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bcc: H. J. McAlduff, ORO
E. E. Hall, GMR/H
J. A. Harris, PI
J. R. Buchanan, ORNL
A. A. Wells, ASLB
S. Robinson, SECY
J. Verme, SNM
D. Nussbaumer, DML
J. Saltzman, SLR

Distribution:
AEC PDR
Docket
DR Reading
DRL Reading
RPB-5 Reading
R. S. Boyd
G. Hadlock, OGC
L. Kornblith, CO (2)
N. Dube (w/3 encl.)
D. Skovholt
P. Collins
S. M. Kari (2)

Docket No. 50-244

Rochester Gas and Electric
Corporation
89 East Avenue
Rochester, New York 14604

Attention: Mr. Francis E. Drake, Jr.
Chairman of the Board

Gentlemen:

The Atomic Energy Commission has forwarded to the Office of the Federal Register for filing and publication a notice relating to the proposed issuance of a provisional operating license to Rochester Gas and Electric Corporation for operation of the Robert Emmett Ginna Unit No. 1 reactor. A copy of the notice is enclosed.

The license would authorize Rochester Gas and Electric to operate the Ginna reactor at thermal power levels up to five megawatts. Following our review of (1) the Class I piping analysis and (2) the results of the research and development programs on heat transfer tests of the cooler tubes, fan motor tests, and process instrument transmitters, Rochester Gas and Electric would receive written notification when operation at 1300 megawatts would be allowed in accordance with the provisions of the license and the Technical Specifications. If these items can be completed during the thirty-days waiting period, the license would be issued to authorize operation at the full power of 1300 megawatts thermal.

Copies of the Technical Specifications and the Safety Evaluation referred to in the notice are also enclosed for your use.

Sincerely,

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Enclosures:
1. Federal Register Notice
2. Technical Specifications
3. Safety Evaluation

See Seperate Jacket

cc: Arvin E. Upton, Esquire
LeBoeuf, Lamb, Leiby & MacRae

OFFICE	DRL/RPB-5	OGC	DRL/RP	DRL
SURNAME	SMKari:emh x7791	[signature]	~~DLKnuth~~ RSBoyd	PAMorris
DATE	6/11/69	6/[illegible]/69	6/13/69	6/13/69

UNITED STATES ATOMIC ENERGY COMMISSION

DOCKET NO. 50-244

ROCHESTER GAS AND ELECTRIC CORPORATION

NOTICE OF PROPOSED ISSUANCE OF PROVISIONAL OPERATING LICENSE

Notice is hereby given that the Atomic Energy Commission (the Commission) is considering the issuance of a provisional operating license, set forth below, which would authorize Rochester Gas and Electric Corporation (RG&E) to possess, use, and operate the Robert Emmett Ginna Nuclear Power Plant Unit No. 1, a pressurized, light water moderated, and cooled reactor. The reactor is located at RG&E's Brookwood site, in Wayne County, New York about 16 miles east of the City of Rochester. The reactor is designed to operate at 1300 megawatts thermal; however, until the Commission has reviewed 1) the Class I piping analysis and 2) the results of the research and development programs on heat transfer tests of the cooler tubes, fan motor tests, and process instrument transmitters, the power level will be restricted to 5 megawatts thermal. Upon completion of the above items and upon written notification from the Commission, operation at 1300 megawatts will be allowed in accordance with the provisions of the license and the Technical Specifications appended thereto.

Prior to issuance of the provisional operating license, the facility will be inspected by the Commission to determine whether it has been constructed in accordance with the application, as amended, and the provisions of Construction Permit No. CPPR-19, issued by the Commission on April 25, 1966, as amended by the Commission on November 12, 1966, and April 10, 1967. Upon issuance of the provisional operating license, RG&E will be required to execute an indemnity agreement as required by Section 170 of the Atomic Energy Act of 1954, as amended, and 10 CFR Part 140 of the Commission's regulations.

Within thirty (30) days from the date of publication of this notice in the FEDERAL REGISTER, the applicant may file a request for a hearing, and any person whose interest may be affected by this proceeding may file a petition for leave to intervene. Requests for a hearing and petitions to intervene shall be filed in accordance with the Commission's regulation (10 CFR Part 2). If a request for a hearing or a petition for leave to intervene is filed within the time prescribed in this notice, the Commission will issue a notice of hearing or an appropriate order.

For further details with respect to this proposed provisional operating license, see (1) the application for provisional operating license (Amendments No. 6 through 19) filed during the period of January 18, 1968, through April 16, 1969, (2) the report of the Advisory Committee on Reactor Safeguards, dated May 15, 1969, (3) a related safety evaluation prepared by the Division of Reactor Licensing, (4) the Technical Specifications which are incorporated in the proposed license and designated as Appendix A thereto, and (5) the Special Nuclear Materials Transfer Schedule, designated as Appendix B to the license, all of which will be available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D. C. Copies of items

(2) and (3) above may be obtained at the Commission's Public Document Room or upon request addressed to the Atomic Energy Commission, Washington, D. C. 20545, Attention: Director, Division of Reactor Licensing.

FOR THE ATOMIC ENERGY COMMISSION

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Dated at Bethesda, Maryland
this day of June, 1969.

JUN 13 1969

ROCHESTER GAS AND ELECTRIC CORPORATION

DOCKET NO. 50-244

PROPOSED PROVISIONAL OPERATING LICENSE

The Atomic Energy Commission (the Commission) having found that:

a. The application for provisional operating license (Amendments No. 6 through 19, dated January 18, 1968, April 9, 1968, September 23, 1968, September 30, 1968, October 10, 1968, October 16, 1968, December 2, 1968, December 6, 1968, Jaunuary 31, 1969, February 3, 1969, February 12, 1969, March 14, 1969, March 28, 1969, and April 16, 1969, respectively) complies with the requirements of the Atomic Energy Act of 1954, as amended, and the Commission's regulations set forth in Title 10, Chapter 1, CFR;

b. The facility has been constructed in accordance with the application, as amended, and the provisions of Provisional Construction Permit No. CPPR-19, as amended;

c. There are involved features, characteristics and components as to which it is desirable to obtain actual operating experience before the issuance of an operating license for the full term requested in the application;

d. There is reasonable assurance (i) that upon satisfactory completion of 1) the Class I piping analysis and 2) the research and development programs on heat transfer tests of the cooler tubes, fan motor tests, and process instrument transmitter tests that the facility can be operated at power levels not in excess of 1300 megawatts thermal in accordance with this license without endangering the health and safety of the public and (ii) that such activities will be conducted in compliance with the rules and regulations of the Commission;

e. The applicant is technically and financially qualified to engage in the activities authorized by this license, in accordance with the rules and regulations of the Commission;

f. The applicant has furnished proof of financial protection to satisfy the requirements of 10 CFR Part 140;

g. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the public;

Provisional Operating License No. DPR- is hereby issued to Rochester Gas and Electric Corporation (RG&E), as follows:

1. This license applies to the Robert Emmett Ginna Nuclear Power Plant Unit No. 1, a closed cycle, pressurized, light water moderated and cooled reactor, and electric generating equipment (the facility). The facility is located on the applicant's site on the south shore of Lake Ontario, Wayne County, New York, about 16 miles east of the City of Rochester, and is described in license application Amendment No. 6, "Final Facility Description and Safety Analysis Report," as supplemented and amended (Amendments No. 7 through 19).

2. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses RG&E:

A. Pursuant to Section 104b of the Atomic Energy Act of 1954, as amended, (the Act) and 10 CFR Part 50, "Licensing of Production and Utilization Facilities," to possess, use, and operate the facility as a utilization facility at the designated location on RG&E's Brookwood Site;

B. Pursuant to the Act and 10 CFR Part 70, "Special Nuclear Material," to receive, possess and use at any one time up to 2300 kilograms of contained uranium-235 in connection with operation of the facility;

C. Pursuant to the Act and 10 CFR Part 30, "Rules of General Applicability to Licensing of Byproduct Material," to receive, possess and use 72.6 microcuries of neptunium-237 composed of six sealed sources contained in irradiation surveillance capsules; to receive, possess and use 600 curies of polonium-beryllium contained in encapsulated form as primary source rods in neutron source assemblies; and to possess and use 65,000 curies of antimony-beryllium contained in encapsulated form as a secondary source; and

D. Pursuant to the Act, and Parts 30 and 70, to possess, but not to separate, such byproduct and special nuclear material as may be produced by operation of the facility.

3. This license shall be deemed to contain and is subject to the conditions specified in the following Commission regulations in 10 CFR Part 20, § 30.34 of Part 30, § 40.41 of Part 40, § 50.54 and 50.59 of Part 50, and § 70.32 of Part 70, and is subject to all applicable provisions of the Act and rules, regulations and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified below:

   A. Maximum Power Level

      RG&E is authorized to operate the facility at steady state power levels up to a maximum of 5 megawatts thermal until the items in section d are satisfactorily completed at which time operation at steady state power levels up to 1300 megawatts thermal will be authorized.

   B. Technical Specifications

      The Technical Specifications contained in Appendix A[1/] attached hereto are hereby incorporated in this license. RG&E shall operate the facility at power levels not in excess of 1300 megawatts thermal in accordance with the Technical Specifications, and may make changes therein only when authorized by the Commission in accordance with the provisions of § 50.59 of 10 CFR Part 50.

1/ This item was not filed with the Office of the Federal Register, but will be available for public inspection in the Public Document Room of the Atomic Energy Commission.

C. Reports

In addition to the reports otherwise required under this license and applicable regulations:

(1) RG&E shall inform the Commission of any incident or condition relating to the operation of the facility which prevented or could have prevented a nuclear system from performing its safety functions. For each such occurrence, RG&E shall promptly notify by telephone or telegram the appropriate Atomic Energy Commission Regional Office listed in Appendix D of 10 CFR Part 20, and shall submit within ten (10) days a report in writing to the Director, Division of Reactor Licensing (Director, DRL), with a copy to the Division of Compliance.

(2) RG&E shall report to the Director, DRL, in writing within thirty (30) days of its observed occurrence any substantial variance disclosed by operation of the facility from performance specifications contained in the Final Facility Description and Safety Analysis Report (safety analysis report) of the Technical Specifications.

