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MAY 29 1969

Docket No. 50-220

Niagara Mohawk Power Corporation
300 Erie Boulevard West
Syracuse, New York 13202

Attention: Mr. Minot H. Pratt
Vice President and
Executive Engineer

Gentlemen:

With my letter to you dated April 23, 1969, you were sent a copy of a report from the Advisory Committee on Reactor Safeguards concerning the Committee's review of your application for a license to operate the Nine Mile Point Nuclear Station at power levels up to 1538 thermal megawatts.

In view of the Committee's report, 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 Niagara Mohawk Power Corporation for operation of the Nine Mile Point Nuclear Station at thermal power levels not to exceed 1538 megawatts. A copy of the notice is enclosed, together with a copy of a related safety evaluation prepared by the Division of Reactor Licensing.

Sincerely yours,

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

PDR Cy. sent directly to
S. Robinson By DRL,
PER H. Steele

Enclosures:
Fed. Reg. Notice
DRL Safety Evaluation

bcc: J. R. Buchanan, ORNL
A. A. Wells, ASLBP

cc: George F. Trowbridge, Esquire
Shaw, Pittman, Potts, Trowbridge & Madden

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UNITED STATES ATOMIC ENERGY COMMISSION

DOCKET NO. 50-220

NIAGARA MOHAWK POWER 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 Niagara Mohawk Power Corporation (Niagara Mohawk) to possess, use, and operate the Nine Mile Point Nuclear Station, a single cycle, forced circulation, boiling water reactor. The reactor is located on the approximately 1500-acre Nine Mile Point site on the southeast shore of Lake Ontario in the Town of Scriba, Oswego County, New York, about seven miles northeast of the City of Oswego. The proposed license would authorize Niagara Mohawk to operate the Nine Mile Point Nuclear Station at power levels not to exceed 1538 megawatts (thermal) 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-16 issued by the Commission on April 12, 1965. Upon issuance of the provisional operating license, Niagara Mohawk 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 regulations (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 (application Amendments No. 2 through 13) filed during the period of May 29, 1967 through April 10, 1969, (2) the report of the Advisory Committee on Reactor Safeguards dated April 17, 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

Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Dated at Bethesda, Maryland
this 29th day of May, 1969.

NIAGARA MOHAWK POWER CORPORATION

DOCKET NO. 50-220

PROPOSED PROVISIONAL OPERATING LICENSE

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

a. The application for provisional operating license (application Amendments Nos. 2 through 13, dated May 29, 1967, July 14, 1967, September 6, 1967, May 16, 1968, September 27, 1968, October 14, 1968, November 4, 1968, January 17, 1969, January 17, 1969, March 10, 1969, March 28, 1969 and April 10, 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-16;

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 the facility can be operated at power levels not in excess of 1538 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 Niagara Mohawk Power Corporation (Niagara Mohawk), as follows:

1. This license applies to the Nine Mile Point Nuclear Station, a single cycle, forced circulation, boiling light water reactor, and electric generating equipment (the facility). The facility is located on the Nine Mile Point site on the southeast shore of Lake Ontario in Oswego County, New York, approximately seven miles northeast of the City of Oswego and thirty-six miles northwest of Syracuse, and is described in license application Amendment No. 2, "Final Safety Analysis Report," as supplemented and amended (Amendments Nos. 3 through 13).

2. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses Niagara Mohawk:

   A. Pursuant to Section 104b of the Atomic Energy Act of 1954, as amended (the Act), and Title 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 the Nine Mile Point site;

   B. Pursuant to the Act and Title 10, CFR, Part 70, "Special Nuclear Material," to receive, possess and use at any one time up to 3800 kilograms of contained uranium 235 in connection with operation of the facility;

C. Pursuant to the Act and Title 10, CFR, Part 30, "Rules of General Applicability to Licensing of Byproduct Material," to receive, possess, and use in connection with operation of the facility 24 curies of Cobalt 60 as a sealed source; 430 millicuries of Cobalt 60 as two sealed sources of not more than 400 millicuries and 30 millicuries each; 500 microcuries of Cobalt 60 as five sealed sources not to exceed 100 microcuries each; 101 millicuries Cobalt 60, 3 millicuries Strontium 90, 101 millicuries Iodine 131, 102 millicuries Cesium 137, and 13 millicuries of any byproduct material with Atomic Nos. between 3 and 83, inclusive, in any chemical and/or physical form; 12,500 curies Antimony 122-124 as five sealed sources not to exceed 2,500 curies each; and six curies Americium 241 as a sealed 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, Section 30.34 of Part 30, Section 40.41 of Part 40, Sections 50.54 and 50.59 of Part 50, and Section 70.32 of Part 70, and is subject to the additional conditions specified below:

   A. Maximum Power Level

   Niagara Mohawk is authorized to operate the facility at steady state power levels up to a maximum of 1538 megawatts thermal.

