1242	Fire Protection System - Relay and Computer (MUX) Rooms - P&ID	9.5-3
33013-1245	Component Cooling Water System - P&ID	9.2-4, Sheet 1
33013-1246		
Sheet 1	Component Cooling Water System - P&ID	9.2-4, Sheet 2
Sheet 2	Component Cooling Water System - P&ID	9.2-4, Sheet 3
33013-1247	Residual Heat Removal System - P&ID	5.4-7
33013-1248	Spent Fuel Pool Cooling System - P&ID	9.1-6
33013-1250		
Sheet 1	Service Water System, Safety Related - P&ID	9.2-1, Sheet 1
Sheet 2	Service Water System, Safety Related - P&ID	9.2-1, Sheet 2
Sheet 3	Service Water System, Safety Related - P&ID	9.2-1, Sheet 3

Drawing Number	Title	Historical Link to UFSAR Figure Number
33013-1251		
Sheet 1	Service Water System, Non Safety Related - P&ID	9.2-2, Sheet 1
Sheet 2	Service Water System, Non Safety Related - P&ID	9.2-2, Sheet 2
33013-1252	Condensate System - P&ID	10.4-1
33013-1256	Technical Support Center HVAC System - P&ID	9.4-17
33013-1258	Reactor Coolant Pressurizer - P&ID	5.1-1, Sheet 2
33013-1259	Miscellaneous Liquid Waste Disposal - P&ID	11.2-1
33013-1260	Reactor Coolant - P&ID	5.1-1, Sheet 1
33013-1261	Containment Spray - P&ID	6.2-11
33013-1262		
Sheet 1	Safety Injection and Accumulators - P&ID	6.3-1, Sheet 1
Sheet 2	Safety Injection and Accumulators - P&ID	6.3-1, Sheet 2
33013-1263	Reactor Coolant System Overpressure Protection, Nitrogen Accumulator System - P&ID	5.2-1
33013-1264	Chemical and Volume Control, Letdown - P&ID	9.3-14
33013-1265		
Sheet 1	Chemical and Volume Control, Charging - P&ID	9.3-13, Sheet 1
Sheet 2	Chemical and Volume Control, Charging - P&ID	9.3-13, Sheet 2
33013-1266	Chemical and Volume Control, Boric Acid - P&ID	9.3-15
33013-1267	Chemical and Volume Control, Holdup Tanks to Gas Strippers - P&ID	9.3-18
33013-1268	Chemical and Volume Control, Boric Acid Evaporator to Monitor Tanks - P&ID	9.3-17
33013-1269	Chemical and Volume Control, Reactor Makeup Water System - P&ID	9.3-16
33013-1270		
Sheet 1	Waste Disposal - Liquid, Waste Drains, Holdup Tank, Spent Resin Tanks - P&ID	11.2-3, Sheet 1
Sheet 2	Waste Disposal - Liquid, Waste Drains, Holdup Tank, Spent Resin Tanks - P&ID	11.2-3, Sheet 2
33013-1271	Waste Disposal - Liquid, Waste Condensate Tanks - P&ID	11.2-4

Drawing Number	Title	Historical Link to UFSAR Figure Number
33013-1272		
Sheet 1	Waste Disposal - Liquid, Reactor Coolant Drain Tank - P&ID -	11.2-2, Sheet 1
Sheet 2	Waste Disposal - Liquid, Reactor Coolant Drain Tank - P&ID	11.2-2, Sheet 2
33013-1273		
Sheet 1	Waste Disposal - Gas - P&ID	11.3-2, Sheet 1
Sheet 2	Waste Disposal - Gas - P&ID	11.3-2, Sheet 2
33013-1274	Waste Disposal - Gas, H_2 and N_2 and Gas Analyzer - P&ID	11.3-1
33013-1275		
Sheet 1	Waste Disposal - Gas, Hydrogen Recombiner - P&ID	6.2-79, Sheet 1
Sheet 2	Waste Disposal - Gas, Hydrogen Recombiner - P&ID	6.2-79, Sheet 2
33013-1276	Waste Disposal - Liquid, Polishing Demineralizers - P&ID	11.2-5
33013-1277		
Sheet 1	Steam Generator Blowdown - P&ID	10.7-6, Sheet 1
Sheet 2	Steam Generator Blowdown - P&ID	10.7-6, Sheet 2
33013-1278		
Sheet 1	Nuclear Sampling System - P&ID	9.3-10, Sheet 1
Sheet 2	Nuclear Sampling System - P&ID	9.3-10, Sheet 2
33013-1279	Postaccident Sampling System - P&ID	9.3-12
33013-1607	Fire Protection System Yard Loop - P&ID	9.5-4
33013-1863	Containment HVAC Systems, Containment Recirculating and Cooling System, Postaccident Charcoal Filters - P&ID	9.4-1
33013-1864	Containment HVAC Systems, Containment Auxiliary Charcoal Filters, Refueling Water Ventilation, Reactor Compartment and Control Rod Drive Cooling - P&ID	9.4-2
33013-1865	Containment HVAC Systems, Purge Supply - P&ID	9.4-3
33013-1866	Containment HVAC Systems, Purge Exhaust and Penetration Cooling - P&ID	9.4-4

Drawing Number	Title	Historical Link to UFSAR Figure Number
33013-1867	Control Building HVAC System, Control Room HVAC Control Room Postaccident Charcoal Filters, Control Room Lavatory Exhaust - P&ID	6.4-1
33013-1868	Control Building HVAC System, Relay Room Cooling, Battery Room Cooling and Ventilation - P&ID	9.4-18
33013-1869	Auxiliary/Intermediate Building HVAC Systems Cooling for Charging, Safety Injection, Containment, Spray, RHR, and Standby Auxiliary Feedwater Pumps, Nitrogen and Hydrogen Storage Vents - P&ID	9.4-8
33013-1870	Auxiliary/Intermediate Building HVAC Systems, Volume Control Tank Exhaust, Auxiliary Building Charcoal Filter, Auxiliary Building 1G Filter - P&ID	9.4-7
33013-1871	Auxiliary/Intermediate Building HVAC Systems, Intermediate Building Exhaust System, Spent Fuel and Decon Pit Exhaust System, Main Auxiliary Building Exhaust System - P&ID	9.4-6
33013-1872	Auxiliary/Intermediate Building HVAC Systems, Building Supply Air Systems - P&ID	9.4-5
33013-1873	Turbine/Miscellaneous Building HVAC Systems, Ventilation for Diesel Generators, Feed Pumps, Oil Storage, Turbine Building Gas Bottle Storage, Elevator, and Screen House - P&ID	9.4-9
33013-1874	Turbine/Miscellaneous Building HVAC Systems, Condensate Demineralizer (AVT) Building Ventilation - P&ID	9.4-10
33013-1875	Service Building HVAC Systems, Controlled Access Exhaust System and Air Handling Unit 1C - P&ID	9.4-11
33013-1876	Service Building HVAC Systems, Air Handling Units 1B and 1D - P&ID	9.4-12
33013-1877	Service Building HVAC Systems, Air Handling Unit 1A and Return Air Fan 1A - P&ID	9.4-15
33013-1878	Service Building HVAC Systems, Miscellaneous Service Building HVAC Systems - P&ID	9.4-16
33013-1879	Service Building HVAC Systems, Air Handling Unit 1E - P&ID	9.4-13
33013-1881	Service Building HVAC Systems, Service Building North End HVAC System - P&ID	9.4-14

Drawing Number	Title	Historical Link to UFSAR Figure Number
33013-1885		
Sheet 1	Circulating Water - P&ID	10.6-1, Sheet 1
Sheet 2	Circulating Water - P&ID	10.6-1, Sheet 2
33013-1886		
Sheet 1	Service Air System - P&ID	9.3-1, Sheet 1
Sheet 2	Service Air System - P&ID	9.3-1, Sheet 2
33013-1887	Instrument Air, Containment Building - P&ID	9.3-3, Sheet 1
33013-1888	Instrument Air, Containment Building - P&ID	9.3-3, Sheet 2
33013-1889	Instrument Air, Auxiliary Building - P&ID	9.3-4, Sheet 1
33013-1890	Instrument Air, Auxiliary Building - P&ID	9.3-4, Sheet 2
33013-1891	Instrument Air, Auxiliary Building - P&ID	9.3-4, Sheet 3
33013-1892	Instrument Air, Auxiliary Building - P&ID	9.3-4, Sheet 4
33013-1893	Instrument Air, Intermediate Building - P&ID	9.3-5
33013-1894		
Sheet 1	Instrument Air, Turbine Building - P&ID	9.3-6, Sheet 1
Sheet 2	Instrument Air, Turbine Building - P&ID	9.3-6, Sheet 2
33013-1895	Instrument Air, Turbine Building - P&ID	9.3-6, Sheet 3
33013-1896	Instrument Air, Turbine Building and Screen House - P&ID	9.3-7
33013-1897		
Sheet 1	Instrument Air, Condensate Demineralizer (AVT) Building - P&ID	9.3-8, Sheet 1
Sheet 2	Instrument Air, Condensate Demineralizer (AVT) Building - P&ID	9.3-8, Sheet 2
33013-1898	Instrument Air, Service Building - P&ID	9.3-9, Sheet 3
33013-1899		
Sheet 1	Instrument Air, Service Building - P&ID	9.3-9, Sheet 1
Sheet 2	Instrument Air, Service Building - P&ID	9.3-9, Sheet 2
33013-1900		
Sheet 1	Instrument Air Compressors - P&ID	9.3-2, Sheet 1
Sheet 2	Instrument Air Compressors - P&ID	9.3-2, Sheet 2

Drawing Number	Title	Historical Link to UFSAR Figure Number
33013-1901	Turbine Lube-Oil System - P&ID	10.7-9
33013-1903	Extraction Steam - P&ID	10.7-3
33013-1904	Turbine Gland Steam and Drains - P&ID	10.7-7
33013-1905	Gland Sealing Water - P&ID	10.4-5
33013-1907	Primary Water Treatment Chemical Supply Tanks - P&ID	9.2-5
33013-1908		
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1.8 CONFORMANCE TO NRC REGULATORY GUIDES

1.8.1 CONFORMANCE TO AEC SAFETY GUIDES

The information in this section represents the position of the R. E. Ginna Nuclear Power Plant in August 1972 at the time when RG&E applied for a Full-Term Operating License with respect to the AEC Safety Guides for Water Cooled Nuclear Power Plants, numbers 1 through 29. The information has not been generally updated. It has been revised to remove incorrect or misleading information. References to sections and figures refer to this UFSAR unless the references are to the original FSAR, in which case it is so stated and the referenced information has not been incorporated into the UFSAR.

1.8.1.1 Safety Guide 1 - Net Positive Suction Head for Emergency Core Cooling and Containment Heat Removal System Pumps

The net positive suction head (NPSH) of the residual heat removal pumps is evaluated for normal plant shutdown operation and for both the injection and recirculation phase operations of the design-basis accident. Recirculation operation gives the limiting NPSH requirements and the NPSH available is determined from the containment water level, the temperature and pressure of the sump water, and the pressure drop in the suction piping from the sump to the pumps.

The NPSH for the safety injection pumps is evaluated for both the injection and recirculation phase operations of the design-basis accident. The end of injection phase operation gives the limiting NPSH requirement and the NPSH available is determined from the elevation head and vapor pressure of the water in the refueling water storage tank (RWST) and the pressure drop in the suction piping from the tank to the pumps.

The NPSH for the containment spray pump is evaluated for both the injection and recirculation phase operations of the design-basis accident. The end of the injection phase operation gives the limiting NPSH requirement and the NPSH available is determined from the elevation head and vapor pressure of the water in the refueling water storage tank (RWST) and the pressure drop in the suction piping from the tank to the pumps.

1.8.1.2 Safety Guide 2 - Thermal Shock to Reactor Pressure Vessels

The effects of safety injection water on the integrity of the reactor vessel following a postulated loss-of-coolant accident have been analyzed using data on fracture toughness of heavy section steel both at beginning of plant life and after irradiation corresponding to approximately 40 years of equivalent plant life. The results show that under the postulated accident conditions, the integrity of the reactor vessel is maintained.

Fracture toughness data are obtained from a Westinghouse experimental program which is associated with the Heavy Section Steel Technology (HSST) Program at Oak Ridge National Laboratory and Euratom programs. Since results of the analyses are dependent on the fracture toughness of irradiated steel, efforts are continuing to obtain additional confirmatory data. Data on 2 in. thick specimens became available in 1970 from the HSST Program. This data indicated a strong temperature dependence with a rapid increase in toughness at approximately nil ductility temperature. Presently, 4 in. thick specimens are being irradiated and these will be tested in the spring of 1974. The HSST Program is scheduled for completion by 1974, at which time the reactor vessel thermal shock program will have been completed.

A detailed analysis considering the linear elastic fracture mechanism method, along with various sensitivity studies, was submitted to the AEC staff and members of the Advisory Committee on Reactor Safety.

Revised material for this report plus additional analysis and fracture toughness data were presented at a meeting with the Containment and Component Technology Branch on August 9, 1968, and forwarded by letter for AEC review and comment on October 29, 1968.

The analysis for the pressurized water reactor under the postulated conditions of Safety Guide 2 shows that no thermal shock problem exists. It is not anticipated that the continuing HSST Program will lead to any new conclusions about reactor vessel integrity under loss-of-coolant accident conditions.