(3) RG&E shall report to the Director, DRL, in writing within thirty (30) days of its occurrence any significant changes in transient or accident analysis as described in the safety analysis report.

(4) As soon as possible after the completion of six months of operation of the facility (calculated from the date of initial criticality), RG&E shall begin submitting reports in writing in accordance with the requirements of the Technical Specifications.

D. Records

RG&E shall keep facility operation records in accordance with the requirements of the Technical Specifications.

4. Pursuant to § 50.60 of 10 CFR Part 50, the Commission has allocated to RG&E for use in the operation of the facility 14,567 kilograms of uranium-235 contained in uranium in the isotopic ratios specified in the application. Estimated schedules of special nuclear material transfers to RG&E and returns to the Commission are contained in Appendix B [1/] which is attached hereto. Transfers by the Commission to RG&E in accordance with Column 2 in Appendix B will be conditional upon RG&E's return to the Commission of material substantially in accordance with Column 3 (including the subcolumns headed "Scrap" and "Depleted Fuel").

SEE 9/25/69 HR to RG&E

1/ This item was not filed with the Office of the Federal Register, but will be available for public inspection in the Public Document Room of the Atomic Energy Commission.

June 19, 1969

SAFETY EVALUATION

BY THE

DIVISION OF REACTOR LICENSING

U. S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

ROCHESTER GAS AND ELECTRIC CORPORATION

ROBERT EMMETT GINNA NUCLEAR POWER PLANT UNIT NO. 1

DOCKET NO. 50-244

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1.0 INTRODUCTION

The Rochester Gas and Electric Corporation (RG&E, applicant) submitted Amendment No. 6, dated January 18, 1968, to its application requesting a Provisional Operating License for the Robert Emmett Ginna Nuclear Power Plant Unit No. 1 (Ginna, facility). The facility, which will utilize a closed circulation Westinghouse pressurized water reactor (PWR), has been under construction since issuance of a construction permit on April 25, 1966. It is located on an 338-acre site in the township of Ontario, Wayne County, New York. This site is on the south shore of Lake Ontario, about 16 miles east of Rochester, and 40 miles west-southwest of Oswego, New York.

Our technical safety review of the design of the facility has been based on Amendments Nos. 6 through 19. All of these documents are available for review at the Atomic Energy Commission's Public Document Room at 1717 H Street, Washington, D. C. In the course of the review, we have held numerous meetings with the applicant to discuss and clarify the technical material submitted. In addition to our review, the Advisory Committee on Reactor Safeguards (ACRS) reviewed the application and met with both us and the applicant to discuss the facility. The ACRS report on Ginna, dated May 15, 1969, is attached to this safety evaluation.

Our evaluation of overall facility performance was based on a power level of 1300 megawatts thermal (Mwt), which will be the licensed power level. However, because the plant is designed for ultimate power operation at 1520 Mwt, we reviewed the capability of the plant engineered safety features and the radiological consequences of accidents for the ultimate power level of 1520 Mwt. Before operation at any power level in excess of 1300 Mwt will be permitted, the applicant must submit an application for license amendment for our review and approval.

Based upon our evaluation of the facility as presented in subsequent sections, we have concluded that the Robert Emmett Ginna Nuclear Power Plant Unit No. 1 can be operated as proposed without endangering the health and safety of the public.

2.0 SITE AND ENVIRONMENT

2.1 Site Description

The site, which consists of approximately 338 acres, is located in the township of Ontario, Wayne County, New York, on the south shore of Lake Ontario. The minimum distance from the facility to the site boundary (excluding the boundary on the lake front) is 1550 feet. The distance to the nearest offsite residence is about 2000 feet. Based upon estimates for the year 1990, 455 people will live within one mile, 1560 within two miles, and 14,491 will live within five miles of the site. A low population zone distance of three miles has been selected.

2.2 Meteorology

The applicant has collected approximately two years of onsite meteorological data at the Ginna site, which include measurements of wind speed, wind direction, and temperature at various elevations on a 150-foot tower. The measured average wind speed at the 50-foot elevation is approximately 11 mph and wind speeds lower than 1 mph occur less than 1% of the time. Inversion conditions were found to occur 37% of the total time, regardless of wind direction. These results are not unusual for typical lakeside sites and are consistent with conditions postulated for analysis of potential releases of radioactivity from the facility.

2.3 Hydrology

Our consultants from the Department of the Army, Coastal Engineering Research Center, (CERC), have evaluated the potential elevations of flooding at the site. The design lake level as recommended by our consultants was 251.9 ft msl. This maximum water level value was based upon a lake stage of 248.05 ft plus incremental heights for rainfall (0.40), wind tide (2.15), pressure effect (0.27), and wave effect (1.00). In addition, consideration of wave runup and overtopping would require some incremental protection above this level. The applicant proposed permanent protection to an elevation of 254 ft, which our consultants advised could result in water overflowing the breakwall by 4.9 ft per wave for the maximum probable design storm conditions. The applicant has agreed to modify its design to accommodate the maximum probable flood. Until the permanent protection features are completed, the applicant will provide temporary protection to assure no wave overtopping with the postulated storm. The temporary protection will be based upon current lake water elevation as measured at a gaging station in Rochester.

An accidental spill of radioactive liquids in the plant area, if it should occur, will flow into the lake either directly, via the Deer Creek channel, or with the ground water. There is little or no potential for contamination of wells in the area of the site because of such a spill.

Since the water intake for Ontario, New York, is within 6000 feet of the discharge point from Ginna, lake dilution was of particular interest for this facility. The applicant performed diffusion studies of liquid effluents into Lake Ontario. These dispersion studies were reported in Appendix 2A of the FSAR. As indicated in these studies, a twenty-fold or greater dilution should occur before the plant discharge reaches the area of the nearest water intake. Our hydrologic consultants at the U. S. Geological Survey have informed us that the lake dilution estimates made by the applicant are reasonable. In the Technical Specifications for the facility, releases of radioactive effluents to the discharge canal are limited such that concentrations at the point of discharge do not exceed 10 CFR Part 20 limits for unrestricted areas. Dispersion in the lake would futher dilute such releases.

We have concluded that the hydrologic characteristics of the site do not present any unusual problems, and are acceptable with respect to the health and safety of the public.

2.4 Geology and Seismology

The foundation material supporting the plant consists of compact granular soils and bedrock with allowable bearing values of 3 to 6 tons and 30 to 40 tons per square foot, respectively. The major nuclear station structures bear on the Queenstown Formation - alternating strata of thinly to thickly bedded, dense, fine-grained sandstone, silty sandstone, and sandy siltstone, with occasional thin beds of fissure shale, horizontally bedded. In some cases, the structures are founded on a thick layer of natural or compacted granular soils immediately above bedrock. We and our structural consultants have reviewed the foundation designs and consider them to be adequate.

Our consultant, the U. S. Department of Interior Geological Survey, studied the geologic aspects of the site during our review of this facility prior to issuance of a construction permit. They concluded that the geology of the site provided an adequate founding medium for the facility building and structures. We agree with this conclusion. No information was developed during excavation or construction which changed this conclusion.

The applicant's seismic design bases specify that (a) for a maximum ground acceleration of 0.08g, resultant stress levels for critical components, equipment and structures necessary to ensure a safe and orderly shutdown will not exceed code allowables; and (b) for ground accelerations of 0.20g, there will be no loss of function of critical structures and components necessary to ensure a safe and orderly shutdown. Based upon the report provided at the construction permit stage by our seismic consultant, the U. S. Coast and Geodetic Survey, we have concluded that these design basis accelerations are acceptable. Structures, equipment and components designed to these conditions are designated as Class I. The facility design has been reviewed by our consultant, Nathan M. Newmark Consulting Engineering Services of Urbana, Illinois which has concluded, and we agree, that the facility was generally designed and constructed in accordance with the seismic design criteria. The design of Class I piping systems was not adequately documented in the Final Safety Analysis Report. Unitl this design information is submitted and reviewed to our satisfaction, the Ginna license will limit power to 5 Mw(t).

2.5 Environmental Radiation Monitoring

The principal requiremsnts for the applicant's environmental radiation monitoring program are listed in the Technical Specifications. The extent of sample analyses is dependent upon the radioactivity being released by the facility. The higher the release rate from the facility, the more comprehensive is the required analyses of these samples. The applicant proposes to collect air, surface water, well water, food materials, marine organisms, and lake bottom sediments. Recommendations from our consultants, the Fish and Wildlife Service of the U. S. Department of the Interior, have been incorporated into the applicant's environmental radiation monitoring program. We conclude that the applicant's program will be adequate for monitoring the radiological aspects of plant operation on the environs and assessing the health and safety aspects of the release of radioactivity to the environment from the operation of the plant.

3.0 FACILITY DESIGN

The following sections briefly describe the design of those systems and features of the Ginna plant that are important to its safe operation. The reactor is a closed cycle, pressurized, light water moderated and cooled reactor. The principal design features and the materials of construction for this reactor are similar to those which have been reviewed and approved for other pressurized water reactors. The adequacy of most of these features has been demonstrated from a safety standpoint from operating experience. In addition, the design of the Ginna reactor incorporates many of the features of the new generation Westinghouse-designed PWR's. These include a pressure vessel design using Section III of the ASME Boiler and Pressure Vessel Code, use of accumulators, use of part length rods, use of burnable poisons, and use of Zircaloy cladding on the fuel.

3.1 Reactor Design

3.1.1 Reactor Coolant System

The primary coolant system, including the pressure vessel and all piping, is designed for a pressure of 2485 psig and a temperature of 650^{o}F. All material in contact with the primary coolant is stainless steel (or inconel in the steam generators). The primary coolant will be pumped through each of two loops by pumps rated at 90,000 gpm (at 252 feet of head).

The reactor pressure vessel has an internal diameter of about 132 inches and an overall height of about 39 feet. All inlet and outlet nozzles for the primary coolant are located in a plane above the level of the active core. There are no large vessel penetrations below this level. The upper head closure will be sealed with gaskets. The vessel was designed and constructed in accordance with Section III Class A of the ASME Code.