B. Technical Specifications

The Technical Specifications contained in Appendix A[1/] attached hereto are hereby incorporated in this license. Niagara Mohawk shall operate the facility at power levels not in excess of 1538 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 Section 50.59 of 10 CFR Part 50.

C. Reports

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

(1) Niagara Mohawk 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 as described in the Technical Specifications. For each such occurrence, Niagara Mohawk shall promptly notify by telephone or telegram the appropriate Atomic Energy Commission Regional Compliance Office listed in Appendix D of 10 CFR 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) Niagara Mohawk 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

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 Commission.

from performance specifications contained in the Final Safety Analysis Report (safety analysis report) or the Technical Specifications.

(3) Niagara Mohawk 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), Niagara Mohawk shall begin submitting reports in writing in accordance with the requirements of the Technical Specifications.

D. Records

Niagara Mohawk shall keep facility operating records in accordance with the requirements of the Technical Specifications.

4. Pursuant to Section 50.60, Title 10, CFR, Part 50, the Commission has allocated to Niagara Mohawk for use in the operation of the facility 14,321 kilograms of uranium 235 contained in uranium in the isotopic ratios specified in the application. Estimated schedules of special nuclear material transfers to Niagara Mohawk and returns to the Commission are contained in Appendix B[1/] which is attached hereto. Transfers by the Commission to Niagara Mohawk in accordance with column 2 in Appendix B will be conditioned upon Niagara Mohawk's return to the

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.

Commission of material substantially in accordance with column 3 (including the subcolumns headed "Scrap" and "Depleted Fuel").

5. This license is effective as of the date of issuance and shall expire eighteen (18) months from said date, unless extended for good cause shown, or upon the earlier issuance of a superseding operating license.

FOR THE ATOMIC ENERGY COMMISSION

Peter A. Morris, Director
Division of Reactor Licensing

May 26, 1969

SAFETY EVALUATION

BY THE

DIVISION OF REACTOR LICENSING

U.S. ATOMIC ENERGY COMMISSION

IN THE MATTER OF

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR STATION

DOCKET NO. 50-220

TABLE OF CONTENTS

1.0 INTRODUCTION

The Niagara Mohawk Power Corporation (applicant) submitted Amendment No. 2, dated May 29, 1967, to its application requesting a provisional operating license for the Nine Mile Point Nuclear Station (NMP, facility, plant). The facility, which will utilize a single cycle, forced circulation General Electric boiling water reactor (BWR) has been under construction since issuance of a construction permit on April 12, 1965 by the Commission. It is located on a 1500-acre site on the southeast shore of Lake Ontario in the Town of Scriba, Oswego County, New York, seven miles northeast of the City of Oswego and 36 miles northwest of the City of Syracuse.

The technical safety review of the design of the facility has been based on Amendment Nos. 2 through 13. All of these documents are available for review at the Atomic Energy Commission's Public Document Room, 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 the applicant and us to discuss the facility. The ACRS report on NMP dated April 17, 1969, is attached to this safety evaluation.

Our evaluation of overall facility performance was based on a thermal power level of 1538 megawatts (Mwt) which will be the licensed power level. However, because the plant is designed for ultimate power operation at 1779 Mwt, we reviewed the capability of the plant engineered safety features and the radiological consequences of accidents at the ultimate power level of 1779 Mwt. Before any increase in power level in excess of 1538 Mwt can be permitted, the applicant must submit a request, including a complete technical evaluation in support of the request, for the license amendment. The information would be reviewed by the staff and the ACRS before any increase in power would be authorized.

Based upon our evaluation of the facility as presented in subsequent sections, we have concluded that the Nine Mile Point Nuclear Station can be operated as proposed without endangering the health and safety of the public.

2.0 SITE AND ENVIRONMENT

2.1 Site Description

The Nine Mile Point site consists of approximately 1500 acres located on the shore of Lake Ontario about 5.5 miles northeast of the nearest boundary of Oswego, New York. The general site and adjacent areas are characterized by gently rolling hills which gradually increase in elevation with distance from the lake. The surrounding land is used principally for agriculture, and the population density is low with the nearest permanent residence about a mile southwest of the station.