1.8.1.3 Safety Guide 3 - Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Boiling Water Reactors

This safety guide is not applicable to the R. E. Ginna Nuclear Power Plant which is a pressurized water reactor.

1.8.1.4 Safety Guide 4 - Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors

Safety Guide 4 gives the assumptions used by the AEC to evaluate the design basis loss-of-coolant accident. This methodology was used by RG&E at that time to perform loss-of-coolant accident analyses. Current information is provided in Chapter 15.

1.8.1.5 Safety Guide 5 - Assumptions Used for Evaluating the Potential Radiological Consequences of a Steam Line Break Accident for Boiling Water Reactors

This safety guide is not applicable to the R. E. Ginna Nuclear Power Plant which is a pressurized water reactor.

1.8.1.6 Safety Guide 6 - Independence Between Redundant Standby (Onsite) Power Sources and Between Their Distribution Systems

The electrically powered safety systems are divided into two groups so that loss of either one will not prevent safety functions from being performed.

Each ac load group has a connection to the preferred (offsite) power source. In a situation where offsite power is not available, two diesel generators supply standby power to separate redundant load groups. There is no automatic connection between either the diesel generators or the load groups.

The dc system consists of two separate batteries, each connected to two battery chargers, which supply separate dc load groups. The Ginna design includes automatic transfers between the load groups. However, necessary fusing and electrical interlocks are provided to prevent paralleling of the two dc systems.

Routing and separation standards applicable to existing cables are those that were invoked at the time of cable installation. For more information, see Section 8.3.1.4.

1.8.1.7 Safety Guide 7 - Control of Combustible Gas Concentrations in Containment Following a Loss-of-Coolant Accident

Two hydrogen recombiner units are installed in the Ginna containment. The purpose of these units is to prevent the uncontrolled post-accident buildup of hydrogen concentrations in the containment.

The recombiner system consists of two full-rated subsystems, each capable of maintaining the ambient H_2 concentration at 2% by volume. Each subsystem contains a combustor, fired by an externally supplied fuel gas, employing containment air as the oxidant. Hydrogen in the containment air is oxidized in passing through the combustion chamber. Hydrogen gas is also used as the externally supplied fuel in order that noncondensable combustion products are avoided which would cause a progressive rise in containment pressure. Oxygen gas is made up through a separate containment feed to prevent depletion of 0_2 below the concentration required for stable operation of the combustor.

Each recombiner is equipped with an air supply blower to deliver primary combustion air and quench air to reduce the unit exhaust temperature, an ignition system, and associated monitoring and control instrumentation. The system is qualified to perform its function in a post-accident environment.

1.8.1.8 Safety Guide 8 - Personnel Selection and Training

Personnel selection and training for Ginna Station were completed before ANSI-N18.1, Proposed Standards for Selection and Training of Personnel for Nuclear Power Plants, was published. However, the existing personnel and positions conformed very closely with the requirements of ANSI-N18.1. Since that time, selection of personnel, their qualifications, training, and retraining were done to conform to ANSI-N18.1-1971 and subsequent regulatory guides. As of 2020, Ginna follows the guidelines of ANSI/ANS 3.1-2014 as supplemented by Regulatory Guide 1.8, June 2019, for personnel selection and training.

1.8.1.9 Safety Guide 9 - Selection of Diesel-Generator Set Capacity for Standby Power Supplies

The diesel-generator capacities were based on a conservative evaluation of power requirements in the event of a loss-of-coolant accident simultaneous with a loss of station reserve power supply.

Each of the generators has a nameplate continuous rating of 1950 kW with a 0.8 power factor at 900 rpm with three-phase, 60-cycle, 480-V operation. The units also have extended ratings of 2300 kW for 0.5 hr. and 2250 kW for 2 succeeding hours. While paragraph 2 of the Safety Guide regulatory position does not specifically apply to the load ratings of the Ginna diesels, it does indicate the desired conservatism. During the initial injection phase, which lasts less than 2.5 hr., the power requirement is less than 90% of the 2-hr limit of 2250 kW. Once this initial phase is completed, the power requirements are less than 95% of the continuous duty rating of the diesel.

During preoperational testing, the diesel was operated at the power levels specified above. The power required to run the safeguards loads under preoperational testing was less than that estimated because of the difficulties in simulating accident loads. The containment air, for instance, was less dense than that experienced in an accident and thus reduced the power loading. Because of this the diesel was tested at rated rather than actual load.

Both diesels are capable of starting, accelerating, and attaining rated voltage within 10 seconds of a loss of voltage on a safeguards bus. During testing, the loading sequence and timing has been checked and has performed satisfactorily. During this loading sequence, the voltage has not dropped below 75% of rated output and has returned to within 10% of rated voltage within 40% of the load sequence time interval. A load loss from 100% to zero power will not cause an overspeed trip of either diesel. Frequency checks during tests have not been addressed specifically, however, no unusual variations have been noticed.

The suitability of both diesels was confirmed through preoperational testing and in periodic testing done since that time.

1.8.1.10 Safety Guide 10 - Mechanical (Cadweld) Splices in Reinforcing Bars of Concrete Containments

Tension splices for bar sizes larger than #11 were made with Cadweld splice. To ensure the integrity of the Cadweld splice, the quality control provided for a random sampling of splices in the field. The selected splices were removed and tested to destruction. A sampling of splices was initially tested to destruction to develop an average ($\overline{X}$) and deviation (σ). Sufficient samples were tested to provide a 99% confidence level that 95% of the splices met the specification requirements. The distribution established permitted the development of the lower limit below which no test data should fall. If the result of any test fell below this limit, the subsequent or previous splice was sampled. If the result was above the lower limit, the process was considered to be in control. If this result was again below the lower limit, the process average was recalculated and an engineering investigation was required to determine the cause of the excess variation and to reestablish control the average of all tests was required to remain above the minimum tensile strength. As additional data became available, the average and standard deviation were updated. The actual frequency of testing carried out was one specimen for each 25 splices made for each crew for the first 250 splices made by that crew and one test for each 100 splices thereafter. In addition, where deformed bars were attached to structural steel members, specimens were made and tested to ensure that the weld of the splice to the member did not fail before the rebar or the splice. The frequency of testing these specimens was the same as that for the normal splice.

In sampling the Cadweld splices a test was concurrently performed on the rebar. Where the rebar failed prior to the splice, a check was provided on the ultimate strength of the rebar, thus providing a check on conformance with the manufacturer’s certifications and the ASTM standards. In addition, certified mill test reports were received from the rebar supplier and checked for conformance with specification requirements.

Where the special large size bars (i.e., 14S and 18S) were spliced, the Cadweld process was used so that the connection could develop the required minimum ultimate bar strength. Where Cadweld splice was used, including in the cylinder and dome, the splices were staggered a minimum of 3 ft. An exception to this practice is in the vicinity of the large openings. Where reinforcing bars are anchored to plates or shapes, such as is the case for the dome bars anchored into the cylinder and the interrupted hoop bars at penetrations, the Cadweld splices all occur on one plane. Lapped splices are detailed in accordance with ACI-63.

Where Cadweld splices were used to anchor reinforcing bars to a structural steel member, a procedure of testing coupons was used to demonstrate that the welding process was under control. This procedure required each welder to initially make coupons as qualification procedure. The procedure was repeated at a frequency of one coupon for each 100 production units. Each coupon required testing of two Cadweld connections.

In addition, the welding procedure complied with the specifications of the American Welding Society and provided for 100% visual inspection of welds.

1.8.1.11 Safety Guide 11 - Instrument Lines Penetrating Primary Reactor Containment

The containment pressure transmitter instrument lines penetrate the containment. These must be open following an accident, but have a manual isolation valve outside containment. Therefore, Safety Guide 11 is met as well as General Design Criteria 56 on another defined basis.

1.8.1.12 Safety Guide 12 - Instrumentation for Earthquakes

A strong motion accelerograph is installed at the Ginna plant and is located in the basement of the intermediate building. This location was chosen rather than the basement of the containment since it more easily facilitates periodic surveillance of the instrument (this would be difficult should the instrument be located in the basement of the containment), and the retrieval of the shock record can more readily be made.

The response of the accelerograph located in the basement of the intermediate building will be virtually the same as one located in the basement of the containment.

1.8.1.13 Safety Guide 13 - Fuel Storage Facility Design Basis

The spent fuel pool (SFP) is a reinforced-concrete structure with a seam-welded stainless steel plate liner. This structure is designed to withstand the anticipated earthquake loadings as a Seismic Category I structure so that the liner prevents leakage even in the event the reinforced concrete develops cracks.

All structures have been designed for wind loads in accordance with the requirements of the State of New York State Building Construction Code. The wind loads tabulated in this code are based on a design wind velocity of 75 mph at a height of 30 ft. above grade level. In addition, the spent fuel pool (SFP) has been evaluated with regards to tornado winds and missiles and found to be acceptable.

Interlocks have been provided on the auxiliary building crane to prevent the crane hook from passing over stored fuel and thus prevent heavy loads from being dropped on the spent fuel.

The area around the spent fuel pool (SFP) is enclosed by the auxiliary building. In addition to other ventilation systems in this building, a ventilation system is provided to provide a sweep of air specifically across the top of the spent fuel pool (SFP). Originally, air was only passed through a high efficiency particulate air filter before being exhausted to the atmosphere. Early in 1971, however, a charcoal filter, to be placed into operation during MODE 6 (Refueling), was added to this discharge system to filter out the iodine in the air and thus improve the design to account for the assumption that all fuel rods in one fuel bundle might be breached if a MODE 6 (Refueling) incident occurred.

The fuel pool has been evaluated on the basis of dropping a fuel cask into the spent fuel pool (SFP). While some damage could possibly occur to the liner, the cask will not break through the reinforced concrete to cause a major leak. In any case, the crane moving the cask would be single-failure proof, thus precluding the need to postulate the cask drop occurrence.

There are no spent fuel pool (SFP) designs, permanently connected systems, and/or other features that by maloperation or failure could cause loss of fuel storage coolant to the extent that fuel would be uncovered. A maloperation or failure in the filtering or cooling systems will not cause the fuel to be uncovered.

The spent fuel pool (SFP) is provided with level monitoring equipment which gives an alarm in the control room if the level drops. The radiation level just above the spent fuel pool (SFP) is also monitored. A reading of this level is indicated locally and at the control room. A radiation level above the setpoint will cause an alarm on the control board. The filtering system associated with the air just above the spent fuel pool (SFP) is always in operation. Before being exhausted from the plant this air always passes through high efficiency particulate air filters first. During MODE 6 (Refueling) operations this air is also filtered with impregnated charcoal filters. The addition of the charcoal filters to the airstream is done manually.

A spent fuel pool (SFP) cooling system is installed to remove decay heat. Also, nonseismic makeup systems including the fire protection system, are provided to add coolant to the pool.

1.8.1.14 Safety Guide 14 - Reactor Coolant Pump Flywheel Integrity

Precautionary measures, taken to preclude missile formation from primary coolant pump components, ensure that the pumps will not produce missiles under any anticipated accident condition.

The primary coolant pumps run at 1189 rpm, and may operate briefly at overspeeds up to 109% (1295 rpm) during loss of outside load. For conservatism, however, 125% of operating speed was selected as the design speed for the primary coolant pumps. For the overspeed condition, which would not persist for more than 30 seconds, pump operating temperatures would remain at about the design value.

Each component of the primary pumps has been analyzed for missile generation. Any fragments would be expected to be contained by the heavy stator.

The most adverse operating condition of the flywheels is visualized to be the loss-of-load situation. The following conservative design and operation conditions minimize missile production by the pump flywheels. The flywheels are fabricated from rolled, vacuum-degassed, ASME SA 533 Type B steel plates. Flywheel blanks are flame-cut from the plate, with allowance for exclusion of flame affected metal. A minimum of three Charpy V-notch tests are made from each plate parallel and normal to the rolling direction, to determine that each blank satisfies design requirements. A nil ductility transition temperature less than +10°F is specified. The finished flywheels are subjected to 100% volumetric ultrasonic inspection. The finished machined bores are also subjected to magnetic particle or liquid penetrant examination.

These design fabrication techniques yield flywheels with primary stress at operating speed to less than 50% of the minimum specified material yield strength at room temperature (100°F to 150°F). Bursting speed of the flywheels has been calculated on the basis of Griffith-Irwin's results (*Reference 1*) to be 3900 rpm, more than three times the operating speed.

A fracture mechanics evaluation was made on the reactor coolant pump flywheel. This evaluation considered the following assumptions:

A. Maximum tangential stress at an assumed overspeed of 125% compared to a maximum expected overspeed of 109%.

B. A through crack through the thickness of the flywheel at the bore.

C. 400 cycles of startup operation in 40 years.

Using critical stress intensity factors and crack growth data attained on flywheel material, the critical crack size for failure was greater than 17 in. radially and the crack growth data was 0.030 in. to 0.60 in. per 1000 cycles.

The original inservice inspection program included a complete ultrasonic volumetric inspection and surface examination of all exposed surfaces at approximately 10-year intervals, and in-place ultrasonic volumetric examination of areas of higher stress concentration at the bore and keyway at approximately 3-year intervals. This was consistent with Safety Guide 14. The new inservice inspection program is described in Section 5.4.1.2.5.