The Ginna plant does not have isolation valves in the coolant lines between the reactor vessel and the steam generator. The omission of these valves has safety significance in the event of a rupture of a steam generator tube. Should such a rupture occur during operation, primary system coolant would leak to the secondary system due to the pressure differential between the primary and secondary systems and, without isolation valves, this leakage could not be isolated. As a consequence, some blowdown of the secondary system to the atmosphere might occur. If radioactivity is present in the primary system some escape of radioactive material to the atmosphere would likely result. On the other hand, omission of the primary isolation valves eliminates the possibility of a cold water accident caused by starting operation of an isolated and cold coolant loop, because no means of loop isolation is available. We have concluded that the absence of the isolation valves is acceptable from a safety standpoint on the basis of the analysis discussed in Section 4.3 of this evaluation.

The reactor core will contain 121 fuel assemblies each of which contains 179 cylindrical fuel rods. Fuel rods will consist of UO_2 fuel pellets clad in Zircaloy-4 tubes with each end sealed by a welded plug. Each tube will have an outer diameter of 0.422 inch and a length of 144 inches. Fuel assemblies with similar confirgurations have been tested in water cooled power reactors that are presently in operation. Use of Zircaloy as the fuel cladding material and the increased core length increase the probability of axial xenon power instabilities. For this reason, part-length control rods are provided for power shaping, and burnable poison rods will be used in the initial core loading to ensure a negative moderator temperature coefficient. After burnup of the initial fuel charge, the moderator coefficient will be inherently negative, hence burnable poison rods will not be required for subsequent core loadings.

We have evaluated the applicant's proposed use of the part length control rods for controlling axial power distribution. For this core, where diametral xenon power redistribution is not anticipated, dependence is placed upon the external neutron monitoring system to position the part length rods for axial power shaping.

Sensitivity tests were performed at the Connecticut Yankee facility to show that the external monitors would provide sufficient information for determining in-core power distributions. As part of the startup test program, additional tests will be performed at Ginna to verify the expected instrument sensitivities.

Our evaluation of the mechanical design of the part length rods, burnable poisons, and Zircaloy-4 cladding is given in the following subsections.

The reactivity of the reactor will be controlled by 29 full length control rods having a calculated combined worth of 7.1% Δk at the end of core life, and by 4 part length control rods. Short term reactivity changes will be made by repositioning the full length rods; however, long term reactivity effects will be compensated by adjustment of the boric acid concentration in the reactor coolant. As is the case in previously licensed reactors using boric acid control, rod insertion limits are administratively controlled to ensure sufficient reactivity is available for shutdown. Insertion of a control bank beyond the limits causes an alarm and alerts an operator to add boric acid using normal methods. A second alarm indicative of a greater deviation is also provided.

We evaluated the method of reactor control as originally proposed by the applicant, and were concerned with the design capability for alerting an operator to unusual power distributions within the core. If, for example, one control rod in a fully inserted control bank were to be fully withdrawn from the core, local power peaking would occur which could cause encroachment on a safety limit. Since all alarms to warn of this condition utilized the plant computer, we concluded that a redundant alarm system independent of the computer was required. The applicant will install this alarm system. The Technical Specifications define the requirements for these alarms.

3.1.2 Part Length Absorber Rods

Four part length absorber rod assemblies have been added to the Ginna reactor since our review at the construction permit phase. The function of the rods is to control axial power distribution. Each part-length stainless-steel-clad rod will contain silver-indium-cadmium alloy in the bottom 36 inches, with the remaining portion (follower) filled with aluminum oxide. The four assemblies of 16 rods each are distributed in the core in a cylindrical pattern so as to minimize their effect on the radial power distribution. The reactivity worth of all of these rods moving from their position of maximum worth to full insertion is approximately 0.3% Δk. The rods will be positioned manually by the operator as a bank. The drive mechanisms for the part-length rods is a roller nut, lead screw arrangement. The rods are driven in or out by a rotor assembly within the pressure housing. Sequential pulses to a stator located outside of the housing will cause rotation in 15 degree increments per pulse. Maintaining power on any one of the six segments of the stator is sufficient to overcome the rundown torque of the mechanism. If there is a loss of power, a mechanical brake is actuated which prevents rod motion.

3.1.3 Burnable Poison

Burnable poison rods in the form of borosilicate (pyrex glass) contained in stainless steel tubes will be located in unused control rod positions. A total of 528 poison rods will be used in the initial core loading, but will not be required for later fuel charges.

Use of the burnable poison will eliminate the positive moderator temperature coefficient of reactivity which would otherwise be present in the first core. The negative moderator temperature coefficient is desirable in damping power redistributions due to xenon effects, and in preventing the occurrence of a small power burst in the event of a loss-of-coolant accident.

3.1.4 Zircaloy Clad Fuel

The Ginna reactor will use Zircaloy-4 clad fuel elements. The fuel rods are 0.422 inch in diameter and the clad is 24 mils thick. Experimental irradiation of Zircaloy clad fuel elements has been undertaken at Saxton and at Zorita. Although the irradiation tests have not been completed, the test results to date have not revealed any unexplained behavior.

There are R&D programs relating to the performance of Zircaloy under accident conditions which will not be completed by the time Ginna will be in operation. Our evaluation of the Zircaloy clad under accident conditions is discussed in Section 3.3 of this evaluation.

3.1.5 Conclusion

We have concluded that the reactor design features for the Ginna plant are adequate.

3.2 Containment Vessel

3.2.1 Containment Design

The principal structural features of this plant which differ from previously licensed nuclear power plants are the use of the concrete containment vessel with vertical prestressing strands anchored into rock, and the use of conventional mild steel bar reinforcing in the circumferential direction.

The reactor containment is a steel-lined concrete vertical right cylinder with a flat base and a hemispherical dome. It is prestressed vertically and reinforced circumferentially with mild steel bars. The following are its principal dimensions: wall thickness, 3'-6"; dome thickness, 2'-6"; base slab thickness, 2'-0"; height to dome springline, 99'-0"; inside diameter, 105'-0"; cylinder and dome liner thickness, 3/8"; base liner thickness, 1/4"; net free volume, 997,000 cu. ft. The 90-0.25" BBRV vertical tendons are attached to identically sized grouted rock anchors which rely only on the weight of the rock and not on any rock tensile strength. Since the rock anchors take all vertical loads, the base slab, not having large shears imposed on it, is not as significant structurally as in other designs.

The dome design consists of rebars placed in three directions, which terminate at a continuous plate imbedded in the prestressed compression zone. The dome-to-cylinder discontinuity contains vertical crack initiators between the tendon anchorages. They will ensure that the pressure load will not produce significant cracking in the dome without vertical cracks also being present in the cylinder. The initiators will permit uniform propagation of cracks below the discontinuity, as is assumed in the model upon which the design is based.

The containment wall rests on flat neoprene bearing pads made of two layers of neoprene between three steel shims. This provides a sliding and rotating hinge at the base of the wall which reduces the moments in the lower portion of the wall. Horizontal restraint of the wall is provided by high strength radial bars anchored into the wall and the base mat. These are installed so that they do not impose a rotational restraint on the wall. We and our consultants evaluated the design of the hinge and find it acceptable.

The Ginna containment, under structural proof testing, or simultaneous maximum earthquake and design basis accident, will crack vertically. The resulting structure then can be characterized as a series of vertical prestressed panels held together by the circumferential reinforcing. These panels are also subjected to differential vertical shears which are resisted by the dowel action of the outer ring of circumferential reinforcing. In analyzing the structures, the applicant did not consider the inner circumferential reinforcing, aggregate interlock in the concrete, or the contribution of the liner in resisting shears. It is our opinion, and that of our structural consultants, that this design provides adequate shear capacity in the containment structure.

The applicant has stated that the Ginna containment design can withstand a symmetrical failure of 5% of the individual tendon wires or 3 adjacent tendons without a loss of containment function. We have evaluated this capability and have concluded that it provides an appropriate margin of safety.

The 1/4-inch diameter cold drawn and stress-relieved prestressing wires conform to ASTM A 421-59T, Type BA, "Specifications for Uncoated Stress-Relieved Wire for Prestressed Concrete," with a minimum ultimate tensile stress of 240,000 psi. Each BBRV tendon is made up of 90 parallel wires with 3/8" diameter cold-formed button heads at the ends which bear on a perforated steel anchor head. On the basis of our initial review of Ginna at the construction permit stage, and of our subsequent reviews of the BBRV system in connection with several other construction permits, we have concluded that the BBRV prestressing system, as installed at the Ginna plant, will furnish adequate reliability and margins of safety.

The containment liner is insulated with Johns-Manville Vinylcel, a closed-cell polyvinyl chloride foam, and covered with metal sheeting. This insulation is provided on the sidewalls and to a point 15'-0" above the spring line.

The liner is carbon steel plate conforming to ASTM A 442-60T Grade 60, with a minimum yield point of 32,000 psi. It is 1/4-inch thick at the base, and 3/8-inch thick at cylinder and dome. Anchorage to concrete is by means of stagger welded 3-inch channels on the cylinder and studs in the dome. The bottom plate is covered with 2 feet of concrete fill and anchored to structural tees. All seam welds are covered with test channels. The liner design has been checked for buckling potential by the applicant and by us. The liner design and installation are satisfactory.

Penetration sleeves conform to SA-106, Grade B, with a minimum yield strength of 31,000 psi at 300°F, and are designed according to the ASME Boiler and Pressure Vessel Code, Section III, Nuclear Vessels.

We made extensive review of the primary containment openings for equipment and personnel when a design change was made from steel opening frames to reinforced concrete frames. We concluded that the revised design was acceptable.

We have concluded that the containment design is acceptable.

3.2.2 Containment Vessel Testing

Preoperational testing consisted of structureal proof tests at several pressures up to 115% of design pressure. At all levels (35, 50, 60, and 69 psig), strain, deflection and rotation readings were taken and compared with acceptance limits established by the applicant to determine whether the building met its structural requirements. We reviewed the limits and agreed that they were acceptable. The results of the structural proof test showed that all values were within acceptable limits.

Initial leak rate testing has been conducted at 60 psig design pressure for 24 hours and at a reduced pressure of 35 psig. The acceptance criterion was 0.25% of the free volume per day at 60 psig which was met with adequate margin.

Periodic monitoring for leakage can be done by the use of liner seam weld test channels by integrated leak rate tests, and by local leak detection at valves andpenetrations.