The shortest distance from the reactor to the site boundary is about 4000 feet, which has been established as the outer boundary to the 1500-acre exclusion area. The applicant proposed that the distance to the nearest substantial residential area in the City of Oswego be the low population zone radius, a distance of about 5.5 miles from the reactor. However, in order to assure that no area within the boundary of the City of Oswego lies within the low population zone, we have concluded that a radius of 4 miles for the low population zone is acceptable for this facility. The total population within this distance, including summer visitors, is approximately 2000.

2.2 Meteorology

Meteorological data were taken with a 200-foot tower onsite during 1963 and 1964. These data have been analyzed to obtain frequency distributions of stability, wind direction, and wind speed as a function of time of day and months of the year. From these analyses, the sea breeze inversions caused by movement of air over the cold lake during April, May, and June are clearly evident, as are similarly induced lapse conditions during November and December. We have also considered the effects of wind loadings on plant shutdown capability. Sufficient onsite meteorological data have been accumulated at the site to derive limits for routine gaseous releases and to justify the conservatism of the meteorological assumptions used in the accident analyses described in Section 6.0.

2.3 Limnology

Flood protection is provided so that the plant can be safely shut down for a flooding level as high as approximately 261 feet above mean sea level (MSL). The maximum flood height recorded applicable to this site is 249.3 feet above MSL.

All radioactive liquid effluents discharged from the Nine Mile Point site will be less than 10 CFR Part 20 limits. The applicant has in addition, however, conducted extensive tests to determine limnological characteristics of Lake Ontario in the vicinity of the plant. Results of these tests were used to demonstrate that a minimum short-term dilution factor of approximately three would occur at any point along the lake shore. It was also shown that plant effluents would not be carried to the same area of the lake more than 10% to 15% of the time. We and our consultant, the U. S. Geological Survey (Radio-Hydrology Branch), have concluded that there are no limnological conditions requiring special attention for this facility.

2.4 Geology and Seismology

Data obtained from borings made since the construction permit was issued have confirmed the original conclusions regarding the acceptability of this site. The general nature of the rock at all depths is very sound, with no evidence of faults or connected joints.

The facility was designed to withstand the effects of an earthquake corresponding to a maximum horizontal ground acceleration of 0.11g. Facility components and structures are designed such that the loads caused by an earthquake of this magnitude in combination with operating loads would not exceed code allowable stresses. We and our consultant, Dr. Newmark (Nathan M. Newmark Consulting Engineering Services), have reviewed the seismic design of the facility and found it acceptable.

We will require the applicant to install a strong-motion seismograph to record data related to ground motion during a seismic event at the site. These data would be employed in the subsequent evaluation of the effects of the seismic event on the safe operation of the facility.

2.5 Environmental Considerations

Preoperational environmental monitoring has been conducted at this site to determine background conditions. The applicant will continue to conduct an environmental radiation monitoring program in order to determine the effect of operations at this facility. The operational monitoring program will include measurement of atmospheric radioactivity, fallout, domestic water, surface water, aquatic biota, and foodstuffs. 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 program proposed by the applicant is adequate with respect to monitoring the radiological effects of plant operation on the environs.

3.0 FACILITY DESIGN

3.1 Reactor Core

3.1.1 General

The reactor is a single cycle, forced circulation, boiling water reactor producing steam for direct use in the steam turbine. The Nine Mile Point (NMP) facility is similar to the recently approved Oyster Creek facility (Docket No. 50-219). The core containing the reactor fuel is located within a cylindrical shroud inside the reactor vessel. Water, which serves as both the moderator and coolant, enters the bottom of the reactor core, and flows upward through the fuel assemblies where boiling produces steam. The steam-water mixture is separated by steam separators and dryers mounted on the shroud. The separated water mixes with the incoming feedwater in an annulus formed by the shroud and the wall of the reactor vessel and is returned to the core inlet via five external recirculation pumps. The steam is passed through the dryers to the turbine-generator for the production of electricity.

3.1.2 Mechanical Design

The overall active height of the core is 12 feet and the equivalent diameter is 12.6 feet. The reactor core will consist of 532 fuel assemblies each of which contains 49 cylindrical fuel rods in a 7 x 7 square array. Each fuel rod consists of compacted and sintered uranium dioxide pellets enclosed in zircaloy tubes (cladding). The tubes are sealed by zircaloy plugs welded into each end. A fuel rod is approximately one-half inch in diameter and 12 feet long.