1.8.1.15 Safety Guide 15 - Testing of Reinforcing Bars for Concrete Structures

The 1972 codes for testing of reinforcing bars for concrete structures were not available at the time that Ginna Station was built. The codes and practices followed do generally conform to these standards, however.

The concrete reinforcement used in the containment building and other Seismic Category I structures is deformed bar intermediate grade billet-steel conforming to the requirements of ASTM A15-64, Specifications for Billet-Steel Bars for Concrete Reinforcement, with deformations conforming to ASTM A305-56T, Deformed Bars for Concrete Reinforcement. Special large size concrete reinforcing bars are deformed bars of intermediate grade billet-steel conforming to ASTM A408-64, Specifications for Large Size Deformed Billet Steel Bars for Concrete Reinforcement. Reinforcing steel conforming to these specifications has a tensile strength of 70,000 psi to 90,000 psi and a minimum yield point of 40,000 psi.

All splicing and anchoring of the concrete reinforcement is in accordance with ACI 318-63. There was no splicing of bars by arc welding. The special large size bars were spliced by the Cadweld process.

It is to be noted that intermediate grade reinforcing steel is the highest ductility steel commonly used for construction. Certified mill reports of chemical and physical tests were submitted to the engineer, Gilbert Associates, Inc., for review and approval. Each bar was branded in the deforming process to carry identification as to the manufacturer, size, type, and yield strength, for example:

- B -Bethlehem.
- 18 -Size 18S.
- N -New billet steel.
- Blank -A-15 and A-408 steel.
- 6 -A-432 (60,000 psi yield).
- 7 -A-431 (75,000 psi yield).

Because of the identification system and because of the large quantity, the material was kept separated in the fabricator’s yard. In addition, when loaded for mill shipment, all bars were properly separated and tagged with the manufacturer’s identification number.

Visual inspection of the bars was made in the field for inclusions and representative randomly selected samples of reinforcing bar stocked onsite were tested for user’s tensile tests.

The specifications stipulate that "arc welding concrete reinforcement for any purpose including the achievement of electrical continuity shall not be permitted unless noted otherwise on the drawings."

Concrete cover of reinforcing bar was at least the minimum specified by ACI-318.

1.8.1.16 Safety Guide 16 - Reporting of Operating Information

During the initial operating period that Ginna Station was producing power, reporting followed the intent of the regulations in effect at that time, specifically 10 CFR 20, 40, 50, 70, and 73. Therefore, RG&E conformed to the guidance of Safety Guide 16 as well as complying with all reporting requirements set forth in the Technical Specifications.

New reporting requirements have been instituted since this initial period and other requirements have been altered. RG&E has continued to comply with current NRC requirements. These include regulations such as 10 CFR 20, 21, 26, 50, 55, 70, 73, and 74, and selected NRC bulletins and generic letters such as GL 97-02. Other reporting requirements are contained in the Technical Specifications, Offsite Dose Calculation Manual (ODCM), and Technical Requirements Manual (TRM). Many of these various reporting requirements are addressed in plant procedures.

1.8.1.17 Safety Guide 17 - Protection Against Industrial Sabotage

The Rochester Gas and Electric Corporation submitted a proprietary document, Security at the Ginna Facility, to the AEC by cover letter dated October 8, 1971. This document describes in detail the implementation by RG&E of those sections of the Safety Guide applying to control of access and selection of personnel. The Security Plan was updated by RG&E submittals of January 19, 1978, and April 12, 1983. The plan is maintained current in compliance with 10 CFR 50.54(p).

1.8.1.18 Safety Guide 18 - Structural Acceptance Test for Concrete Primary Reactor Containments

1.8.1.18.1 Structural Integrity Test

After completion of the construction of the entire containment vessel, a structural integrity test was performed, where a pneumatic pressure of 69 psig (115% of the design pressure of 60 psig) was maintained for approximately 4 hours. The pressurization of the vessel was done so as to permit readings and measurements which are more fully described hereafter. The readings and measurements were made during the initial pressurization (with pressure maintained a minimum of 3 hr at 0 psig, 14 psig, 35 psig, 60 psig, and at maximum test pressure of 69 psig, and thereafter during depressurization at 60 psig, 35 psig, and 0 psig. Except for the maximum pressure level (69 psig), the vessel pressure was slightly increased above the level at which the measurements were taken; and the pressure was then reduced to the specified value and observations made after at least 10 minutes to permit an adjustment of strains within the structure. Because the structure is so large, displacement measurements were made with sufficient precision to serve as confirmation of previously calculated response.

The test program further included, in addition to displacement measurements, a continuous visual examination of the vessel to observe concrete cracking. Observations of the entire vessel surface were made from existing or temporary platforms with special attention given to pertinent locations, including major discontinuities. A complete description of the instrumentation used to measure response is described below.

Predicted displacements developed for an internal pressure of 69 psig, which is the maximum pressure for the structural proof test, is included below. Although strain measurements were made, no predicted measurements are provided consistent with agreements previously documented in Appendices A, B, and C of Gilbert Associates, Inc., Report GAI 1720 (*Reference 2*). Strain values obtained, however, are analyzed to determine magnitude and direction of principal strains.

Maximum predicted crack widths for specifications are described below.

1.8.1.18.2 Instrumentation

The installation of all targets, linear variable differential transformers, whitewash for crack observations, load cells, tapes, strain gauges, photoelastic disks, cameras, junction boxes, wires, readout instruments, support structures, and platforms were completed prior to initiating pressurization of the vessel. The location for all instrumentation is shown in Table I of GAI 1720 (*Reference 2*). In addition, the covers on the enclosures over the tendon anchors and the wax surrounding the anchor head were removed to permit inspection of the anchorage, including button heads, during the test. People were stationed at the three locations for theodolite measurements, at the ledge for tendon anchorage inspection, and at each location where crack measurements were made. These people were equipped with communication means to maintain contact with a control located in the intermediate building at elevation 253 ft 6 in. where read-out instruments were located. In addition, three people were available to travel over accessible walkways to inspect the outer vessel surface.

The type of instruments used were as follows:

1. Jig transit with scales and targets.
2. Invar tapes.
3. Linear variable differential transformers.
4. Strain gauges.
5. Rosette strain gauges.
6. Photoelastic disks.
7. Load cells.

1.8.1.18.3 Displacement Measurements

Cylinder base rotation and displacement were measured utilizing linear variable differential transformers at three azimuths, one of which was directly below the equipment access opening. At each azimuth two linear variable differential transformers were located near the base of the structure with 6 ft. vertical separation. These radial displacements were used to determine the actual base rotation. Also, at each azimuth one linear variable differential transformer was used to determine the vertical displacement of the elastomer pad.

Radial displacement measurements were made at a total of 15 locations using a jig transit, base targets, and mounted scales.

A base target was attached to the structure at each of three different azimuths around the base of the cylinder. Five scales were attached (at each azimuth), three along the height of the cylinder and one each just above and below the ledge (i.e., elevation 343 ft. 2 in.). Relative radial displacements were determined at each scale location by aligning the transit with the base target and by plunging the scope up from the base target to each scale. Variations in the scale readings from the original reading indicated the amount of displacement.

The vertical displacement of the cylinder at the top (relative to the base ring at three azimuths for side wall elongation and average tendon strain) was determined using three invar tapes. The tapes were mounted at the ledge and extended down to the base ring, where weights tensioned the tapes. A scale at the base was read using an engraved mark on the tape to indicate relative elongations.

Linear variable differential transformers were utilized at 28 locations on concrete around the equipment access opening to measure horizontal and vertical displacements. Along the horizontal axis, on one side only, six horizontal and six vertical displacements were obtained to a point 21 ft. out from the edge of the hole. An identical set of displacements was obtained on the vertical axis above the hole. Additionally, on the horizontal and vertical axis, of those displacements previously mentioned, another point on each axis was selected to measure vertical and horizontal displacements at a point 2 ft from the opposite edge of the hole.

Displacement measurement accuracies are as follows: the jig transits, using an optical micrometer, had a resolution of 0.001 in. and an accuracy of 0.005 in. to 0.010 in. The linear variable differential transformers and associated instrumentation had a resolution of better than 0.001 in. and an accuracy of 0.002 in. to 0.005 in.

1.8.1.18.4 Strain Measurements

A total of 46 reinforcing bars were instrumented for strain measurements, 28 were at locations similar to linear variable differential transformer displacement measurement locations around the equipment access opening, and 18 were at locations above and below the ledge.

The liner was instrumented with rectangular rosettes at six locations, to indicate general strain in regions unaffected by geometric discontinuities, and at 32 locations around four typical penetrations. Eight rosettes were used at each penetration.

Strain gauges were attached to the tendon-anchorage bearing plates at tendons 13, 53, 93, and 133.

Load cells were installed under the button head of tendons 13, 53, 93, and 133. The strain gauges on reinforcing bars and associated instrumentation had a resolution of 0.4 micro-inch per inch strain and an accuracy of 2 to 3 micro-inches per inch. The strain gauges on the steel liner had a resolution of 1 micro-inch per inch and an accuracy of approximately 5 micro-inches.

The strain gauges on the bearing plates and the associated instrumentation had a resolution of 1 micro-inch per inch and an accuracy of approximately 5 micro-inches per inch. The instrumentation utilized for the tendon load cell had a measuring accuracy of 0.5% of full load capacity.

Photoelastic disks, 1.5 in. to 2 in. in diameter, were placed on the liner, around the same four penetrations where strain gauges were installed, to qualitatively augment the local values indicated by the strain gauges. Approximately 15 disks were located in one quadrant for each of four penetrations. (This resulted in approximately 25% surface coverage up to one diameter away from the opening.)

1.8.1.18.5 Test Results

Reading and recording of all measurements were made just prior to pressurizing, after depressurizing, and at each pressure increment, except that only one quadrant of photoelastic disks at each penetration were photographed while the structure was pressurized.

The identification and location of the instruments are shown on Figures 2 through 5 of GAI Report No. 1720 (*Reference 2*). These instruments were located in such a way that the actual response of the vessel during the test was determined and verified, with the criteria established prior to the performance of the test. The location of scales and gauges are as described in Table I of GAI Report No. 1720.

The results of the structural integrity test showed the stresses, strains, and displacements were within the specified limits and the GAI predicted results. The whitewash areas revealed crack patterns and spacings in good agreement with the GAI prediction; there was no horizontal cracks in dome concrete except for construction joints. The base shear restraint was stiffer than anticipated. The strains and displacements of the cylinder wall, the discontinuity of dome and cylinder wall, and dome revealed that the structural stiffness of the containment vessel is greater than anticipated.

The structural capacity of the containment met and exceeded its imposed criteria. A detailed analysis and description of the Ginna containment structural integrity test is contained in GAI Report No. 1720.

1.8.1.19 Safety Guide 19 -Nondestructive Examination of Primary Containment Liners

1.8.1.19.1 Test Provisions

The weld seams in the liner plate are covered with a test channel to permit testing for leaks. Except for the equipment access hatch, all penetrations provide a double barrier against leakage and can be pressurized to permit testing of leak-tightness.

All penetrations through the containment reinforced concrete pressure barrier for pipe, electrical conductors, ducts, and access hatches are of the double barrier type.

In general, a penetration consists of a sleeve embedded in the reinforced concrete wall and welded to the containment liner. The weld to the liner is shrouded by a test channel which is used to demonstrate the integrity of the joint. The pipe, duct, or access hatch passes through the embedded sleeve and the ends of the resulting annulus are closed off, generally by welded end plates. Piping penetrations have a bellows type expansion joint mounted on the exterior end of the embedded sleeve where required to compensate for differential motions. The only exceptions to providing an annulus about piping occurs for the three drain lines from sump B.

Penetrations are designed with double seals so as to permit individual testing at the required test pressure.

All penetrations are provided with test canopies over the liner to penetration sleeve welds. Each canopy, except those noted below, is connected to and pressurized simultaneously with the annulus between the pipe and sleeve penetration when under test. The exceptions are the canopy for the fuel transfer penetration which must be pressurized independently of the annulus because of the separation posed by the transfer canal liner and the three pipe penetrations in sump B in which only the canopies are pressurized as there are no annuli.

1.8.1.19.2 Examination of Welds

All welded joints for the penetrations including the reinforcement about the openings (i.e., sleeve to reinforcing plate seam) were fully radiographed in accordance with the requirements of the ASME Nuclear Vessels Code for Class B Vessels, except that non-radiographable joint details were examined by the liquid penetrant method. For fully radiographed welds, acceptance standards for porosity are as shown in Appendix IV of the Nuclear Vessels Code. (The ASME Unfired Pressure Vessels Code states that porosity is not a factor in the acceptability of welds not required to be fully radiographed.)

Longitudinal and circumferential welded joints of the liner within the main shell, the welded joint connecting the dome to the cylinder, and all joints within the dome were inspected by the liquid penetrant method and spot radiography. All penetrations including the equipment access door and the personnel locks were examined in accordance with the requirements of the ASME Nuclear Vessels Code for Class B Vessels. All other shop fabricated components, including the reinforcement about openings, were fully radiographed. All other joint details were examined by the liquid penetrant method. Full radiography is performed in accordance with the procedures and governed by the acceptability standards of Paragraph N-624 of the ASME Nuclear Vessels Code. Spot radiography is performed in accordance with the procedures and governed by the standards of Paragraph UW-52 of the ASME Unfired Pressure Vessels Code. Methods for liquid penetrant examination were in accordance with Appendix VIII of the ASME Unfired Pressure Vessels Code.