The applicant proposes to inspect 14 tendons spaced equally about the containment at intervals of 6 months, 1 year, 3 years, and every 5 years thereafter, following the structural test. This inspection will be visual for the tendon ends, and will also include prestress confirmation liftoff readings. Each of 40 tendons includes an extra 1/4-inch diameter wire specimen which can be removed to check for corrosion.

We have concluded that the above program provides appropriate means for determining the contained reliability of the containment structure and its leak tight capabilities.

3.3 Emergency Core Cooling System (ECCS)

3.3.1 System Design

The emergency core cooling system (ECCS) is designed to deliver borated cooling water automatically to the core in the event of the loss of either primary or secondary coolant. The performance criteria for the ECCS design are: (1) the cladding temperatures will be limited to less than the clad melting point; (2) gross distortion of the core or fragmentation of the clad will not occur; and (3) the total metal-water reaction will involve less than 1% of the metal in the fuel cladding. These criteria will be met for all size breaks in the primary system up to and including a double-ended rupture of the largest pipe (29-inch ID) with power available only from an onsite diesel generator.

The ECCS for this plant consists of (a) one high pressure coolant injection and recirculation subsystem (HPS), (b) one low pressure cooling injection and recirculation subsystem (LPS), and (c) one accumulator subsystem.

The equipment in the high pressure portion of the system consists of two accumulators and three safety injection system (SIS) pumps. The SIS pump suctions are normally aligned with the two boric acid tanks, which contain a 12% concentration of boric acid. When these tanks are emptied, the suction of the SIS pumps is automatically transferred to the refueling water storage tank. This suction transfer is actuated by redundant low level instrumentation in the boric acid tanks. In addition, if the isolation valves in the lines to the boric acid tanks fail to open within two seconds after receipt of a loss-of-coolant accident (LOCA) signal, then the SIS pump suctions are automatically transferred to the refueling water storage tank.

The LPS contains two low head pumps (each having a capacity of 200 gpm at 280 ft head), which are also used for residual heat removal (RHR). These pumps take suction from the refueling water storage tanks during coolant injection. When this tank has been emptied, the LPS suction is realigned by the reactor operator to the containment sump to initiate coolant recirculation.

The design basis for the ECCS is that the performance criteria be met by the operation of one accumulator, one low pressure pump (RHR), and two SIS pumps, all delivering at rated capacity. For intermediate size breaks (between 4 and 10-inch equivalent diameter), at least two of the SIS pumps in conjunction with either one accumulator or one RHR pump must function. For smaller breaks (less than 4 inches), only two of the SIS pumps are required.

The capacities of the accumulators are based on the assumption that one accumulator injection line injects into the broken line and therefore no water is added to the vessel from that accumulator. The SIS pumps are sized assuming loss of water injection through one of the four injection lines. However, the leak can be isolated by the operator, by means of two motor-operated valves which are provided to isolate two of the SIS pumps form either of the high pressure headers.

We have performed a failure mode analysis of the as-built ECCS and have concluded that sufficient component redundancy is provided to ensure adequate coolant injection at both high and low vessel pressure even if a single active component fails to operate. For this plant both the LPS and HPS could be used for long-term cooling, although use of the LPS is the applicant's preferred method. Thus, two completely independent paths from the containment sump to the reactor vessel have been provided, and long-term coolant recirculation would not be interrupted by the failure of any single active or passive component.

3.3.2 Accumulator Nitrogen

The water in the accumulators is injected into the reactor core by nitrogen gas pressure. After the injection of water is complete, the nitrogen gas may enter the core.

The applicant has analyzed a spectrum of breaks to determine (a) how much nitrogen will reach the core, (b) the effect of this nitrogen on core heat transfer, and (c) the effect of the limiting assumption that all the nitrogen reaches the core. Based on our review of this analysis, we conclude that the limited amount of nitrogen which will reach the core for the worst break will not significantly affect ECCS performance. Furthermore, since the clad temperature transient is reversed before nitrogen can enter the vessel, even if all the nitrogen were to reach the core, it would only affect the cooldown rate and not the peak clad temperature.

3.3.3 Power to ECCS Systems

The power to the ECCS pumps is arranged so that two of the pumps are connected to separate electrical buses. Each bus is connected to only one of the two diesel generators. The third pump can be connected to either of the two buses by the operator.

All of the ECC subsystems can accomplish their functions when operating on emergency (onsite) power. If one of the two diesel generators fails to start, a minimum of one low head and two high head pumps would be available for operation. The diesel loads and the ECCS starting sequence are arranged so that the system will be pumping at minimum acceptable capability, assuming no component failures after the diesel failure, within about 28 seconds following initiation of a LOCA signal.

3.3.4 Containment Isolation

All of the ECCS injection lines which penetrate containment have at least one check valve inside containment, close to the reactor coolant system, and a remote-control motor-operated valve outside containment. These valves reduce the probability of a failure within the ECCS causing a LOCA and they permit isolation of a failure in an ECC subsystem following a LOCA. The ECCS suction lines from the containment sump have remote-control motor-operated isolation valves inside and outside the containment. The operators of the valves inside containment are located so as to ensure that they will not be flooded in the post-accident containment environment. These valves also ensure that in the event of a failure in the suction pipe between either valve and containment during long-term coolant recirculation, post-accident radioactivity release can be limited.

Auxiliary building sump level indicators and alarms and flow instrumentation are adequate to warn the operator of recirculation loop failure.

3.3.5 Conclusion

We conclude that the ECCS will (a) limit the peak clad temperature to well below the clad melting temperature, (b) limit the fuel clad-water reaction to less than 1% of the total clad mass, (c) terminate the clad temperature transient before the geometry necessary for core cooling is lost and before the clad is so embrittled as to fail upon quenching, and (d) reduce the core temperature and remove decay heat for an extended period of time. This protection is provided for all sizes and locations of pipe breaks, up to and including the instantaneous double-ended rupture of the largest reactor coolant pipe.

3.5 Containment Spray and Air Filtration Systems

A combination of separate chemical additive spray and charcoal adsorber systems is provided for removal of radioactive iodine from the containment atmosphere. We have reviewed each of these systems in terms of their iodine removal capabilities and, in addition, have evaluated the chemical compatibility of the spray solution with other reactor components.

Two independent spray subsystems are provided, each with a flow rate of 1250 gpm. Sodium hydroxide is used as the chemical additive. The effectiveness of this chemical for the design proposed has been verified in a large number of experiments conducted at Oak Ridge National Laboratory. The rate of addition is controlled to give a constant pH value ranging from 8.7 to 9.5 for the incoming solution, with flow continuing over a maximum seven-hour period.

The air filtration systems are located within the containment and are of the recirculation type. They are equipped with demisters, high-efficiency particulate air filters, and charcoal adsorbers in series. Each unit has a design flow rate of 38,000 cfm, with a face velocity of 40 fpm. In our accident dose calculations, we have assumed that 90% of the containment volume would pass through the charcoal adsorbers, and that 90% of the inorganic iodides are removed in each pass through these units. We have also conservatively assumed that no organic iodides are removed by the charcoal adsorbers, because in the event of an accident, the charcoal units would be exposed to a high temperature and high relative humidity environment, which would make them less effective for organic iodides removal.

The removal constants for inorganic iodides that we used in our accident calculation for the spray system and filter system are as follows. For the spray system, on the basis of our conservative model for one of two subsystems operating, a removal constant of 3.7 hr^{-1} is obtained. For the filter system, with only one fan operating a removal constant of 1.7 hr^{-1} is obtained. For both the spray and filters operating simultaneously, overall removal constant of 4.5 hr^{-1} results. We have conservatively assumed that ten per cent of the iodine is not removed by either filters or sprays. On the above basis, we have estimated the overall iodine reduction factor is 5.0 for the initial two-hour period and 7.8 for the initial eight-hour period.

We conclude that there is reasonable assurance that these systems would be effective in the unlikely event of a loss-of-coolant accident, and that the values used for iodine removal are conservative.

3.5 Auxiliary Systems

3.5.1 Residual Heat Removal System

The Residual Heat Removal (RHR) System normally removes the core decay heat at a controlled rate after the reactor has been shutdown and cooled to 350°F. Part of the system is used during a loss-of-coolant accident as described in the section on ECCS.

The system is similar to the systems in other reactors and contains two pumps and two heat exchangers. Failure of one component would not interrupt core cooling, but would only reduce the rate of cooldown.

The system is isolated from the reactor coolant system by two remotely operated valves. The valves are interlocked so that they cannot be opened until the reactor coolant pressure is within the design limits for the RHR system.

The design conforms to current practice and is acceptable.

3.5.2 Component Cooling System

During plant operation, the Component Cooling Water System transfers heat from the Chemical and Volume Control System and other plant components to the service water. During shutdown or after an accident, the system supplies cooling water to the Residual Heat Removal system pumps and heat exchangers.

The system consists of two pumps, two heat exchangers, a surge tank and connecting piping and valves. Although the pumps and heat exchangers are redundant, they are connected by single pipe headers. A failure in any header would disable the system. However, under accident conditions, the RHR system could continue to recirculate spilled reactor coolant with the heat being removed by the containment fan coolers rather than the RHR heat exchangers.

The system components are designed to criteria based on the ASME Code, Section VIII, and the USASI B31.1 Code. Class I seismic criteria were used in the design. We have reviewed the design of this system and have concluded that it is adequate.

3.5.3 Fuel Pool Cooling

The spent fuel pit cooling loop is designed to remove the decay heat from the stored spent fuel elements. The system is designed to remove the heat generated by one and one-third cores.

The system consists of a pump, a heat exchanger and a bypass filter and demineralizer. Water purity is maintained with a skimmer pump and filter.

The piping is arranged so that failure of any one pipe will not drain the water in the pool below the top of the fuel elements. Normally, the temperature of the water in the pool is maintained at approximately 120°F. However, if the pool water were not to be cooled and if the pool were to contain one and two-third cores, the water temperature would rise 40-60°F in eleven hours, which we consider sufficient time in which to reestablish cooling of the pool water.

We have concluded that the design is adequate.

3.5.4 Service Water System

The service water system is designed to supply cooling water, during normal operation, to the component cooling system, spent-fuel pool cooling system heat exchangers and also to equipment in the power conversion plant and the air conditioning system. After a postulated accident, the system supplies cooling water to the containment coolers, the diesel generators, the component cooling system heat exchangers, and the auxiliary feedwater pumps.