Four fuel assemblies rest on a support casting mounted on top of each control rod guide tube. Each guide tube, with its fuel support casting, rests on a control rod drive housing. The housing is welded to a stub tube which in turn is welded to the bottom head of the reactor pressure vessel.

Control of the reactor to accommodate fuel burnup and fission product poisoning and to shut the reactor down is accomplished by control rods. The 129 control rods are cruciform-shaped, enter the reactor core from the bottom, and are manipulated by independent mechanisms. Each control rod contains stainless steel tubes filled with compacted boron carbide powder which is a neutron absorbing medium. The tubes are held in a cruciform array by a stainless steel sheath that extends the full length of the control rod. In addition to the control rods, 232 temporary control curtains which are fixed in the core are used to compensate for the excess reactivity change between initial and equilibrium cores. The curtains are made of boron-stainless steel sheets and are located in the spaces between the fuel channels. Spaces between the channels also contain in-core instrumentation and neutron sources necessary for plant operation.

On the basis of our review, we have concluded that the core design features for the NMP facility are adequate.

3.1.3 Thermal and Hydraulic Design

Operation of the reactor at 1538 Mwt with rated recirculation flow results in core thermal and hydraulic conditions which are similar to those of currently operating BWR's. The Big Rock Point reactor (Docket No. 50-155) has operated at average heat fluxes and primary coolant system flow rates which are about the same as NMP. The Dresden 1 reactor (Docket No. 50-10) has been run with exit steam void fractions and steam quality comparable with those expected in NMP.

Recently the Gundremmingen (KRB) Nuclear Power Station (General Electric BWR, similar in design to NMP) has been placed in operation in Germany at the design power level of 801 Mwt. A preliminary review of the results of the accumulated operating data, presented in the General Electric report, APED-5698, indicates satisfactory performance.

We have reviewed the analyses of normal plant operation as well as the various transients that can be expected to occur during the operating lifetime of the plant. Transients can be induced by control rod withdrawals, changes in the recirculation flow rate, addition of cold water and change in system pressure. For all of the transients reviewed, the minimum critical heat flux ratio (MCHFR) remains well above unity, which is the assumed fuel rod damage limit. The limiting transient that would affect local regions of the core was found to result from a control rod withdrawal until stopped by a rod-block system. For this case, the calculated MCHFR remains above 1.2 using the critical heat flux data given in the General Electric report, APED-3892, "Burnout Limit Curves for Boiling Water Reactors." For other transients reviewed wherein the entire core is affected, the MCHFR remains above 1.8. Based on our review of the various transients and the plant protection system, we conclude that an adequate margin against fuel rod cladding damage is available in the facility.

3.1.4 Reactivity Control

Reactor power can be controlled by either movement of control rods or variation in reactor coolant recirculation system flow rate. A standby liquid control system is also provided as a backup shutdown system. These aspects, as well as certain other plant features related to reactivity control, are discussed below.

Control rods are used to bring the reactor through the full range of power (from shutdown to full power operation), to shape the reactor power distribution, and to compensate for changes in reactivity due to fuel burnup. There are 129 individual control rod drives and hydraulic control systems. Each drive has separate control and scram devices. A common hydraulic pressure source for normal operation and a common dump volume for scram operation are used for the drives.

On the basis of our review of the drive system design and the supporting evidence accumulated from operation of similar systems in other reactors, we conclude that the installed system will meet the functional performance requirements for the facility in a safe manner.

Control rod worths at power levels below 10% of rated power (1538 Mwt) are limited by the rod worth minimizer (RWM), a device which utilizes a computer to restrict control rod patterns such that rods which are moved are worth no more than 1% Δk, and that no control rod worth will exceed 2-1/2% Δk. The inputs to the computer are pre-selected control rod drive patterns and current control-rod-drive mechanism positions. The outputs consist of alarms and rod-block signals when the safe rod sequence (one of two stored in the computer and selected by the operator) is not followed. On the basis of our review, we conclude that the RWM serves a useful role in assuring that the control rod worths will not become so large as to result in serious damage in the event of a control-rod-drop accident.

At reactor power levels above 10%, the applicant does not intend to use the control rod worth minimizer to limit rod worths although it may do so. The maximum control rod worth that could be established for reactor power levels in excess of 10% is 3.8%Δk. Calculations of the consequences of a control-rod-drop accident where a control rod is assumed to fall by gravity from the core region with a rod worth of 3.8%Δk and reactor power in excess of 10% indicate that the peak fuel enthalpy is less than 200 cal/gm which is less than the enthalpy required for incipient fuel melting for the NMP fuel, i.e., 220 cal/gm. Accordingly, we conclude that use of the RWM at power levels above 10% is not required.