1.8.1.19.3 Pressure Tests

All piping penetrations and personnel locks were pressure tested in the fabricator's shop to demonstrate leak tightness and structural integrity.

In order to ensure that the joints in the liner plate and penetrations as well as all weld connections of test channels were leak tight, it was required that all welds be examined by detecting leaks at 69 psig test pressure using a soap bubble test or a mixture of air and Freon, and 100% of detectable leaks be arrested. These tests were preliminary to the performance of the initial integrated leak rate test which ensured that the containment leak rate was no greater than 0.1% of the contained volume in 24 hours at 60 psig.

The liner weld seams were also examined by pressurizing the test channels to design pressure (60 psig) with a mixture of air and Freon, and checking all seams with a halogen leak detector. All detectable leaks were corrected by repairing the weld and retesting.

1.8.1.19.4 Quality Control Provisions

The following quality control provisions were employed in the welding procedure for the liner:

The qualification of welding procedures and welders was in accordance with Section IX, Welding Qualifications, of the ASME Boiler and Pressure Vessel Code. Contractor shall submit welding procedures to the Engineer for review.

The qualification tests described in Section IX, Part A, include guided bend tests to demonstrate weld ductility. All penetrations shall be examined in accordance with the requirements of the ASME Nuclear Vessels Code for Class B Vessels. Other shop fabricated components including the reinforcement about openings shall be fully radiographed. All non-radiographable joint details shall be examined by the liquid penetrant method.

Conformance to this code was adhered to in all applicable cases.

1.8.1.20 Safety Guide 20 - Vibration Measurements on Reactor Internals

A vibration analysis and test program was developed for Ginna Station by Westinghouse Corporation. The preoperational test program and its results are discussed in Section 14.6. The results show that the vibration of the reactor internals for the Ginna plant are well within the existing criteria.

A program was conducted during the first MODE 6 (Refueling) shutdown of the Ginna reactor (March 1971) to inspect and evaluate the performance of the reactor internals and core components. This inspection program was based on an inspection of all components, with emphasis on the thermal shield area since the thermal shield has previously been the most vulnerable problem area.

The structures inside and outside of the lower internals, the upper internals, three control rod drive shafts, and all rod cluster control assembly control rods were inspected using a closed-circuit underwater television and/or boroscope. All of the inspections performed by television were recorded on video tape; photographs were taken through the boroscope to record that portion of the inspection. This inspection revealed no problem areas in any of the items inspected.

The inspection program is described in Westinghouse report WCAP 7780, October 1971, Robert E. Ginna Nuclear Generating Station, March 1971 Refueling Shutdown Reactor Internals and Core Components Evaluation.

1.8.1.21 Safety Guide 21 - Measuring and Reporting Effluents From Nuclear Power Plants

Starting on January 1, 1972, plant effluent monitoring and reporting was prepared in the format given in Appendix A of Safety Guide 21 and submitted to the State of New York on a monthly basis. A report in the format of Appendix A was provided to the AEC for the year 1971. The Technical Specifications, as revised on March 1, 1972, followed the intent of Safety Guide 21 for measuring and recording the plant effluents. Technical Specifications provide the requirements for a Radiological Effluent Controls Program. Plant records will be maintained to demonstrate that the sensitivity of analysis is within the limits given in the safety guide.

An onsite meteorological tower was fully operational early in 1965 and was used extensively in the collection of preoperational meteorological data. During early 1972, the recording instrumentation was relocated inside the turbine building, and subsequently the data collection was moved to the Plant Process Computer System (PPCS). Data are currently being used in upgrading calculations of dilution factors for radiological releases.

Preoperational onsite meteorological data were evaluated to provide a basis for controlled radiological gas release limits, accident analysis, and storm prediction criteria in the FSAR.

Basic and critical meteorological parameters are recorded at the Ginna site. See Section 2.3.3 for additional details. This information provides RG&E with the capability of assessing the potential dispersion characteristics of radioactive releases to the environment through the atmosphere. Such assessments provide RG&E with the ability to demonstrate that operations are well within the limits of 10 CFR 20. Current practice is to maintain effluent releases within 10 CFR 50, Appendix I limits, as specified in the Offsite Dose Calculation Manual (ODCM).

1.8.1.22 Safety Guide 22 - Periodic Testing of Protection System Actuation Functions

The plant protection system has been designed to permit periodic testing to extend to and include actuation devices and actuated equipment whenever practicable. While it is not possible to operate all actuation devices (such as trip of control rods) or significantly vary most of the operating parameters (such as coolant pressure) during operation, it is possible to test most equipment when the plant is in full power operation.

The bistable portions of the protective system (i.e., relays, bistables, etc.) provide trip signals only after signals from analog portions of the system reach preset values. Capability is provided for calibrating and testing the performance of the bistable portion of protective channels and various combinations of the logic networks during reactor operation.

The analog portion of a protective channel (i.e., sensors and amplifiers) provides analog signals of reactor or plant parameters. The following means are provided to permit checking the analog portion of a protective channel during reactor operation:

A. Varying the monitored variable.

B. Introducing and varying a substitute transmitter signal.

C. Cross checking between identical channels or between channels which bear a known relationship to each other and which have readouts available.

During operation it is also possible to test the pumps used in a safety injection. For instance, each high-head safety injection pump can be and is tested in accordance with the inservice pump and valve testing program.

Testing that cannot be done during operation is completed during MODE 6 (Refueling) shutdowns. The safety injection system is tested to see that as a system it can perform according to design. When completed, the test shows that separate and redundant actuation signals are operative and that the valves and pumps that are required for safety injection are indeed operable.

Where the ability of a system to respond to a bona fide accident signal is intentionally bypassed for the purpose of performing a test during reactor operation, the expansion of the bypass condition to redundant systems is prevented. In addition, the condition is automatically indicated to the reactor operator in the main control room.

1.8.1.23 Safety Guide 23 - Onsite Meteorological Programs

The Ginna plant site meteorology is described in Section 2.3. The 2 year preoperational meteorological program data is summarized in Section 2.7 of the original FSAR.

These data were utilized by the NRC and RG&E for accident analysis and gaseous release limit determination during the initial license application for a 1300 MWt rating and, more recently, during the review of the application by RG&E to increase its licensed power level from 1300 MWt to 1520 MWt. More information on the meteorological tower is provided in the discussion of Safety Guide 21.

1.8.1.24 Safety Guide 24 - Assumptions Used for Evaluating the Potential Radiological Consequences of a Pressurized Water Reactor Radioactive Gas Storage Tank Failure

The activity in a gas decay tank is taken to be the maximum amount that could accumulate from operation with cladding defects in 1% of the fuel rods. The maximum activity is obtained by assuming the noble gases xenon and krypton are accumulated with no release over a full core cycle. This postulated amount of activity, one reactor coolant system equilibrium cycle inventory, is 4.6×10^4 Ci equivalent Xenon-133. This value is particularly conservative because some of this activity would normally remain in the coolant, some would have been dispersed earlier through the stack, and the shorter lived isotopes would have decayed substantially. Current assumptions for postulated activity are provided in Section 15.7.1.1.4.

Samples taken from gas storage tanks in pressurized water reactor plants in operation show no appreciable amount of iodine.

To define the maximum doses, the release is assumed to result from gross failure of a gas decay tank giving an instantaneous release of its volatile and gaseous contents to the atmosphere.

The maximum whole-body beta-gamma dose, based on meteorology previously described in Safety Guide 4, is less than a few rem (less than three). This is well below the 25 rem guide line value in 10 CFR 100.

1.8.1.25 Safety Guide 25 - Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors

The Ginna spent fuel pool (SFP) charcoal filter system was designed and constructed prior to the issuance of Safety Guide 25. An analysis based on Regulatory Guide 1.25 was performed and was described in Section 15.7.3 of the original FSAR. Radiological consequences were calculated to be less than 34 rem to the thyroid at the exclusion area boundary, which was well below the 10 CFR 100 exposure guidelines. An analysis based on Regulatory Guide 1.25 was performed and is described in Section 15.7.3. Current calculated radiological consequences are provided in Section 15.7.3.

1.8.1.26 Safety Guide 26 - Quality Group Classification and Standards

Although Safety Guide 26 was not in effect when Ginna Station was constructed, RG&E subsequently classified the systems in Ginna Station in accordance with this guide.

1.8.1.27 Safety Guide 27 - Ultimate Heat Sink

The circulating water intake system of Ginna Station is designed to provide a reliable supply of Lake Ontario water, regardless of weather or lake conditions, to a suction of the condenser circulating water pumps, house service water pumps, and the fire water pumps. With two pumps operating, the nominal flow of the circulating water system is approximately 333,000 gpm. Operation of a single circulating water pump reduces the nominal flow rate by about 50%.

In meeting the high reliability requirements of this safety guide, the intake system is completely submerged below the surface of the lake. A 10 ft. diameter reinforced concrete lined tunnel, driven through bedrock, extends 3100 ft. northerly from the shore line. The tunnel rises vertically and connects to a reinforced-concrete inlet section. The minimum mean monthly lake level of record (243.0 ft. msl) will result in a depth of water of 26 ft. above the lowest entrance into the intake structure.

The probability of water stoppage due to plugging of the inlet has been reduced to an extremely low value by incorporating certain design features in the system. This includes removal of all 24 plant original heater racks from the faces of the intake structure in order to mitigate the effects of frazil ice on plant operations, as the unheated portions of the heater racks provided an anchor point for frazil ice accumulation.

Redundant traveling water screens, located in the screen house will remove trash from the cooling water. At conditions of full flow (approximately 355,000 gpm) the velocity at the intake screen racks is 0.8 ft/sec. The plant cooling water requirements during an accident would be approximately 10,000 gpm, which would result in a velocity of 0.02 ft/sec.

In addition, water enters on a full 360-degree circle thereby protecting against the possibility of stoppage by a single large piece of material. The low velocity, plus the submergence, provides assurance that floating ice will not plug the intake. The only phenomenon that is credible to contribute to the plugging would be the accumulation of frazil ice on the screen racks. For this reason, all 24 plant original heater racks were removed as of September 2018 to mitigate the effects of frazil ice accumulation on plant operations; specifically, to prevent ice buildup on the unheated metal portions of the heater racks that has been shown to restrict flow and lower Screenhouse level.

Warm water recirculation is provided in the screen house to melt any ice that might reach this point. Additional information is provided in Section 2.4 and Appendix 2A. Refer to Section 10.6.2.1 for an update to this historical information.

1.8.1.28 Safety Guide 28 - Quality Assurance Program Requirements

The standards, specifications, and guidelines existing at the time Ginna Station was constructed, pertinent to quality assurance, were at least met or exceeded. Details of the quality assurance program implemented are described in Chapter 1 of the original FSAR.

A quality assurance program was instituted for the operation, maintenance, and system redesign of the Ginna plant that conformed to the guidelines of N45.2-1971.

1.8.1.29 Safety Guide 29 - Seismic Design Classification

Although this Safety Guide had not been published at the time of the Ginna Station design and construction, the seismic classifications generally conform to this Guide. The seismic classification of equipment is provided in Section 3.2 and in the UFSAR system descriptions and is noted on the Ginna piping and instrumentation diagrams (P&IDs).

1.8.2 CONFORMANCE TO DIVISION I REGULATORY GUIDES

The information in this section represents the position of the R. E. Ginna Nuclear Power Plant with respect to certain of the NRC Division 1 Regulatory Guides in December 1973. The information was submitted to the NRC as Supplement 1 to the Technical Supplement Accompanying the Application for a Full-Term Operating License. Regulatory Guides 1.3, 1.5, and 1.5.6 are not applicable to the R. E. Ginna Nuclear Power Plant and are not discussed. Regulatory Guides 1.4, 1.10, 1.15, 1.17, 1.18, 1.19, and 1.29 are addressed because either the guides or the R. E. Ginna positions were revised since the submission of the positions relative to like-numbered Safety Guides presented in Section 1.8.1. Regulatory Guides 1.30 through 1.143 were not addressed as Safety Guides in Section 1.8.1 and are included in this section.

1.8.2.1 Regulatory Guide 1.4 - Assumptions Used for Evaluating the Potential Radiological Consequences of a Loss-of-Coolant Accident for Pressurized Water Reactors

This subject is discussed in Section 1.8.1.4.

1.8.2.2 Regulatory Guide 1.10 - Mechanical (Cadweld) Splices in Reinforcing Bars of Category I Concrete Structures

This subject is discussed in detail in Section 1.8.1.10.

1.8.2.3 Regulatory Guide 1.15 - Testing of Reinforcing Bars for Category I Concrete Structures

This subject is discussed in detail in Section 1.8.1.15.

1.8.2.4 Regulatory Guide 1.16 - Reporting of Operating Information

This subject is discussed in detail in Section 1.8.1.16.

1.8.2.5 Regulatory Guide 1.17 - Protection of Nuclear Plants Against Industrial Sabotage

This subject is discussed in detail in Section 1.8.1.17.

1.8.2.6 Regulatory Guide 1.18 - Structural Acceptance Test for Concrete Primary Reactor Containments

This subject is discussed in detail in Section 1.8.1.18.