The system includes four pumps, three of which have the capacity to supply the normal loads. Under accident conditions, one pump is sufficient to supply the essential equipment. The piping system is arranged so that there are at least two flow paths to each essential load, and non-essential uses are automatically isolated. Valving is provided to isolate any single failure and permit continued operation of the system.

The system is designed to Class I seismic criteria. Leakage into the system from the containment coolers is monitored by radiation alarms. Since the system is normally in operation, any failures will be evident immediately. We conclude that the system is adequate.

3.5.6 Fuel Handling System

The fuel handling system is designed to transfer spent fuel to the storage pool, and to transport and provide storage for new fuel.

The spent fuel assemblies after withdrawal from the core are transferred along the refueling canal, through the fuel transfer tube, and into the storage pit by means of a conveyor car. The water level above the fuel during the transfer is sufficient to limit the maximum radiation level at the water surface to less than 5 mr/hr. New fuel is lowered into the storage pool and delivered to the core by the same equipment operating in a reverse procedure. Storage of new fuel is limited to one-third of a core or 40 assemblies. Spent fuel storage is provided for 1-2/3 cores. Both new and spent fuel racks are spaced 21 inches apart and are designed to limit the effective multiplication factor to 0.90 or less, even if the assemblies are flooded with unborated water. Refueling will be done with borated water containing 2000 ppm boron. The fuel handling system is essentially the same as for previously reviewed and approved reactors and is acceptable.

3.6 Instrumentation and Control

3.6.1 General

Our review of the Ginna instrumentation encompased the following subjects: (1) Reactor Trip System; (2) Safety Injection Initiating System; (3) Containment Isolation Initiating System; (4) Containment Spray Initiating System; (5) Steam Line Isolation Initiating System; (6) Rod Control System; (7) Separation of Control and Protection Systems.

The Commission's General Design Criteria, and the Proposed IEEE Criteria for Nuclear Power Plant Protection Systems (IEEE No. 279, dated August 30, 1968) served, where applicable, as the bases for judging the adequacy of the instrumentation and control systems.

3.6.2 Reactor Trip System

The reactor trip system is designed on a channelized basis to achieve isolation between redundant protection channels.

Our review indicates that the basic design approach of the reactor trip system is satisfactory. The review included considerations of channel redundancy, independence, testability, automatic removal of operating bypasses, and manual trip initiation. Redundant instrumentation which is located in the control room is housed in separate racks. The test procedure is identical to that which has been reviewed previously and found to be satisfactory.

The applicant has demonstrated conformity of this design to all sections of IEEE-279 with the exception of Section 4.4 (Equipment Qualification). The design requirements for environmental conditions of pressure, temperature and humidity have been specified; however, the testing in progress is not scheduled for completion until July, 1969.

We have determined that the design of the reactor trip system is satisfactory. Power levels will be restricted to 5 Mwt until we review the results of the R&D program on the environmental tests of the process instrument transmitters.

3.6.3 Safety Injection Initiation

Safety Injection is automatically initiated by the following signals: (a) Low Pressurizer Pressure in coincidence with Low Pressurizer Water Level, (b) High Containment Pressure, or (c) Low Steam Pressure in either steam generator. Signals (a) and (b) provide diverse methods of detecting ruptures of the primary system.

We have determined that independence of the logic channels is preserved, except as follows: one channel operates a pump (and associated equipment) which is connected to one of the two a.c. buses, and the "swing" pump, which can be connected to either a.c. bus (but not both). The other channel operates the remaining pump at the other bus, and the swing pump. If there are no a.c. or circuit failures, all three pumps will actuate with the swing pump connected to one a.c. bus. The bus selected is determined by a pair of timers, with different settings (approximately 5 to 7 seconds) which are initiated by the safety injection signal and the presence of voltage at the respective emergency bus. Each timer operates one of the two breakers between the swing pump and the respective a.c. bus. Presence of voltage at one timer operates an interlock which interrupts the other timer. Further, the closure of either breaker "interlocks out" the other. It is essential that both breakers never be closed simultaneously when the station is relying on diesel generator power since this would tie the independent a.c. buses together and possibly initiate a loss of both buses. We have concluded that the interlocks for this specific design satisfy the single failure criterion and are otherwise adequate to preclude the simultaneous closure of both breakers.

We conclude that the design of the Safety Injection Initiation System satisfies all applicable criteria and is acceptable.

3.6.4 Containment Isolation Initiation

The following signals automatically initiate containment isolation: (a) Any automatic safety injection initiation signal, or (b) high containment activity.

We reviewed the Containment Isolation Initiating circuits for redundancy and indpendence of instrumentation, logic channels and d.c. sources, and found them to be satisfactory.

We verified that each logic channel controlled one valve. Thus a failure in either channel, or in the d.c. supply to the channel, could affect only one valve. Each channel is fed from one of the two d.c. sources.

We conclude that the design of the Containment Isolation Initiating system conforms to applicable criteria, and that the dual channel concept has been properly implemented.

3.6.5 Containment Spray Initiation

Containment spray is initiated by two redundant logic channels similar to the Safety Injection Initiating system. Each channel operates one containment spray pump, one of two parallel valves in its discharge line, and one of two parallel valves in the alternate pump's discharge line.

The two logic channels are energized from separate d.c. sources. The two discharge valves operated by one channel are energized from one motor control center. A second motor control center operates the other two valves (one in each discharge line).

Each of the redundant channels of instrumentation operates two relays, the contacts of which comprise the respective logic trains.

We conclude that the containment spray initiated circuits conform to all applicable criteria and are acceptable.

3.6.6 Steam Line Isolation Initiation

The following signals initiate Steam Line Isolation: (a) High Steam Flow in that steam line in coincidence with any safety injection signal closes the valve in that line, (b) High Containment Pressure closes both valves.

Each steam line isolation valve (one valve per line) is controlled by four solenoid valves. The solenoid valves initiate isolation by closing off the air supply to the isolation valve operator and by venting the operator. Each of these functions is accomplished by redundant solenoid valves.

The instrumentation is redundant, and the logic circuits are dual channel, similar to those used to initiate the ECCS. We have reviewed the logic channels and determined that no single failure (except for mechanical failure of an isolation valve itself) can prevent closure of any isolation valve. For example, the loss of a d.c. source affects only one vent solenoid and one cutoff solenoid for each isolation valve. Loss of either logic channel has the same effect as loss of a d.c. voltage source.

During low power-low steam flow operation, the (high) flow setpoint is lowered in order to sense a line break under these conditions. The sub-system which lowers this setpoint when required, is a two-of-four logic matrix which receives inputs from four "T_{avg}" channels. This matrix satisfies the requirements of IEEE-279 relating to multiple set points.

We conclude that the design of the steam line initiating circuits conforms to all applicable criteria, and is acceptable.

3.6.7 Rod Control System

The rod control system at Ginna utilizes solid state components in lieu of rotation cams, switches, and relays.

For the purposes of control, the 33 rods are divided into symmetrical banks which are further divided into groups. There are four banks of control rods (A, B, C and D), one bank of shutdown rods, and one bank of part length rods. Each bank, except the part length bank, consists of one or more groups that are moved sequentially such that the groups in that bank are always within one step of each other. Maximum rod speed, as limited by the pulser, is 72 steps per minute. The mechanisms themselves have an inherent mechanical limit of approximately 77 steps per minute.

All full length rod power supplies are interrupted by a reactor trip signal; thus, they cannot retain voltage at any coil under scram conditions. The connection of any external d.c. source to the stationary or movable coils will be alarmed by circuits which compare the demand current (from the slave cyclers) to the actual current.

The a.c. portion of the scram bus is a single bus interrupted by two trip breakers and is of the same design as that approved for recent plants. We have concluded that a single bus is adequate because of the high power (400 kW) and the non-standard frequency and voltage required for an external source to disable the scram function. Further, no single-phase operation, such as would occur in the event of a single short circuit around the breakers, can hold up the rods. In addition, the exposure to external sources is limited by the short run of enclosed bus duct above the cabinets.

Position indication is displayed on individual meters energized from a linear variable differential transformer (LVDT) on each drive. Each LVDT also operates its respective "rod-bottom" light. The demand position of each group is displayed on counters, and any deviation between an LVDT indication and the demand position results in an alarm.

We have reviewed the rod position indicating system and have concluded it is satisfactory since it displays the positions of all rods simultaneously, and automatically alerts the operator in the event of a deviation. Any single fault will result in a deviation alarm and a visible indication derived from comparing the LVDT and demand indicators. Further, we understand that the LVDT components are fabricated from materials with a high temperature rating and that additional cooling of the LVDT coils is being provided. We have concluded that these precautions provide reasonable assurance against spurious LVDT indications.

Based on the foregoing, we conclude that the rod control and indicating systems are acceptable.

3.6.8 Separation of Control and Protection

We have reviewed the applicant's analysis concerning the separation of control and protection functions, and have concluded that, with respect to protection against single, random failures, the designs satisfy Section 4.7 of IEEE-279. Section 4.7 states that, where a plant condition which requires protective action can be brought on by malfunction of the control system, and the same malfunction prevents proper action of a protection system channel or channels designed to protect against the resultant unsafe condition, the remaining portions of the protection system shall independently satisfy the single failure criterion and all design basis functional requirements. For example, although a failure of any one of the four Nuclear Flux channels or "T_{avg}" channels could initiate a control system malfunction and cause a power excursion, a two-of-three protection system would be left to cope with the excursion.

With respect to systematic failures, we are pursuing with the applicant's supplier (Westinghouse) the capability of the designs to withstand such events.

3.6.9 Conclusion

We conclude that the instrumentation and control systems satisfy the proposed IEEE Criteria for Nuclear Power Plant Protection Systems and applicable portions of the Commission's General Design Criteria, and therefore, are acceptable.

3.7 Electrical Systems

3.7.1 Offsite Power

Offsite power is fed into the plant via two 34.5 kV lines which terminate at the single startup transformer. One line is run underground and originates at the plant substation approximately a half mile from the plant. The other is an overhead line which originates at Substation 204, located a distance of 3.5 miles from the plant substation. The closest point of approach of the overhead line to the plant substation is approximately one mile.