1.8.2.7 Regulatory Guide 1.19 - Nondestructive Examination of Primary Containment Liner Welds

A description of the inspection methods employed during construction is presented in Section 1.8.1.19.

1.8.2.8 Regulatory Guide 1.26, Revision 3 - Quality Group Classifications & Standards for Water, Steam, and Radioactive - Waste Containing Components of Nuclear Power Plants

A classification process is established within station procedures to identify components, systems, and structures that are safety related (SR), safety significant (SS), or Non-Nuclear Safety (NS). Criteria are based on information contained in the Updated Final Safety Analysis Report (UFSAR), licensing commitments, guidelines contained in NRC Regulatory Guides, and functional guidance derived from ANSI/ANS 51.1 -1983.

1.8.2.9 Regulatory Guide 1.29, Revision 3 - Seismic Design Classification

The Ginna plant components, systems, and structures were classified for seismic design as tabulated in Section 3.2. Current seismic classifications are provided in Section 3.2, applicable sections of the UFSAR, and on the Ginna P&IDs. Comparison of the Ginna plant seismic classification system with that recommended by Regulatory Guide 1.29 shows close agreement between the two classification systems.

<u>Plant Operation</u>

Seismic design requirements for existing structures, systems, and components performing functions listed in positions C.1 and C.3 of the Regulatory Guide are specified in the UFSAR. New structures, systems, and components, and configuration changes meet the seismic design requirements of this regulatory guide or the UFSAR. The pertinent quality assurance requirements of 10CFR50, Appendix B are applied as required by positions C.1 and C.4 of this Regulatory Guide, irrespective of an item's seismic design. Portions of existing structures, systems, and components with failure consequences described in position C.2 of this guide are designed and constructed to seismic requirements specified in the UFSAR. New structures, systems, and components, and configuration changes meet the design and construction seismic requirements of the UFSAR or this Regulatory Guide. A quality assurance program similar to 10CFR50, Appendix B is applied to the SSE failure prevention function of these items. These items are not considered basic components pursuant to 10CFR21.

1.8.2.10 Regulatory Guide 1.30 - Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electrical Equipment

Regulatory Guide 1.30 and the related IEEE Standard 336-1971 were published after the construction of the R. E. Ginna Nuclear Power Plant. The IEEE Standard 336-1971 is, however, discussed in Section 1.8.3 as it applied to Ginna Station in August 1972.

<u>Plant Operation</u>

Operational commitments to this Regulatory Guide are discussed in detail in the Quality Assurance Topical Report (QATR). The QATR is cited in Section 17.2 of the UFSAR and is submitted to the NRC in accordance with the provisions of 10 CFR 50.54(a).

Requirements for checks, calibrations, and tests of instrument channels are given in the Technical Specifications.

1.8.2.11 Regulatory Guide 1.31 - Control of Stainless Steel Welding

Regulatory Guide 1.31 was published after the fabrication cycle for the Ginna plant. However, the stainless steel welding for the Ginna plant meets the intent of Regulatory Guide 1.31.

All welding was conducted using those procedures that have been approved by the ASME Code Rules of Section III and IX. The welding procedures were qualified by nondestructive and destructive testing according to the ASME Code Rules of Section III and IX.

When these welding procedure tests were performed on test welds made from base metal and weld metal materials which were from the same lots of materials used in the fabrication of components, additional testing was frequently required to determine the metallurgical, chemical, physical, corrosion, etc., characteristics of the weldment. The additional tests that were conducted on a technical case basis are as follows: light and electron microscopy, elevated temperature mechanical properties, chemical check analysis, fatigue tests, intergranular corrosion tests, or static and dynamic corrosion tests within reactor water chemistry limitations.

The following welding methods were tested individually and in multiprocess combinations, using the following energy input ranges for the respective method as calculated by the formula:

$$H = \frac{(E)(I)(60)}{S}$$

(Equation 1.8-1)

where:

H = J/in.

E = volts

I = amperes

S = travel speed, in./min

Welding Process Method	Energy Input Range (kJ/in.)
Manual shielded tungsten arc	20 to 50
Manual shielded metallic arc	15 to 120
Semiautomatic gas shielded metal arc	40 to 60
Automatic gas shielded tungsten arc-hot wire	10 to 50
Automatic submerged arc	60 to 140
Automatic electron beam-soft vacuum	10 to 50

The interpass temperature of all welding methods was limited to 350°F maximum. All full penetration welds were inspected in accordance with Article NB5000 of the 1965 ASME Section III Code rules. Welding materials were required to conform and were controlled in accordance with Subarticle NB2400 of the 1965 ASME Section III Code rules.

In addition, the austenitic stainless steel welding material used for joining austenitic stainless steel base materials in the reactor coolant pressure boundary, systems required for reactor shutdown and emergency core cooling, and the core structural load-bearing members conforms to ASME Material Specifications SA-298 and SA-371. These materials were tested and qualified according to the requirements stipulated in the 1965 ASME Boiler and Pressure Vessel Code Sections II, III, and IX, respectively. All of these welding materials conform to ASME weld metal analysis A-7.

Plant Operation

Regulatory Guide 1.31 is the basis for stainless steel welding procedures. Each procedure is designed to produce high quality welds using the variables and methods outlined in the procedure. Qualification of these procedures is done in accordance with Section III and Section IX of the ASME Boiler and Pressure Vessel Code.

In production welding, strict control is maintained to ensure that every step that may affect the quality of the final weld is supervised and checked for compliance with the proper criteria and that the welding procedure is being followed. The consumables used for stainless steel welding jobs meet the requirements of Section II of the ASME Code and are purchased with actual chemical composition and mechanical properties certified. All stainless steel welds are nondestructively examined to verify their quality and code compliance.

1.8.2.12 Regulatory Guide 1.32 - Use of IEEE Standard 308-1971, Criteria for Class IE Electric Systems for Nuclear Power Generating Stations

Conformance to IEEE Standard 308-1971 is discussed in Section 1.8.3. Regulatory Guide 1.32 (formerly Safety Guide 32, August 1972) identifies two areas of possible conflict between IEEE Standard 308 and Criterion 17: availability of offsite power and battery charger supply.

The availability of offsite power is discussed fully in Chapter 8. The electrical power system is designed with a single station auxiliary (startup) transformer, which gives immediate access to two independent sources of offsite power. In the event that this access is not available, either of the two backup diesel generators is capable of supplying safeguards loads. As an independent additional source of offsite power, the unit auxiliary transformer can be supplied from the normally outgoing power feeder by disconnecting the flexible generator bus disconnects. This can be accomplished in a short time, (less than 8 hr) after which all the vital loads could be supplied from the unit auxiliary transformer. Because of the multiple immediate access power sources, the one delayed access power source conforms to Regulatory Guide 1.32 and General Design Criteria 17.

The battery chargers are discussed in Section 1.8.3. Operating experience has proven that the battery charger capacity is more than sufficient to supply all long-term plant loads while restoring the batteries from the minimum charge to the fully charged state.

Station Battery Surveillance frequencies are controlled under the Surveillance Frequency Control Program as defined in Technical Specifications.

1.8.2.13 Regulatory Guide 1.33 -Quality Assurance Program Requirements (Operation)

ANSI N18.7-1972, Administrative Controls for Nuclear Power Plants, and ANSI N45.2-1971, Quality Assurance Program Requirements for Nuclear Power Plants, were used as a basis for developing the initial Ginna Station Operational Quality Assurance Program that is cited in Section 17.2. Appendix A to Regulatory Guide 1.33 was used as guidance in developing procedures for operating and maintenance activities.

Plant Operation

Operational commitments to this Regulatory Guide are discussed in detail in the Quality Assurance Topical Report (QATR). The QATR is cited in Section 17.2 of the UFSAR and is submitted to the NRC in accordance with the provisions of 10 CFR 50.54(a).

1.8.2.14 Regulatory Guide 1.34 -Control of Electroslag Weld Properties

Regulatory Guide 1.34 was published after the construction of the Ginna Nuclear Power Plant; however, the electroslag welding performed for the Ginna plant meets all of the guidelines of Regulatory Guide 1.34. The specific applications of electroslag welding for the Ginna plant were for the shop assembly welds of the primary coolant system, 90-degree piping elbows, and the reactor coolant pump casings, as discussed in detail in Section 5.2.3.1.2.

Assembly of the elbows was accomplished using a procedure specifying the following parameters:

A. Slag - electrically conductive type ARCOS BV-1 Vertomax or equivalent; pool depth 1 to 2 in.

B. Current - 60 cycle ac; 500 to 620 amp.

C. Voltage - 44 to 50 V.

D. Feed rate - 35 lb/hr; 1/8-in. single wire; 8 to 10 oscillations/min, nominal 2-in. oscillation.

Assembly of the pump casings was accomplished using a similar procedure, with identical welding parameters, but using two and three wires.

No electroslag welding is now being done at the Ginna plant and it is not anticipated that any will be done in the future.

1.8.2.15 Regulatory Guide 1.35 - Inservice Surveillance of Ungrouted Tendons in Prestressed Concrete Containment Structures

The tendon surveillance program for Ginna Station as required by the Technical Specifications is in accordance with Regulatory Guide 1.35, Revision 2. A detailed discussion of this inservice surveillance program is provided in Section 3.8.1.7.

1.8.2.16 Regulatory Guide 1.36, Revision 0 - Nonmetallic Thermal Insulation for Austenitic Stainless Steel

Although Regulatory Guide 1.36 had not been published before the completion of construction of the R. E. Ginna Nuclear Power Plant, the quality of the thermal insulation applied to austenitic stainless steel components was carefully specified and checked.

The practice employed during construction of the Ginna plant meets the requirements of Regulatory Guide 1.36 and is more stringent in several respects. The tests for qualification specified by the guide (ASTM C692-71 or RDT M12-1T) allow use of the tested insulation material if no more than one of the metallic test samples crack. Westinghouse procedure rejected the tested insulation material if any of the test samples cracked. The procedure followed for the R. E. Ginna Nuclear Power Plant was more specific than the procedures suggested by the guide, in that the Westinghouse specification required determination of leachable chloride and fluoride ions from a sample of the insulating material.

Experience has shown that of the three analysis methods allowed under ASTM D512 and ASTM D1179 for leachable chloride and fluoride, the referee method, which was used in the analysis of the Ginna insulation, is the most accurate and most suitable for nuclear applications.

Plant Operation

Insulating materials are not considered basic components pursuant to 10CFR21 and thus the supplier is not required to have a quality assurance program to cover the testing, lot control, and contamination control provisions of this Regulatory Guide. A quality assurance program similar to 10CFR50, Appendix B is applied to insulating materials on or near Ginna Station safety related stainless steel piping and components.

1.8.2.17 Regulatory Guide 1.37, Revision 0 - Quality Assurance for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants

The Ginna plant obtained its construction permit in April 1966. Regulatory Guide 1.37 and related ANSI Standard N45.2.1-1973 were published in 1973; therefore, these standards were not available during the construction phase of the Ginna plant. However, a formal program for the cleaning of the fluid components of the power plant was followed and documented.

The flushing water for the nuclear steam supply system met the following maximum water chemistry specifications: chlorides, maximum ppm -0.15; undissolved solids, maximum ppm -5.0; conductivity, maximum mhos/cm -5; pH -6.0 to 8.0; and visual clarity -no turbidity, oil, or sediment.

Pipe and units large enough to permit entry by personnel were cleaned by locally applying approved solvents (Stoddard solvent, acetone, and alcohol) and demineralized water. A line or equipment was considered clean when flush cloths showed no grindings, filings, or insoluble particulate matter larger than 40 microns (naked eye visibility lower limit) or oil stains visible to the naked eye. The final cleaned equipment was free of visible dust, grit, rust, weld splatter, scale, oil, grease, pickling solution residue, cleaning fluid film, or other foreign matter. Only iron-free aluminum, oxide grinders were used to remove trapped foreign particles.

The cleaning of the component cooling system was accomplished first by flushing separate lines to waste and, second, by flushing the complete system. Stainless steel strainers were installed and utilized during the second phase. The system was considered clean when no significant buildup was noted on the strainers. The demineralized water used met the same water chemistry specifications as the nuclear steam supply system flushing water and was treated with 100 ppm hydrazine for oxygen control.

For the secondary plant, the condensate and feedwater system was cleaned by manual cleaning of condenser surfaces and hotwells, cold water flush, and alkaline cleaning. The main steam system cleaning procedures included manual cleaning, cold water flush, alkaline cleaning, and acid cleaning.

These examples indicate the concern for system cleanliness during construction of the R. E. Ginna Nuclear Power Plant, even before the existence of the current guidelines.

Plant Operation

For new construction activities, the cleanliness requirements of ANSI N45.2.1-1973 as modified by the Regulatory Guide are followed. Consistent with Position C.2 of the Regulatory Guide, the cleanliness requirements of this standard are used when applicable to maintenance on operating systems. The cleanliness requirements applied to operational systems are established in station procedures.

1.8.2.18 Regulatory Guide 1.38, Revision 2 - Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants

Regulatory Guide 1.38 and related ANSI Standard N45.2.2-1972, were published after the construction of the R. E. Ginna Nuclear Power Plant.

However, each piece of equipment has detailed equipment specifications. The detailed requirements for preparation of equipment for shipment were included in the equipment specifications. These included sealing of all openings, protection of nozzle preparations, the use of dessicants if required, etc. Where required, the suppliers submitted detailed plans for review and approval.