The breakers at Substation 204 are operated from a d.c. battery separate from the d.c. source for the plant substation. Each substation can receive power from several 115 kV lines which connect to the main RG&E transmission network.

Our review indicates that the single startup transformer is the only portion of the offsite power system which renders the system vulnerable to a single failure. Should operation of the engineered safety features be required, this vulnerability would persist for approximately eight hours which represents the time required to route power to the plant via the main transformer.

In view of the high reliability of transformers and the consequent low probability of failure during any specific eight hour interval, we believe that the installation of a single startup transformer is acceptable.

The applicant has also presented an analysis to verify that the sudden loss of the Ginna plant electrical output will not result in an instability in the offsite power system. A scheduled loss-of-load test of a 1000 Mwe plant (Ravenswood No. 3) operating at full load has been conducted on the New York Power Pool, without causing system instability. In addition, an unscheduled loss of the same plant operating at 800 Mwe did not upset the system stability.

We agree with the applicant that the loss of the Ginna Plant, operating at full load (520 Mwe), would not upset the external system stability.

3.7.2 Onsite Power

Onsite power is furnished, if required, by two diesel engine generating sets rated at 1950 kW for continuous operation, and at 2250 kW for operation for a two-hour period. Either diesel can supply the required safety system loads. The maximum expected load for each diesel occurs during the initial safety injection phase during a loss-of-coolant accident and is 2010 kW, which is essentially the continuous rating. This load is less than the two-hour rating and would be sustained for no more than 1-1/2 hours. All subsequent loading is less than the continuous rating. The diesels and loads are divided on a split-bus arrangement. There is no automatic tie between the two buses.

Both diesels are started by a "Safety Injection" signal, and each diesel is started by an undervoltage condition at either of its 480 volt buses. The starting circuits are independent of each other, except that they both rely on the station batteries for control current. Our review indicates that this design is satisfactory since the complete failure of either battery will not prevent both diesels from being started automatically by the other battery.

The diesel generators are located in separate rooms. The batteries are similarly located.

We agree with the applicant that the independent (split bus) arrangement of the diesels is satisfactory, that the diesel loading is satisfactory, and that the location of the diesels and batteries in separate rooms constitutes adequate protection against loss of redundant equipment from a common cause.

3.7.3 Cable Separation

The applicant's criteria relating to the cable tray loading and separation may be summarized as follows: (a) cables, whether power, control, or instrumentation of one train or system are not mixed with cables of a redundant train or system; (b) physical separation is provided between redundant cables for control and instrument systems within a tray by means of a galvanized sheet metal barrier in cable trays; (c) the minimum physical dimensions between redundant power, control and instrument cable trays are 5 inches vertical separation and 2 inches horizontal separation; (d) metal-enclosed 4160-volt buses are used for all major bus runs where large blocks of current are carried; and (e) the routing is such as to minimize exposure to mechanical, fire and water damage.

An ambient temperature of 50°C within the reactor containment and an ambient temperature of 40°C in all other plant areas is the design basis for all power cable ratings.

All a.c. circuits within the plant are protected by three-phase circuit breakers.

We have reviewed the criteria and conclude that they reduce the possibility of cable fires, and provide protection against random and systematic failures. Our conclusion with respect to the low probability of cable fires is based on the limited cable tray loading, and upon derating factors. There is only one layer of 4160-volt cable in a tray, and a derating factor of 0.81 is used. For the 480-volt cables, a derating factor of 0.6 has been used for size #4 and larger, and 0.5 for size #6 and smaller. Further, the pressurizer heater cables have been given extra spacing, and have been derated by a factor of 0.5.

With respect to systematic and random failures, we conclude that the physical separation of redundant cables, and the metal barrier (where used) within a tray provide adequate protection against the propagation of a fire, and against any lesser single event occurring within a tray. The use of three-phase breakers in lieu of fuses should immediately isolate all three phases of a line from a fault occurring in one phase.

3.8 Radiation Monitoring

Radiation monitoring at the facility is provided by the Operational Radiation Monitoring System and the Area Radiation Monitoring System. In general, the operational monitors give early warning of plant malfunctions and the area monitors warn the operators of increasing radiation levels that could cause hazards to onsite and offsite personnel.

The Operational Radiation Monitoring System consists of monitors in the containment and plant gas effluent, containment fan coolers service water, condenser air ejector, steam generator liquid sample, component cooling system, waste disposal system and spent fuel pit heat exchanger service water.

The Area Radiation Monitors continuously sample environmental air at three sampling stations and measure the radiation intensities at the following four plant locations: (1) Control room, (2) Containment Vessel, (3) Radiochemistry laboratory, (4) Auxiliary building.

In addition to the fixed monitors, there are portable radiation detectors which have the capability of reading radiation intensities up to 10^4 r/hr.

We conclude that the radiation monitoring systems provided for the Ginna plant are adequate.

4.0 ACCIDENT EVALUATION

The applicant analyzed a number of transient and accident situations in the Final Safety Analysis Report and its subsequent amendments. These analyses included: (1) startup accidents, (2) incorrect sequence of rod withdrawal, (3) malpositioning of the part-length rod, (4) boron dilution accidents, (5) loss of reactor coolant flow, (6) loss of external electrical load, (7) feedwater control accidents, (8) steam generator tube rupture, (9) steam line break accidents, (10) control rod ejection accidents, (11) load variation accidents, (12) fuel handling accidents, (13) waste liquid accidents, (14) waste gas accidents, and (15) loss-of-coolant accidents.

We have evaluated all of the accidents presented in the FSAR, and have also made independent analyses of thepotential offsite radiological consequences of those accidents which could cause significant offsite doses. The following subsections contain our evaluation of those accidents resulting in significant offsite doses. These evaluations were made at the minimum exclusion distance of 450 meters for the 2-hour dose, and the low population zone distance of 3 miles for the 30-day dose. We have used the following meteorological assumptions in calculating the doses that could result from the unlikely occurrence of the potential accidents for this plant: from 0-8 hours - ground release, Pasquill Type F, 1 meter per second wind velocity, invariate wind direction, credit for building wake; from 8-24 hours - ground release, Pasquill Type F, 1 meter per second wind velocity 22-1/2 degree sector spread; from 24-96 hours - ground release, Pasquill Type F and 2 meters per second 60% of the period, Pasquill Type D and 3 meters per second 40% of the period, the same 22-1/2 degree sector spread; from 4-30 days - ground release, Pasquill Type C and 3 meters per second, Pasquill Type D and 3 meters per second, Pasquill Type F and 2 meters per second, each type occurring 33-1/3% of the period, and the same 22-1/2 degree sector spread for 33-1/3% of the period.

The potential offsite doses for the postulated design basis accidents are presented in Section 4.6. The postulated accidents which result in the highest offsite doses for this facility are the fuel handling accident and the accident involving the double-ended break of the primary system piping (loss-of-coolant) inside the containment vessel.

4.1 Fuel Handling Accident

We have evaluated the potential consequences of dropping a fuel assembly during refueling. We are not in agreement with the applicant on the amount of radioactive release and the offsite dose that could result from dropping a spent fuel assembly in the storage pit. The applicant has assumed in its calculations that one row of fuel rods (14) fails in the fuel handling accident. The activity assumed to have been released from the failed rods is 1060 curies of xenon-133 and 300 curies of iodine-131. All of the noble gases are assumed released from the surface of the water. Only 10^{-3} of the iodine was assumed to have reached the water surface immediately after the accident with continued convective release from the water.

In our analysis of this accident we assumed that all of the fuel rods (179) in the entire fuel assembly are perforated when the fuel assembly is dropped. We also assumed that 20% of the noble gas and 10% of the halogens contained in the fuel were released upon perforation of the rods. Further, of the halogens released, 90% retention in the fuel storage pool was assumed. The potential dose from this accident is listed in Section 4.6. As indicated, the resulting thyroid dose at the exclusion distance is in excess of 10 CFR Part 100 guideline values.

The applicant has agreed that if the differences between its model and ours cannot be reconciled, the applicant will use our model to determine the amount of radioactivity release and offsite doses which could result from a broken fuel assembly. The applicant has also agreed to take corrective measures, if necessary, to meet 10 CFR Part 100 guidelines doses. The applicant states that it will not handle irradiated fuel until this problem has been resolved.

4.2 Accidental Release of Waste Gas

The components of the waste gas system are not subjected to high pressures or stresses and are of Class I seismic design, and therefore, a rupture or failure of the waste gas tanks is highly unlikely. However, since a relief valve on each waste gas tank may leak or fail open, the applicant agreed to install a rupture disc in series with the relief valve, designed for the same burst pressure as the setting of the relief valve.

We assumed in our evaluation of this accident that the waste tank contained approximately 30,000 curies of activity with an average disintegration energy of 0.7 MeV. The resulting calculated doses for instantaneous release of activity resulted in about 3 rem dose to the whole body which is well within 10 CFR Part 100 guidelines.

4.3 Steam Generator Tube Rupture

We have made a conservative assessment of the potential consequences of tube failure in a steam generator releasing radioactivity contained in the primary coolant to the secondary steam system and thence to the atmosphere. For the analysis of this accident, we assumed that the primary system activity is at the maximum permitted in the Technical Specifications which is $68/\bar{E}$ (μc/cc-Mev). This corresponds roughly to the activity which would result from operation at the design basis of 1% clad defects in the fuel. The distribution of fission products in the coolant was assumed to be the same as that postulated by the applicant in Table 9.2-5 of the FSAR.

Assuming a double-ended rupture of a tube within the steam generator, the primary coolant system will leak into the secondary system. Since there are no isolation valves on the primary coolant system to isolate a failed loop, one-fourth of all the radioactivity contained in the primary coolant was assumed to be transferred to the secondary system. Further, assuming that the iodine concentration in the secondary system is at the limits permitted in Technical Specifications (0.5 μc/cc of I^{131}), releasing the entire amount of activity contained in a steam generator with an iodine reduction factor of ten provides conservative estimates of the potential dose. As indicated in Section 4.7, the resulting dose at the exclusion boundary was on the order of 10 CFR Part 20 limits (average yearly limits for normal operation).

4.4 Rupture of Steam Line

The steam lines for the Ginna plant are Class I piping systems up to and including the isolation valves. Breaks in the Class I steam lines inside the containment could correspond to a break size as large as 4.37 ft^2. Because of the flow measuring nozzles, breaks outside of the containment would be limited to 1.4 ft^2. A failure of a relief valve outside of the containment could also result in an uncontrolled blowdown of the steam generator. The applicant has analyzed the consequences of these failures and the results are presented in Section 14.2.5 of the FSAR.

To mitigate the consequences of postulated failures in the steam system, engineered safety feature trips as follows are provided:

1. Safety Injection signal upon any one of:

   a. simultaneous low pressurizer pressure and low pressurizer level,
   b. high containment pressure, or
   c. low pressure in any steam line;

2. Closure of feedwater pump discharge valves and tripping of the main feedwater pumps upon any safety injection signal;

3. Steam line isolation for each valve upon:

   a. high steam flow in that line coincident with any safety injection signal, or
   b. high containment pressure.

To evaluate the applicant's analyses we made a parametric study of the consequences of various postulated failures. We computed the rate of moderator cooldown for various assumed break sizes, feedwater flow rates, time of valve trips, steam generator heat transfer coefficients and reactor power. The results of our analyses agreed with the calculations of the applicant.

In calculating the potential offsite doses we assumed: (1) that the primary coolant activity was at the maximum allowed in the Technical Specifications, (2) a primary to secondary leak of 10 gpm existed, (3) the secondary system coolant activity was the maximum allowed in the Technical Specifications, (4) a partition factor of 10 was used for the release of iodine, and (5) the secondary system coolant from one steam generator was released.

For the steam line inside the containment, it was additionally assumed that as a consequence of the more severe power transient that 10% fuel failure occurs.

As indicated in Section 4.6, the potential offsite doses are on the order of 10 CFR Part 20 limits for yearly exposures.

4.5 Loss-of-Coolant Accident

In the Ginna plant, as in other pressurized water reactors, the Primary Coolant System is within the containment vessel. Auxiliary systems such as the Chemical and Volume Control System and the Waste Storage Systems are located outside of the containment and can be isolated by valves from the Primary Coolant System. The most severe accident considered for this facility is a rupture of one of the main coolant recirculation lines. In judging the suitability of the containment proposed for this site, we have evaluated an accident wherein a recirculation pipe in the Primary Coolant System is assumed to rupture, causing a complete loss of coolant. In addition, we assumed that 100% of the noble gases, 50% of the halogens, and 1% of the solid fission products in the total core inventory were instantaneously released into the containment volume. These assumptions are the same as those suggested in TID 14844. The iodine reduction factors used were those discussed in Section 3.4. The potential dose is presented in Section 4.6. As indicated, using the maximum leakage rate permitted in the Technical Specifications results in doses less than 10 CFR Part 100 guideline values.

4.6 Summary of Radiological Consequences

The following is a summary of the staff's calculations of the doses for the various postulated accidents:

Accident	Two-hour Dose at Site Boundary Whole Body (Rads)	Two-hour Dose at Site Boundary Thyroid (Rem)	Course-of-Accident Doses at LPZ Outer Boundary Whole Body (Rads)	Course-of-Accident Doses at LPZ Outer Boundary Thyroid (Rem)
1. Refueling Accident	4	1500	.5	180
2. Gas Storage Tank Rupture	3	neg.	0.3	neg.
3. Steam Generator Tube Rupture	0.5	1.5	neg.	neg.
4. Steam Line Rupture Inside Containment	neg.	6	neg.	0.6
5. Steam Line Rupture Outside Containment	neg.	1.5	neg.	neg.
6. Design basis of loss-of-coolant accident with iodine removal by spray and one filter	5	250	1.3	90

4.7 Conclusion

Based on our evaluation we have concluded that the consequences of all the design basis accidents except the fuel handling accident are within the guideline values given in 10 CFR Part 100. As indicated previously, the fuel handling accident will be reevaluated, and if necessary, the applicant has agreed to take corrective action to reduce the potential doses to within 10 CFR Part 100 guideline values.

5.0 Emergency Planning

The applicant has described a comprehensive plan for coping with the consequences of an accident which might affect the general public. Arrangements to deal with radiological emergencies have been made with the responsible agencies of the State of New York and appropriate local officials.

Members of the applicant's onsite staff will furnish information concerning release rates and will cooperate with state and local officials in providing technical advice concerning the potential offsite effects throughout the course of any accident affecting the general public, in accordance with prearranged plans. The applicant possesses the capability for providing offsite monitoring to supplement that provided by the State of New York.

In addition, technical assistance is available through the Radiological Emergency Assistance Team program of the AEC. The applicant has established liaison with the team at the New York Operations Office of the AEC.

Rochester Gas and Electric has made arrangements with two medical doctors trained in radiation medicine, and one of these doctors lives less than half a mile from the site. Strong Memorial Hospital in Rochester has agreed to provide medical care for the Ginna plant, and to make available such support as might be required in the event of an accident at the site, whether or not such an accident should involve the general public.

We have concluded that the arrangements made by the applicant to cope with the possible consequences of accidents at the site are both reasonable and prudent, and that there is adequate assurance that such arrangements will be satisfactorily implemented in the unlikely event that they are needed.

6.0 Conduct of Operations

Responsibility for safe operation of the plant is vested in the Plant Superintendent. He reports to the Division Superintendent, Electric System Planning and Operations, who reports to General Superintendent, Electric and Steam Operations, who, in turn, is responsible to the Vice President, Rochester Gas and Electric Corporation.

Within the onsite operating organization, responsibility for day-to-day operation of the facility rests with the Operations Engineer, reporting to the Plant Superintendent. The Operations Engineer will be a licensed senior reactor operator, as will each Shift Foreman. The shift operating crew will consist of a Head Control Operator, a Control Operator, each of whom will be a licensed reactor operator, and two unlicensed Auxiliary Operators, all under the supervision of the Shift Foreman.

The qualifications of individuals initially proposed to fill professional and semi-professional positions in the onsite operating organization have been described in the Safety Analysis Report. The minimum qualifications for these functional positions are described in the Technical Specifications. We have examined the qualifications of the incumbents and subject to satisfactory completion of necessary examinations for appropriate licenses we conclude that the professional staff is technically competent to operate the facility.

Engineering support to the operating organization will be provided by the Engineering Analysis group, and by Westinghouse and specialist consultant firms. The engineering staff of Rochester Gas and Electric (RG&E) is familiar with the plant, and is capable of handling the preparation and review of design changes and plant modifications originating at the Ginna plant. In addition, the applicant has demonstrated his intent to utilize, as necessary, the services of consultants to augment the nuclear capability of the RG&E staff. Westinghouse will be an active participant in the startup and initial operation of the plant, and will continue to make available direct technical support to the RG&E staff throughout the operating lifetime of the facility. On these bases, we conclude that adequate engineering capability will be available through the RG&E staff and specialist consultants to support the applicant's operating staff.

The applicant proposes to use what has become a relatively conventional two-level committee structure to perform review and audit of plant operation. The first of these committees, the Plant Operations Review Committee, which comprises the senior members of the onsite staff, acts in an advisory capacity to the Plant Superintendent. Independent audit of plant operation is provided by the Review and Audit Committee, the Chairman and Vice Chairman of which are appointed by name by the chief executive officer of the company. The responsibilities and authorities for these committees are delineated in the Technical Specifications. We conclude that the review and audit structure proposed by the applicant is satisfactory.

Based on the above considerations, we conclude that the applicant is technically qualified to operate the plant and has established effective means for continuing review, evaluation, and improvement of plant operational safety.

7.0 Technical Specifications

The applicant's proposed Technical Specifications to the license for Ginna are presented in Amendment No. 14. Included are sections covering safety limits and limiting safety system settings, limiting conditions for operation, surveillance requirements, design features and administrative controls.

We have reviewed these proposed Technical Specifications in detail and have held numerous meetings with the applicant to discuss their contents. Some modifications to the proposed Technical Specifications submitted by the applicant were made to more clearly describe the allowed conditions for plant operation. Based upon our review, we conclude that normal plant operation within the limits prescribed in the Technical Specifications will not result in potential doses in excess of Part 20 limits. Furthermore, the limiting conditions of operation and surveillance requirements will assure that necessary engineered safety features will be available in the event of malfunctions within the plant.

8.0 Report of Advisory Committee on Reactor Safeguards

The Advisory Committee on Reactor Safeguards (ACRS) has reviewed the application for a provisional operating license for the Robert Emmett Ginna Nuclear Power Plant Unit No. 1. The Committee completed its review of the facility at the 109th meeting held during May 8-10, 1969. A copy of the report of the ACRS, dated May 15, 1969, is attached.

The ACRS, in its letter, made several recommendations and noted several items to be resolved by the applicant and the Staff either before plant operation or on an acceptable time scale subsequent to initial operation. These items are discussed in the following paragraphs.

8.1 Flood Level Protection

The applicant is reexamining his estimate of the appropriate design flood level, including still water level, wave action, and wave runup. He has agreed to provide plant protection consistent with the flood level estimates by our consultants. Furthermore, he has agreed to provide temporary protection if needed, until permanent protection requirements are met.

8.2 Installation of an Accelerograph

The ACRS recommends that at least one strong-motion accelerograph be installed, pointing out that a strong-motion accelerograph could minimize the possibility of a lengthy shutdown for inspection in the event that a significant earthquake of otherwise undetermined intensity at the site should occur. The applicant has agreed to install one strong motion accelerograph of a type and at a location to be determined with the staff in advance of its installation. The applicant states that the installation will be completed at/or before the time of the first refueling.

8.3 Capability to Monitor and Alarm Abnormal Power Distribution

The high thermal performance demanded of the fuel in the Ginna reactor, and the potential for axial xenon oscillations, requires that the spatial power distribution in the reactor core and the positions of the control rods be dependably known. In the proposed design, all alarms related to control-rod malpositioning were derived from the on-line computer. The ACRS stated that "good information regarding possible anomalies in the power distribution is important, and that, as a minimum, the power should be reduced appropriately or adequate alternative measures should be taken, when the computer is inoperative."