For example, the reactor vessel supplier provided a cover and seal system to protect all internal surfaces and external stainless steel and machined surfaces from exposure to ambient environments during shipment, storage at the site, and installation. The protective means included pressurized inert gas with covers.

For the reactor internals, the lower assembly was shipped on an up-ending skid, shock-mounted to limit loads transmitted to the assembly during shipment. Prior to installation onto the skid, the lower internals were wrapped in a plastic film and sealed. Internal bracing was used inside the assembly. The upper internal assembly was shipped in a shock-mounted, dual-purpose shipping assembly stand in the vertical position. This package was also wrapped and sealed in a plastic film. Both the skid and the stand had a protective metal covering to provide weather protection and long-term storage protection at the site. All other components had similar protection, as required, against mechanical or environmental damage during shipment and/or site storage.

These detailed examples indicate the concern for components during transportation and handling.

Plant Operation

Ginna currently maintains conformance with this Regulatory Guide.

1.8.2.19 Regulatory Guide 1.39 - Housekeeping Requirements for Water-Cooled Nuclear Power Plants

The housekeeping awareness was generally followed for quality assurance jobs at Ginna Station. This was generally handled through precautions listed in maintenance, repair, and modifications procedures and also through quality control inspection and surveillance. Additional quality assurance information is provided in Chapter 17.

Plant Operation

Operational commitments to this Regulatory Guide are discussed in detail in the Quality Assurance Topical Report (QATR). The QATR is cited in Section 17.2 of the UFSAR and is submitted to the NRC in accordance with the provisions of 10 CFR 50.54(a).

1.8.2.20 Regulatory Guide 1.40 - Qualification Tests of Continuous-Duty Motors Installed Inside the Containment of Water-Cooled Nuclear Power Plants

Conformance to IEEE Standard 334-1971 is fully discussed in Section 1.8.3.

The containment recirculation fan cooler (CRFC) and filtration system fan motors are the only continuous-duty Class 1E motors within the containment. Environmental qualification is discussed in Section 3.11.

1.8.2.21 Regulatory Guide 1.41 - Preoperational Testing of Redundant Onsite Electric Power Systems to Verify Proper Load Group Assignments

This Regulatory Guide describes an acceptable method for verifying power load group assignments for onsite emergency power systems described in Regulatory Guides 1.6 and 1.32. Regulatory Guide 1.6 is discussed in Section 1.8.1.6. Regulatory Guide 1.32 is discussed in Section 1.8.2.10. The underlying standard, IEEE Standard 308-1971, is discussed in Section 1.8.3. Initial startup tests are discussed in Chapter 14.

The capability of adequately supplying the demand of the safeguards bus load groupings was preoperationally demonstrated. Buses 14 and 18 comprise one redundant safeguards train and buses 16 and 17 comprise the other. The two trains were isolated from each other and from offsite power sources. One diesel was started and the timing sequence for starting of all associated equipment was checked against design. The test was repeated for the other diesel. It was particularly important to test the diesels separately since one of the high-head safety injection pumps is designed to operate from either diesel generator, switching to an operating generator if one is not operating. Tests were continued for a sufficient time to guarantee proper starting sequence. The plant auxiliary startup transformer was also used as a power source. All equipment was monitored during the tests.

1.8.2.22 Regulatory Guide 1.42 - Interim Licensing Policy on As Low As Practicable for Gaseous Radioiodine Releases from Light-Water-Cooled Nuclear Power Reactors

Ginna Station is meeting as-low-as-practicable releases for gaseous iodine by the use of charcoal filters on all exhaust air from restricted areas. As a check on the efficiency of the charcoal filter system, all plant vent exhaust air is continually monitored for iodine. A further check is made by monthly analysis of samples of milk taken from nearby dairy herds. These three systems of control are referred to in the Offsite Dose Calculation Manual (ODCM).

In the initial design and construction of Ginna Station, all air purged from the containment vessel passed through high efficiency particulate air and charcoal filters. There was the further option of using a recirculating high efficiency particulate air and charcoal filter system within the containment. Air from high activity areas of the auxiliary building passed through charcoal and all air from restricted areas passed through high efficiency particulate air filters.

Prior to the first spent fuel handling in 1971, a bank of charcoal beds was installed to filter the air from the spent fuel pool (SFP) area. A charcoal filter was also added to the laboratory exhaust air system in 1971. In June 1972, another charcoal and high efficiency particulate air unit was added to filter iodine from the remaining auxiliary building air.

These filter systems are periodically tested for efficiency of operation. A leak test using Freon is done in the plant according to the Ventilation Filter Testing Program schedule and the efficiency of the activated charcoal adsorber is determined by an independent laboratory.

Both the plant vent and the containment vent have an iodine sampler with continuous monitoring. The monitor is read out and recorded in the control room and is programmed to alarm at a fraction of the release limit value calculated by methods described in the Offsite Dose Calculation Manual (ODCM). Action can then be taken, using the appropriate procedure, to meet the 24-hour limit allowed by the ODCM.

Thus the Ginna plant can be shown to meet the guidelines of Regulatory Guide 1.42 on an analytical basis and, in fact, several years of operations confirm this conclusion. Subsequently, 10 CFR 50, Appendix I, was published. Ginna LLC conforms to 10 CFR 50, Appendix I, as described in the Technical Specifications.

1.8.2.23 Regulatory Guide 1.43 - Control of Stainless Steel Weld Cladding of Low-Alloy Steel Components

The R. E. Ginna Nuclear Power Plant reactor vessel and pressurizer carbon steel surfaces in contact with primary coolant were clad with stainless steel type 304 equivalent weld deposit. For the replacement steam generators all ferritic steel surfaces in contact with the primary coolant are clad with weld deposited austenitic stainless steel (Types 308L and 309L) or Alloy 600. These ferritic base steels are either SA-508 Cl 3 or SA-533 Type B Cl 1 procured to fine grain practice and are not considered susceptible to underclad cracking. The Ginna Nuclear Power Plant reactor vessel shell and nozzle forgings were fabricated from SA-508

Class 2 material. However, these surfaces were stainless steel weld clad only by single-wire low energy input weld processes, which are not restricted by Regulatory Guide 1.43. The Ginna pressurizer SA-302 grade B plate and SA-216 WCC casting surfaces in contact with primary coolant were clad with weld deposited stainless steel. These base materials are not restricted by the requirements of Regulatory Guide 1.43.

Underclad cracking is not expected for the Ginna plant stainless steel weld clad components. Of those components clad only the reactor vessel shell and nozzle forgings are SA-508 Class 2 base material. All of the welding processes used to clad components in contact with primary coolant are single-wire low energy input processes.

No stainless steel weld cladding of low-alloy steel components is now being done at the Ginna plant, and it is not anticipated that any will be done in the future.

1.8.2.24 Regulatory Guide 1.44 - Control of the Use of Sensitized Stainless Steel

Regulatory Guide 1.44 was published after the construction of the R. E. Ginna Nuclear Power Plant. However, the R. E. Ginna Nuclear Power Plant meets the intent of Regulatory Guide 1.44.

All austenitic stainless steel materials used in the fabrication, installation, and testing of nuclear steam supply components and systems were handled, protected, stored, and cleaned according to recognized and accepted contemporary methods and techniques. To ensure that these methods and techniques were followed, surveillance of operations was conducted by Quality Assurance personnel of the applicant and the nuclear steam supply system supplier. Stainless steel material from which components were fabricated were procured in the solution heat-treated condition as required by the ASME Section II materials specifications.

Methods and materials used in manufacturing stainless steel components of the Ginna reactor coolant pressure boundary are described in detail in a letter dated October 6, 1970, from Edward J. Nelson, RG&E, to Peter A. Morris, AEC (Docket No. 50-244).

For internals where austenitic stainless steel was given a stress relieving treatment above 800°F, a high-temperature solution heat treatment procedure was used. This was performed in the temperature range of 1600°F to 1900°F with sufficient holding times.

For core support structural load bearing members and stainless steel reactor coolant pressure boundary welds, all welding on stainless steel was conducted by procedures that limit the interpass temperature to 350°F maximum. All of the reactor vessel and pressurizer nozzles, as well as the reactor vessel control rod drive mechanism adapters[2] and reactor vessel head

[2] The control rod drive mechanism (CRDM) adapters on the replacement reactor vessel closure head (RVCH) were fabricated from SA-182 Type F304LN stainless steel forgings also supplied in the solution heat-treated condition (annealed at 1950°F ± 25°F and water quenched). In addition, one sample from each heat of material used for the adapters, was given a simulated postweld heat-treatment (i.e., exposed to a temperature on the sensitizing range (1250°F ± 25°F) for 20 hours) and tested in accordance with ASTM A262, Practice E to verify the absence of sensitization. Therefore, postweld stress relief heat-treatment was not required on the adapters after welding, and consequently no potential for sensitization exists. The metallurgical condition of the adapters in the replacement RVCH is therefore superior to that of the adapters in the original head.

gasket monitor tubes, were postweld stress relief heat-treated for the minimum practical time (3 hours to 11 hours depending on size) at 1125°F ± 25°F. However, the reactor vessel primary coolant nozzles' weld deposits are calculated to contain at least 5% ferrite according to the Schaeffler Diagram. Thus, a duplex (austenite plus ferrite) structure can be expected in the safe ends of these nozzles. The guide recognizes that weld metal with duplex structures have demonstrated adequate resistance to intergranular attack. Although the remainder of the items listed above underwent a process which could result in sensitization, Westinghouse technical background and service experience, as detailed in Westinghouse topical reports, (*Reference 3*) support the conclusion that serious intergranular attack of sensitized stainless steel is unlikely in Westinghouse PWR nuclear steam supply systems, since water chemistry and contamination are kept under control. Water chemistry control is discussed in Sections
5.2.3.2 and 9.3.4.

NOTE: The primary nozzles on the replacement steam generators are integrally forged with the head. Nozzle safe ends are stainless steel forgings welded to Inconel buttering on the ends of the primary nozzles. Thus, the nozzles are not exposed to post weld heat treat temperatures.

In addition, as part of the procedures of the nuclear steam supply system supplier and RG&E, all safe ends were dye penetrant inspected after shop fabrication prior to shipping to the site and were subsequently reinspected upon completion of installation welds at the site. Also, all of the reactor coolant pressure boundary installation welds, including safe ends, were reinspected by dye penetrant upon completion of hydro and hot functional testing. No evidence of discontinuities associated with corrosion were found. Ginna LLC has and will continue to check stainless steel welds according to the inservice inspection program.

Plant Operation

Regulatory Guide 1.44 is now being used as a guide for handling, storing, and the fabrication of all stainless steel material. All welding and related activities are controlled to ensure that the chemical composition of the stainless steel is not affected. When welding is being done, the interpass temperature is maintained below 350°F to ensure the stainless steel will not become sensitized. This temperature is checked using temperature level devices during the welding fabrication process.

1.8.2.25 Regulatory Guide 1.45 -Reactor Coolant Pressure Boundary Leakage Detection System

Methods for detecting leakage from the reactor coolant system boundary are discussed in Section 5.2.5. Two radiation sensitive instruments provide the capability for detection of leakage: the containment air particulate monitor (R-11) and the less sensitive containment radiogas monitor (R-12). Additional monitors include the coolant inventory indication, containment sump A level indication (LT-2039 and LT-2044), sump A pump actuation indication, humidity detector, the condensate measuring system, and others.

Leakage from the reactor coolant system to the component cooling system would be reflected in an increase in the makeup water flow rate but not by the leakage monitors described previously. The radiation monitor in the component cooling system would annunciate in the control room and would initiate closure of the vent line from the surge tank in the component cooling system in the event of leakage to this system.

Sensitivities of some of the systems are discussed in detail in Section 5.2.5.

Airborne radioactivity monitors alarm in the control room. Each actuation of the containment sump pump causes an alarm in the control room. Each time makeup water is added to the primary system, an alarm is sounded in the control room. The time and amount of makeup is logged by the operators.

Calibration is performed on systems at specified frequencies.

The Technical Specifications present in detail leakage limits, instrument sensitivities and limitations on instruments out of service.

1.8.2.26 Regulatory Guide 1.46 - Protection Against Pipe Whip Inside Containment

The reactor vessel, steam generators, reactor coolant pumps, and pressurizer are supported to ensure that a postulated rupture of the main reactor coolant piping does not propagate into failures of connected safety-related systems, such as the Emergency Core Cooling System (ECCS) and secondary systems. Barriers are also provided to minimize the potential for pipe whip and jet impingement.

Additional information concerning protection against dynamic effects due to postulated pipe failures in Ginna Station is provided in Section 3.6.

1.8.2.27 Regulatory Guide 1.47 - Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems

Regulatory Guide 1.47 and the related IEEE Standard 279-1971 were published after the construction of the Ginna plant. The IEEE Standard is, however, discussed in Section 1.8.3.

Bypassing or defeating any portion of a protective channel results in an alarm in the control room indicating the channel affected.

1.8.2.28 Regulatory Guide 1.48 - Design Limits and Loading Combinations for Seismic Category I Fluid System Components

The Ginna Nuclear Power Plant equipment was designed and analyzed to ensure structural integrity and operability. However, Regulatory Guide 1.48 had not been published at the time of the Ginna Station design and construction. The codes and procedures employed in the Ginna design have been widely used and proven adequate by the nuclear industry for the design of components in operating plants.