The applicant has agreed to equip the Ginna facility with a system, independent of the on-line process computer, whose function will be to monitor and alarm any condition of abnormal power distribution of sufficient magnitude to lead to a possible violation of a safety limit. The signal actuating this monitor and alarm will be derived from a comparison of the output current of each individual external ion chamber in either the top or bottom array with the average output currents obtained from its companion ion chambers in that array. A deviation of predetermined magnitude will actuate the alarm. The limits of the provisions made for detection by two independent means of an abnormal power distribution (the in-line computer and the proposed system) from malpositioned rods is included in the Technical Specifications.

8.4 Fuel Handling Accident

As discussed in Section 4.1 we are not in agreement with the applicant on the radioactivity that might be released and the offsite dose that could result from dropping a spent fuel assembly in the storage pit. The applicant has agreed that if the differences between its model and ours cannot be reconciled, the applicant will use our model in order to determine the amount of radioactivity and offsite doses which could result from a broken fuel assembly in the storage pit. The applicant has also agreed to take corrective measures, if necessary to meet dose guideline values of 10 CFR Part 100. Furthermore, the applicant states that it will not handle irradiated fuel until this problem has been resolved.

8.5 Nil Ductility of the Reactor Vessel

The Committee noted that the reactor pressure vessel wall will be exposed to a fairly large fast neutron fluence over the reactor life, which will lead to a sizeable increase in the nil ductility transition temperature and to some degradation in the fracture toughness properties. The ACRS recommends that prior to the accumulation of a peak fluence of $10^{19} n/cm^2$ (approximately ten years of operation), the staff reevaluate the continued suitability of the currently proposed reactor vessel startup, cooldown and operating conditions, as well as the assurance of vessel integrity despite thermal shock in the unlikely event of a loss-of-coolant accident. We will implement this recommendation.

8.6 Primary System Inspection

The Committee suggests that consideration be given to a program of monitoring the pressure vessel and other parts of the primary system for signs of excessive internal vibration or structural damage during the service life of the plant. Regular inspection intervals of the pressure vessel and the primary system is covered in the Technical Specifications.

8.7 Containment Spray Compatibility

The applicant documented its results of the containment spray solution compatibility studies with the exposed materials in the containment. We have completed our review of the material submitted and conclude that the results are acceptable.

8.8 Hydrogen Generation

The applicant has installed two hydrogen recombiners in the Ginna containment vessel. We have not established the need for recombiners in the event of a loss-of-coolant accident. We will continue our review of this item with the applicant as well as with the supplier (Westinghouse).

8.9 Piping Analysis

Class I piping of the Ginna plant was analyzed as a one-mass system. We are of the opinion that a one-mass system is not a generally acceptable method for analyzing such systems; however, we recognize that for the majority of piping systems, use of this method may result in an acceptable design. The applicant has agreed to supplement the analysis of the response of Class I piping to seismic loading. This additional evaluation will include a tabulation of calculated piping stresses at the most critical Class I piping locations and the corresponding allowable stresses; a tabulation of the most critical Class I piping support loads, including maximum seismic loads, a description of the type and location of each support analyzed, its design and ultimate capacity, and the seismic amplification factor associated with the location of the support in the building; and an evaluation of the results of these tabulations. Furthermore, an onsite inspection of this piping and its supports will be carried out by qualified personnel. The applicant states that as a result of the evaluation and inspection, any modifications that may be needed will be completed on or before July 31, 1969. Until this information is submitted and reviewed to our satisfaction, reactor power will be limited to 5 Mw(t).

8.10 Conclusion

The applicant has agreed that the recommendations of the ACRS will be carried out. We will implement the recommendations of the ACRS on all of the foregoing matters during operation of the facility under the eighteen-month term of the provisional operating license. The ACRS concluded in its letter that if due regard is given to the foregoing, the Ginna Unit No. 1 can be operated at power levels up to 1300 Mwt without undue hazard to the health and safety of the public.

9.0 Common Defense and Security

The application reflects that the activities to be conducted would be within the jurisdiction of the United States and that all of the directors and principal officers of the applicant are American citizens. The applicant is not owned, dominated or controlled by an alien, a foreign corporation or a foreign government. The activities to be conducted do not involve any restricted data, but the applicant has agreed to safeguard any such data which might become involved in accordance with the regulatory requirements. The applicant will rely upon obtaining fuel as it is needed from sources of supply available for civilian purposes, so that no diversion of special nuclear material to military purposes is involved. For these reasons and in the absence of any information to the contrary, we have found that the activities to be performed will not be inimical to the common defense and security.

10.0 Conclusion

Based upon our review of the application as presented and discussed in this evaluation and the report of the Advisory Committee on Reactor Safeguards, we have concluded that the Robert Emmett Ginna Nuclear Power Plant Unit No. 1 can be operated as proposed in the Provisional Operating License without endangering the health and safety of the public.

Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

May 15, 1969

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON ROBERT EMMETT GINNA NUCLEAR POWER PLANT UNIT NO. 1

Dear Dr. Seaborg:

During its 109th meeting, May 8-10, 1969, the Advisory Committee on Reactor Safeguards completed its review of the application by the Rochester Gas and Electric Corporation for a license to operate the Robert Emmett Ginna Nuclear Power Plant Unit No. 1 at power levels up to 1300 MWt. The Committee had previously met with the applicant during its 103rd meeting, October 31 to November 2, 1968, to review an important change in the design of the large penetrations of the containment, and again during its 108th meeting, April 10-12, 1969, for a partial review of the application. During the review, Subcommittee meetings were held on October 24, 1968 (at the site); January 23, 1969; March 5, 1969; and May 1, 1969. In the course of the review, the Committee had the benefit of discussions with representatives of the applicant, the Westinghouse Electric Corporation, Gilbert Associates, Inc., and their consultants; of discussions with the AEC Regulatory Staff and its consultants; and of the documents listed. The Committee reported to you on the construction permit application for this plant on March 18, 1966.

The reactor primary fluid system, containment, and engineered safety features all incorporate important developments from the design of previously licensed pressurized water reactors. The developments reflect both economic and safety considerations, and the plant represents the first of the line of Westinghouse reactors currently being licensed for construction.

The applicant is re-examining his estimate of the appropriate design flood level, including still water level, wave action, and wave runup. In the event of disagreement with the AEC Regulatory Staff, he will assure plant protection consistent with the flood estimates by the Staff consultants.

The applicant has agreed to install a strong-motion accelerograph if considered necessary. The Committee believes that at least one strong-motion accelerograph should be installed and, in addition, wishes to point out that a strong-motion accelerograph could minimize the possibility of a lengthy shutdown for inspection in the event that a significant earthquake of otherwise undetermined intensity at the site should occur.

The high thermal performance demanded of the fuel in the Ginna reactor, and the potential for axial xenon oscillations, requires that the spatial power distribution in the reactor core and the positions of the control rods be dependably known. In the proposed design all alarms related to control-rod malpositioning are derived from the on-line computer. The Committee believes good information regarding possible anomalies in the power distribution is important and that, as a minimum, the power should be reduced appropriately, or adequate alternative measures should be taken, when the computer is inoperative.

The applicant and the Regulatory Staff are not in agreement on the radioactivity that might be released and the off-site dose that could result from dropping a spent fuel assembly in the storage pit. The applicant will attempt to reconcile the disagreement but, if necessary, will take corrective measures to satisfy safety criteria in accordance with the Staff model for this postulated accident. The applicant will not handle irradiated fuel until this matter is resolved.

The applicant calculates that the reactor pressure vessel wall will be exposed to a fairly large fast neutron fluence (about 3.7×10^{19}) over the reactor life. This will lead to a sizeable increase in the nil ductility transition temperature and to some degradation in fracture toughness properties. Prior to the accumulation of a peak fluence of 10^{19}, the Regulatory Staff should reevaluate the continued suitability of the currently proposed reactor vessel startup, cooldown and operating conditions, as well as the assurance of vessel integrity despite thermal shock in the unlikely event of a loss-of-coolant accident.

The Committee understands that the applicant is providing means for preoperational monitoring of the pressure vessel and other parts of the primary system for signs of excessive internal vibration or structural damage. The Committee believes the applicant should give consideration to a program of monitoring during the service life of the plant.

The Committee believes the applicant's proposal of an in-service inspection program for the reactor pressure vessel and other portions of the primary system covering the first five years of operation, with a commitment to review the program after that period in the light of then-existing inspection technology, is satisfactory. The applicant has modeled his inspection program on the draft USA code dealing with in-service inspection; the Committee concurs in this approach.

Several Westinghouse reports pertinent to Ginna and other Westinghouse reactors have recently been received and others are expected. Some matters relating to Ginna consequently remain to be resolved by the Staff either before plant operation or on an acceptable time scale subsequent to initial operation. These matters include assurance of long-term compatibility of the containment spray solution with the exposed materials in the containment and verification of the performance of the hydrogen recombiners that may be necessary in the unlikely event of a loss-of-coolant accident; evaluation of the probability and consequence of systematic instrument failures. A more detailed analysis of the dynamic response of a portion of the system piping to an earthquake is also being prepared by the applicant for review by the Staff. The Committee believes that these matters will be resolved satisfactorily.

The Advisory Committee on Reactor Safeguards believes that, if due regard is given to the items mentioned above, the Robert Emmett Ginna Nuclear Power Plant Unit No. 1 can be operated at power levels up to 1300 MWt without undue risk to the health and safety of the public.

Additional remarks of Dr. David Okrent are attached.

Sincerely yours,

Stephen H. Hanauer

Stephen H. Hanauer
Chairman

Attachments:
1. Additional Remarks of
   Dr. David Okrent
2. References

Dr. David Okrent makes the following additional remarks:

"In view of the great importance of pressure vessel integrity to the health and safety of the public, I believe that for welds in the pressure vessel wall that will receive a large integrated fast neutron irradiation over the reactor life it would be prudent for the applicant to commit himself to a more thorough and extensive in-service, non-destructive, volumetric testing program by such means as are or become practical. In particular, within the framework of currently anticipated technology, I would recommend a commitment to 100% ultrasonic inspection of such a weld every ten years. Consideration should also be given to non-destructive, volumetric inspection or monitoring of those steel forgings making up the vessel wall that will be highly irradiated."

References - Robert Emmett Ginna Nuclear Power Plant Unit No. 1

1. Preliminary Facility Description and Safety Analysis Report, Volume 1, Appendices.

2. Amendments 6 - 17 and Amendment 19 to Application for Licenses.