The valves were designed to function at normal operating conditions, maximum design conditions, and earthquake conditions per the detailed equipment specifications. The requirements of the ANSI B31.1, ANSI B16.5, and MSS-SP-66 codes were adhered to in the design. The allowable stresses in the above codes are considerably less than the limits presently proposed by the ASME Task Group on Design Criteria for Class 2 and 3 Components, e.g., the allowable stress in ANSI B16.5 is 7000 psi as opposed to the maximum limit accepted by the ASME task group of 2.4 times the ASME Section VIII allowable stress.

Prior to shipment, the valves were subjected to hydrostatic leak tests in accordance with MSS-SP-61 and functional tests to show that the valves will open and close within the specified time limits when subjected to the design differential pressure. In addition, representative valves were checked for wall thickness to ANSI B16.5 and MSS-SP-66 requirements and subjected to nondestructive tests in accordance with ASME and ASTM codes. After installation of the valves they were subject to cold hydrostatic tests and hot functional tests to verify operation. Also, periodic inservice inspections and operation tests are performed as required.

Active pumps were designed to the requirements of the Standards of the Hydraulic Institute and/or the ASME Code for Pumps and Valves for Nuclear Power, depending on the pumps purchase order date. In addition, the pumps and their supports were designed to withstand horizontal and vertical earthquake forces.

The pumps were hydrostatically tested to 1.5 times the design pressure and were subjected to ASME Section VIII nondestructive tests. Performance tests were conducted to check the capacity, total dynamic head or pressure, and net positive suction head. After the pumps were installed in the plant, they were subjected to cold hydrostatic tests and hot functional tests to verify operation. Also, periodic inservice inspections and operation tests are performed as required.

Additional information is provided in Section 3.9 and in the specific sections of the UFSAR applicable to the fluid system components.

1.8.2.29 Regulatory Guide 1.49 -Power Levels of Water-Cooled Nuclear Power Plants

The R. E. Ginna Nuclear Power Plant is licensed to operate at 1775 MWt, the maximum calculated turbine thermal power. This is less than the guideline of 3800 MWt.

1.8.2.30 Regulatory Guide 1.50 -Control of Preheat Temperature for Welding of Low-Alloy Steel

Regulatory Guide 1.50 was published after the construction of the Ginna Nuclear Power Plant. However, the Westinghouse practice for the Ginna plant was in agreement with the requirements of Regulatory Guide 1.50, except for Regulatory Position 1(b) and 2.

In the case of Regulatory Position 1(b), the welding procedures were qualified within the preheat temperature ranges required by Section IX of the ASME Code. High quality qualification welds were obtained using the ASME qualification procedures.

In the case of Regulatory Position 2, the Ginna pressurizer and steam generators were fabricated without maintaining the preheat temperature until the postweld heat treatment had been performed. However, for the replacement steam generators, either the maximum interpass temperature is maintained four hours or the minimum preheat temperature is maintained eight hours after welding. Additionally, as required by Regulatory Position 2, the soundness of the welds is verified by an acceptable examination procedure appropriate to the weld under consideration.

In the case of the Ginna reactor vessel main structural welds, the practice of maintaining preheat until the intermediate or final postweld heat treatment was followed by the fabricator. For each of the above components, the qualification welds have shown high integrity, using the ASME Boiler and Pressure Vessel Code criteria. In all cases the welding parameters specified in the procedure were closely monitored during production welding.

Regulatory Position 4 of the guide was met for the Ginna plant in that, for ASME Section III Class 1 components, the examination procedures required by Section III and the inservice inspection requirements of Section XI were met.

Plant Operation

The recommended practice of Regulatory Guide 1.50 is followed in the format of the welding procedures used at Ginna Station. Welding procedures are designed according to the criteria outlined in Section III and Section IX of the ASME Boiler and Pressure Vessel Code. All welding procedures are qualified following the preheat, interpass temperature, and heat treatment outlined in the procedure. Production welds are controlled to ensure that the welding procedures, variables, and requirements are carried out properly.

1.8.2.31 Regulatory Guide 1.51 - Inservice Inspection of ASME Code Class 2 and 3 Nuclear Power Plant Components

The original 5-year inservice inspection program, as defined in the Technical Specifications at that time, was developed before ASME Section XI was issued. This program addressed Class 1 components only and completed its first 5-year cycle at the Spring 1974 MODE 6 (Refueling) outage. As a result of pipe whip considerations, some of the Class 2 requirements for main steam and main feedwater were fulfilled during the 1974 MODE 6 (Refueling) outage.

Following the 1974 outage, the inservice inspection program was revised to meet the new Section XI of the ASME Code and Regulatory Guide 1.51 requirements for Class 1, Class 2, and Class 3 Nuclear Plant Components.

1.8.2.32 Regulatory Guide 1.52 - Design, Testing, and Maintenance Criteria for Atmospheric Cleanup System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants

Ginna Station was designed in conformance with the General Design Criteria in effect in 1968. The atmosphere cleanup systems were designed under the applicable criteria (i.e., 41, 52, 58, 59, 61, 62, 63, 64, 65, 70). This is discussed in Sections 9.4.1, 6.2.2, and 6.5.1. The cleanup system was designed to operate under the environmental conditions resulting from a postulated design-basis accident. All components of the cleanup system are compatible with other engineered safety features and have been designed to be consistent with radiation fields and isotopes expected during the design-basis accident. There are no components of systems in unheated compartments. Charcoal filter units are provided with spray systems to limit adsorber fires.

All cleanup systems are designed for ease of maintenance and ready removal of elements. Lighting is provided in the housings and test probe holes for in-place testing are available.

Filter units were tested prior to startup of Ginna Station and are retested according to the schedules of the Ventilation Filter Testing Program. These tests are subcontracted to a reliable vendor who prepares the report of test results. Samples from the charcoal filter trays are sent for organic iodides and elemental iodine efficiency tests according to the Ventilation Filter Test Program (ITS 5.5.10).

1.8.2.33 Regulatory Guide 1.53 - Application of the Single-Failure Criterion To Nuclear Power Plant Protection Systems

This guide endorses the use of IEEE Standard 379-1972, Trial-Use Guide for the Application of the Single-Failure Criterion to Nuclear Power Generating Station Protection Systems. Subjects which are covered in the standard include identification of undetectable failures, analysis of channel interconnections for failures which could compromise independence, testing to determine independence between redundant parts of the protection system, and analysis to show that no single failure can cause a loss of function due to improper connection of actuators to a power source.

Routing and separation standards applicable to existing cables are those that were invoked at the time of cable installation. For more information, see Section 8.3.1.4.

Protection system failure analyses and reliability studies applicable to the Ginna plant were performed as described in the topical report WCAP 7486-L, December 1970, An Evaluation of Anticipated Operational Transients in Westinghouse Pressurized Water Reactors. This report was submitted to the AEC by Westinghouse in March 1971. Subsequent evaluations have demonstrated the conformance of the Ginna Station design to this guide.

1.8.2.34 Regulatory Guide 1.54, Revision 0 - Quality Assurance Requirements for Protective Coatings Applied to Water-Cooled Nuclear Power Plants

Contemporary standards were specified to ensure that protective coatings applied would perform their functions under environmental conditions experienced during MODES 1 and 2 and the design-basis accident and to do so without hazard of interfering with other nuclear components.

One standard specified was SP-5485 dated January 18, 1968, entitled Technical Specification, Painting of Structures and Equipment, Robert Emmett Ginna Nuclear Power Plant Unit No. 1, which includes techniques for preparation of surfaces to be painted, sampling, thickness measurement and control, and a detailed paint schedule including components and paint materials for plant structures and equipment. Also, SP-5339 dated March 31, 1967, entitled Technical Specification for Painting the Interior Surface of the Containment Vessel Dome for the Robert Emmett Ginna Nuclear Power Plant Unit No. 1, gives the specifications for the preparation, application, material, and paint sampling for the interior of the containment dome.

The painting of the containment structure and components inside the containment was governed by Westinghouse process specification PWR 597755, dated February 20, 1968. This specification covered the application of paint systems to equipment and structures in containments which use additive spray systems for fission product removal and/or containment cooling.

Regulatory Guide 1.54 and related ANSI Standard N101.4 were published after construction of the Ginna plant and thus were not available to be applied. However, the previously referenced process specifications demonstrate that care was taken in the selection and application of protective coatings for the Ginna plant.

Plant Operation

For new coatings and configuration changes to existing coatings, which have the potential to adversely affect a safety related function, the quality assurance requirements of 10CFR50, Appendix B, in conjunction with engineering specifications, are used instead of the detailed requirements included in this Regulatory Guide and its referenced standard, ANSI N101.4-1972.

1.8.2.35 Regulatory Guide 1.55 -Concrete Placement in Seismic Category I Structures

All concrete placement for the Ginna plant was accomplished in accordance with the proposed specification for structural concrete for buildings ACI-301 and the detailed construction specification.

In accordance with the specification, the contractor submitted placing drawings, reinforcing bar details, and bar lists, etc., for engineer approval to ensure that the details were in general compliance with the engineering drawings. Construction joints not shown on the drawings were located in accordance with the requirement of the specification and only after their influence on the structural integrity was reviewed and approved in writing by the engineer. Field generated revisions were reviewed and approved by the engineer.

The services of Pittsburgh Testing Laboratory were obtained to ensure the quality control on the job. Well before the concrete work started, representative samples of ingredients for the concrete work were tested and concrete mix design was established to conform to the design requirements. During concrete operation, the Testing Laboratory had an inspector at the batch plant who certified the mix proportions of each batch delivered to the site, took samples of the concrete ingredients, and tested them periodically. Another inspector was stationed at the construction site who inspected rebar, form placements, took slump tests, made test cylinders, checked air content, and recorded weather conditions. Cylinder tests were made in accordance with the provision of the ACI Code.

1.8.2.36 Regulatory Guide 1.57 -Design Limits and Loading Combinations for Metal Primary Reactor Containment System Components

The Ginna containment is a composite structure as opposed to a metal primary reactor containment; thus this guide is not applicable.

1.8.2.37 Regulatory Guide 1.59 -Design-Basis Floods for Nuclear Power Plants

The R. E. Ginna Nuclear Power Plant site has been evaluated for the probable maximum flood coincident with wind and wave activity as outlined in Section 2.4.

The analysis for flood, storm, waves, and hardened protection is generally consistent with Regulatory Guide 1.59. Site Contingency Procedures are available to be implemented in the event of potential flooding conditions. A recent review of Ginna flood protection measures described the conformance of Ginna Station to this guide.

1.8.2.38 Regulatory Guide 1.94, Revision 1 - Quality Assurance Installation, Inspections, and Testing of Structural Concrete and Structural Steel During the Construction Phase of Nuclear Power Plants

This Regulatory Guide applies to plants in the construction phase and was issued after Ginna was built. The specific details of the Ginna controls during construction are discussed in Section 17.1.

1.8.2.39 Regulatory Guide 1.143, Revision 1 - Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants

The specific UFSAR sections discuss the design and quality assurance provisions applied to existing radioactive waste management systems, structures, and components. New systems, structures, and components and configuration changes to existing items meet the design and quality assurance provisions described in the UFSAR sections or those specified by this Regulatory Guide.

1.8.3 CONFORMANCE TO IEEE CRITERIA

The information in this section is generally that submitted in the August 1972 Technical Supplement Accompanying the Application for a Full-Term Operating License as to the adequacy of the R. E. Ginna Nuclear Power Plant design with respect to IEEE Standards 279-1971, 308-1971, 317-1971, 323-1971, 334-1971, 336-1971, 338-1971, and 344-1971.

1.8.3.1 Criteria for Protection Systems for Nuclear Power Generating Stations (IEEE 279-1971)

Conformance with IEEE 279-1971 is discussed in Section 7.1.2.

1.8.3.2 Class 1E Electric Systems for Nuclear Power Generating Stations (IEEE 308-1971)

1.8.3.2.1 Principal Design Criteria

The criteria states that Class 1E electric systems shall be designed to ensure that any design-basis event as listed in Table 1 of the standard will not cause a loss of electric power to a number of engineered safety features, surveillance devices, or protection system devices sufficient to jeopardize the safety of the station. The design-basis events include earthquakes, winds, tornadoes, other natural phenomena, and various postulated accidents.

All electrical systems and components vital to plant safety, including the emergency diesel generators, are designed as Class 1E and are designed so that their integrity is not impaired by the design-basis earthquake, wind storms, floods, or disturbances on the external electrical system. Power, control and instrument cabling, motors, and other electrical equipment required for operation of the engineered safety features are suitably protected against the effects of either a nuclear system accident or of severe external environmental phenomena in order to ensure a high degree of confidence in the operability of such components in the event that their use is required.

The preferred power supply (offsite power) has a voltage variation of not more than plus or minus 10% and a frequency variation of not more than plus or minus 0.5%. Variations of voltage and frequency of the standby power supply (diesel generators) will not degrade the performance of any load to the extent of causing significant damage to the fuel or to the reactor coolant system.

Controls and indicators are provided in the control room and locally for the standby power supply and for the circuit breakers required to switch the Class 1E buses between the preferred and standby power supply. Transfer is automatic on loss of the preferred supply.

All components of the Class 1E electric systems are identified with permanently installed equipment piece-number tags. Design, operating, and maintenance documents for each major component were identified as they were received from the equipment suppliers, and the identification associates each component with its particular system.

Class 1E electrical equipment is physically separated to the extent practical from its redundant counterpart either by distance, barrier walls, or by location on different floors.

Each type of Class 1E electric equipment was designed, manufactured, and tested in accordance with the latest standards in existence at the time of manufacture. This equipment was analyzed to ensure that it would successfully perform its function under normal and design-basis events. In addition to this, preoperational testing was performed to verify equipment operation.

Failure mode analyses have been done for all Class 1E electrical systems. These analyses show that a single component failure does not prevent satisfactory performance of the Class 1E systems required for safe shutdown and maintenance of post-shutdown or postaccident station security.

The Class 1E electric systems are described in detail in Chapter 8. The systems consist of an ac power system, a dc power system, and an instrumentation and a control system to supply acceptable power to the station for any design-basis event.

1.8.3.2.2 Alternating Current Power Systems

1.8.3.2.2.1 General

The ac power systems include power supplies, distribution systems, and load groups arranged to provide ac electric power to the Class 1E loads. Sufficient physical separation, electrical isolation, and redundancy are provided to minimize the occurrence of a common failure mode in the Class 1E systems.

The Class 1E electric system is divided into two redundant load groups. Safety actions by each group of loads is redundant and independent of the safety actions provided by its redundant counterpart. Each load group has access to both the offsite and standby power supply.

Two independent 34.5-kV transmission lines make up the preferred offsite power supply and two independent diesel generators make up the standby power supply.

1.8.3.2.2.2 Distribution Systems

By design, each distribution circuit is capable of transmitting sufficient energy to start and operate all required loads in that circuit. Distribution circuits to redundant equipment are physically and electrically independent of each other, to the extent practical.

Auxiliary devices required to operate dependent equipment are supplied from related bus sections such that loss of electric power in one load group does not cause the loss of function of equipment in another load group. By means of circuit breakers located in the auxiliary building and the screen house (both Seismic Category I structures), it is possible to disconnect portions of the Class 1E system that are located in other than Seismic Category I structures. The distribution system is monitored to the extent that it is shown to be ready to perform its intended function. The surveillance program is included in the Technical Specifications.

1.8.3.2.2.3 Preferred Power Supply

The preferred power supply consists of two 34.5-kV circuits that are independent. This system is designed to furnish the starting and operating power requirements for the shutdown of the station and for the operation of emergency systems and engineered safety features. It also functions as startup power and reserve power for all unit auxiliaries.

A minimum of one circuit is available from the transmission network during MODES 1 and 2.

1.8.3.2.2.4 Standby Power Supply

The standby power supply provides power for the operation of emergency systems and engineered safety features during and following the shutdown of the reactor when the preferred power supply is not available.

The standby sources become available automatically following the loss of the preferred power supply within a time consistent with the requirements of the engineered safety features and the shutdown systems under normal and accident conditions. A failure of any unit of standby power source does not jeopardize the capability of the remaining standby power sources to start and run the required shutdown systems, emergency systems, and engineered safety features loads.

Two 6000 gallon underground storage tanks serve only the two emergency diesel generators. These tanks have the minimum required capacity of 10,000 gallons for 48 hours operation of both diesel generators at load, simultaneously, or one diesel generator at load for 80 hours. See Section 9.5.4 for an update of this historical information. The actual load on a diesel generator needed to place the station in a safe shutdown condition is less than the full-load rating of the diesel generator. This supply allows adequate time for makeup supplies of oil if required. The standby power supplies are started and operated at specified loads on a monthly basis. This program is included in the Technical Specifications.

1.8.3.2.3 Direct Current Power Systems

1.8.3.2.3.1 General

The dc power systems include power supplies, a distribution system, and load groups arranged to provide dc electric power to the Class 1E dc loads and for control and switching of the Class 1E systems. Sufficient physical separation, electrical isolation, and redundancy are provided to minimize the occurrence of common failure modes in the station Class 1E systems and include the following:

a. The electric loads are separated into two redundant load groups.

b. Safety actions by each group of loads are redundant and independent to the safety actions provided by its redundant counterpart.

c. Each redundant load group has access to a battery and two battery chargers.

These items are discussed in Chapter 8.

1.8.3.2.3.2 Distribution System

Each distribution circuit is capable of transmitting sufficient energy to start and operate all required loads connected to it. Distribution circuits to redundant equipment are independent of each other to the extent practical. Auxiliary devices required to operate dependent equipment are supplied from a related bus section to comply with this criterion. It is possible to disconnect portions of Class 1E systems located in Seismic Category I structures from those portions located in other than Seismic Category I structures. The disconnecting means are located in distribution panels in the Seismic Category I battery rooms. The system is monitored with indicators and alarms in the control room to the extent that it is shown to be ready to perform its intended function.

1.8.3.2.3.3 Battery Supply

Each battery supply consists of storage cells, connectors, and connections to the dc distribution system supply breaker. Each battery supply is independent of the other supply and is capable of starting and carrying all required loads. Each battery supply is immediately available during MODES 1 and 2 and following the loss of power from the ac system.

Each battery is kept fully charged and floating across its battery charger. Stored energy is sufficient to operate all necessary breakers to provide an adequate source of power for all connected loads. Battery instrumentation located in the control room indicates the status of the battery supplies.

1.8.3.2.3.4 Battery Charger Supply

The battery chargers provide all the dc power required for normal station operation as long as ac power is available. Each battery can be supplied by a full capacity charger or a full capacity backup charger. Each full capacity charger has sufficient capacity to restore the battery from the design minimum charge to its fully charged state while supplying normal steady-state loads. The two supplies are independent of each other. The capability for isolating each charger is provided by means of circuit breakers in the ac feeder and the dc output circuit.

1.8.3.2.3.5 Protective Devices

Protective devices are provided to isolate failed equipment automatically. Indication is also provided to identify the equipment that is made unavailable.

1.8.3.2.3.6 Performance Discharge Test Provisions

To be sure that all cells, connections, jumpers, etc., satisfactorily handle full-rated current if necessary, each battery has been tested under full load and each component individually examined.

1.8.3.2.4 Vital Instrumentation and Control Power Systems

Dependable power supplies are provided for the vital instrumentation and control systems of the unit including the following.

A. The nuclear plant protection instrumentation and control systems.

B. The engineered safety features instrumentation and control systems.

Power is supplied to these systems in such a manner as to preserve their reliability, independence, and redundancy.

1.8.3.2.5 Surveillance Requirements

Preoperational Equipment Tests and Inspection

The initial equipment tests and inspections were performed with all components installed. They demonstrated the following:

C. All components were correct and properly mounted.

D. All connections were correct and circuits were continuous.

E. All components were operational.

F. All metering and protective devices were properly calibrated and adjusted.

Initial System Test

The initial system test was performed with all components installed. The test demonstrated the following:

A. The Class 1E loads can operate properly on the preferred power supply.

B. The loss of the preferred power supply can be detected.

C. The standby power supply can be started automatically and can accept design load within the design-basis time.

D. The standby power supply is independent of the preferred power supply.

Periodic Tests

The periodic test programs are included in the Technical Specifications. Tests are performed at scheduled intervals to

A. Detect possible deterioration of the system toward an unacceptable condition.

B. Demonstrate that standby power equipment and other components that are not exercised during MODES 1 and 2 of the station are operable. If surveillance tests indicate that any Class 1E systems are degraded, the Technical Specifications impose operating limitations.

1.8.3.3 Electrical Penetration Assemblies in Containment Structures for Nuclear Fueled Power Generating Stations (IEEE 317 - April 1971)

Electrical penetrations are designed and demonstrated by test to withstand, without loss of leak tightness, the containment post-accident environment and meet the following guide that was available during construction: IEEE Proposed Guide for Electrical Penetration Assemblies in Containment Structures for Stationary Nuclear Power Reactors (Eighth Revision). The electrical penetration sleeves, being part of the containment vessel, were designed in accordance with the ASME Boiler and Pressure Vessel Code, Section III, Subsection B, for Class B vessels.

The penetration assemblies are qualified to prevent leakage from the containment under the worst-case environmental conditions associated with a loss-of-coolant accident or main steam line break.

All welded joints for the penetrations including the reinforcement about the openings are fully radiographed in accordance with the requirements of the ASME Nuclear Vessel Code for Class B Vessels except that non-radiographable joint details are examined by the liquid penetrant method. Verification of leak tightness is by means of pressurizing test channels.

There are generally five types of electrical cable penetrations required depending on the type of cable involved:

- Type 1 - High voltage power 4160 V.
- Type 2 - Power, control and instrumentation; 600 V and lower.
- Type 3 - Thermocouple leads.
- Type 4 - Coaxial and triaxial circuits.
- Type 5 - Fiber Optic

All five types of penetration designs are a cartridge type. The cartridge length and the supporting of cables immediately outside containment are designed to eliminate any cantilever stresses on the cartridge flange.

The specification for penetrations cover all aspects of equipment design, manufacture, inspection, qualification, and testing.

1.8.3.4 Qualifying Class I Electric Equipment for Nuclear Power Generating Stations (IEEE 323-April 1971)

The components of the protection system are designed and qualified so that the mechanical and thermal environment accompanying any emergency situation in which the components are required to function does not interfere with that function.

The equipment that must withstand the most severe environment is that which is in the containment. The instrumentation, motors, cables, and penetrations located inside containment are either protected from containment accident conditions or are designed to withstand, without failure, exposure to the worst combination of temperature, pressure, and humidity expected during the required operational period.

Quality standards of material selection, design, fabrication, and inspection governing the above features conformed to the applicable provisions of recognized codes and good nuclear practice.

1.8.3.5 Type Tests of Continuous Duty Class I Motors Installed Inside the Containment of Nuclear Power Generating Stations (IEEE 334-1971)

Of those motors installed within the containment of Ginna Station only the motors on valve operators and the fan motors of the containment air recirculation, cooling, and filtration system are required to be Class I. The valve motors, however, are not subjected to continuous duty. Therefore, IEEE 334-1971 does not apply to them.

The containment recirculation fan cooler (CRFC) and filtration system fan motors are continuous duty. The fans, motors, electrical connections, and all other equipment in the containment necessary for operation of the system are capable of operating under the environmental conditions following a loss-of-coolant accident. These environmental conditions are defined in Section 3.11.

All components are capable of withstanding or are protected from differential pressure which may occur during the rapid pressure rise to 60 psig in 10 seconds.

Any single active component failure in the system will not degrade the overall required heat removal capability.

Overload protection for the fan motors is provided at the switchgear by overcurrent trip devices in the motor feeder breakers. The fan motor feeder breakers can be operated from the control room and can be reclosed from the control room following a motor overload trip.

1.8.3.6 Installation, Inspection, and Testing Requirements for Instrumentation and Electric Equipment During the Construction of Nuclear Power Generating Stations (IEEE 336-1971)

An evaluation of prospective suppliers was conducted prior to awarding of a contract for important components. This evaluation established that the supplier has acceptable design, manufacturing, and quality control capability. The supplier was provided with individual equipment specifications covering all aspects of equipment design, manufacture, inspection, and testing. For Class 1E components, such as those in the reactor coolant system, a specification which defined the quality control requirements was made a part of each purchase order.

The instrumentation and electrical equipment for engineered safety features and reactor protection were subjected to receiving inspection, pre-installation operability and calibration checks, and preoperational functional and calibration tests. The quality assurance requirements during construction are described in Chapter 17; initial tests are described in Chapter 14.

1.8.3.7 Trial Use Criteria for the Periodic Testing of Nuclear Power Generating Station Protection Systems (IEEE 338-1971)

The station has the capability for sensor checks, channel tests, and channel calibration. The testing program is based on the calculations that were presented on the basis of the Technical Specifications.

All protective instrumentation has the capability of being tested and calibrated. Instrumentation that requires testing between reactor shutdowns also has the capability for being tested during MODES 1 and 2. The satisfactory operation of each redundant channel may be verified and credible failures can be detected. A scheduled test program is presented in the Technical Specifications.

All sensor checks and tests are either done by perturbing the monitored variable, introducing a substitute input, or comparing sensors which measure like variables. The test signal amplitude is varied to determine that the protective action will occur when the setpoint is reached. These setpoints include the effects of instrumentation errors.

Written procedures are maintained for all tests. The results are documented and records are kept.

1.8.3.8 Seismic Qualification of Class I Electrical Equipment for Nuclear Power Generating Stations (IEEE 344-1971)

All systems and components designated Class I are designed so that there is no loss of function in the event of the design-basis earthquake ground acceleration acting in the horizontal and vertical directions simultaneously. Subsequent reviews of the qualification of this equipment is described in Section 3.10.

REFERENCES FOR SECTION 1.8

1. Ernest L. Robinson, Bursting Test of Steam-Turbine Disk Wheels, Transactions of ASME, July 1944.

2. Gilbert Associates, Inc., Structural Integrity Test of Reactor Containment Structure, GAI Report No. 1720, October 3, 1969.

3. Westinghouse Electric Corporation, Sensitized Stainless Steel in Westinghouse PWR Nuclear Steam Supply Systems, WCAP 7477-L, WCAP 7477-L Addendum 1, WCAP 7735, May 15, 1973.