A control rod ejection accident is precluded by the control rod housing support structure located below the reactor pressure vessel. This structure serves to limit the distance that a ruptured control rod drive housing could be displaced to no more than three inches. The applicant indicates, and we agree, that control rod displacement of this magnitude would not introduce sufficient reactivity to the core to cause fuel rod failure.

With a given control rod pattern, control of the reactor can also be accomplished by varying the recirculation flow rate which causes a change in the void content in the core and a resultant change in reactor power. The applicant has not proposed to operate the plant initially on automatic flow control; therefore, we have not evaluated the automatic aspects of such operation. If this mode of operation is proposed in the future, it will be evaluated at that time.

The standby liquid control system is designed to bring the reactor to a cold shutdown condition from the full power steady state operating condition at any time in core life independent of the control rod system capabilities. This requires about 14%Δk of shutdown reactivity worth. The liquid control system is designed to inject sufficient sodium pentaborate to provide 17%Δk of negative reactivity, thus a shutdown margin of about 3%Δk is available. The injection rate of the system is adequate to compensate for the effects of xenon burnup.

3.2 Primary Coolant System

The primary coolant system includes the reactor pressure vessel, recirculation loops, relief valves, safety valves and the emergency isolation condenser system. An in-service inspection program for the primary coolant system, including portions of the main steam lines outboard of the isolation valves, as described in the Technical Specifications, has been developed for initial plant operation. As noted in the ACRS letter, Niagara Mohawk will review the program with us after five years of reactor operation, and modify it as necessary based on experience gained during operation. We conclude that the in-service inspection program, with provision for continuing review, is adequate for this plant.

3.2.1 Reactor Pressure Vessel

The NMP reactor vessel is made of high strength alloy carbon steel SA-302, Grade B and was designed for a pressure of 1250 psig and 575°F. The reactor vessel was fabricated, inspected, and tested in accordance with the ASME Boiler and Pressure Vessel Code Section I Power Boilers, 1962 Edition, plus the Nuclear Code Cases in effect in December 1963 (when the vessel was purchased). Furthermore, the vessel manufacturer (Combustion Engineering) was directed by the buyer's (General Electric) purchase specification to use Section VIII of the Code for Unfired Pressure Vessels where Section I Power Boilers did not cover specific details.

The applicant analyzed the reactor system including internals to assess the vibration integrity of the system and found them to be acceptable. The applicant also concluded, and we agree, that no vibration measurements need be performed at the NMP facility since they will be made at the Oyster Creek facility. If necessary, however, the tests can be performed at the NMP facility. In addition, the applicant has stated, as noted in the ACRS letter, that it will study and implement to the extent practical, means of instrumenting and monitoring for vibration or the presence of loose parts in the reactor pressure vessel. The intent of this program is to provide a method of monitoring for excessive vibration while the reactor is in operation.

The NMP reactor vessel contains a number of components manufactured from type 304 stainless steel. During fabrication of the vessel, these parts were furnace sensitized. A detailed and comprehensive program was initiated to examine these components for evidence of intergranular attack (as was detected in the Oyster Creek reactor vessel). Results of this program demonstrated that the 304 stainless steel furnace-sensitized components are free of intergranular attack. During the course of the examination of the furnace-sensitized components, certain field welds were found to contain porosity and/or lack of fusion. An appropriate repair program was devised and implemented to restore these welds to an acceptable condition.

As noted in the ACRS letter, the applicant will install appropriate corrosion test specimens within the reactor vessel. These specimens will be used to provide an early indication of intergranular attack in the unlikely event it occurs. The inspection schedule for these specimens is included in the Technical Specifications.

We have concluded that the inspection and repair program has been adequate and that the pressure vessel is acceptable. Furthermore, we conclude that the corrosion specimens testing program, in conjunction with the primary system surveillance program, will provide adequate means to monitor the status of the reactor pressure vessel throughout its operating lifetime.

3.2.3 Recirculation Piping

Each of the five reactor water recirculation loops contains a motor-driven recirculation pump and motor-operated gate valve for pump isolation and maintenance. The recirculation loop piping is designed for a pressure of 1200 psig and a temperature of 569°F, the recirculation pump casings are designed for a pressure of 1300 psig and a temperature of 575°F and the gate valves are designed for a pressure of 1200 psig and a temperature of 569°F. The recirculation loop piping is of welded construction and has been designed, built and constructed to meet the requirements of ASME Code, Section I and ASA-B31.1 Code for Pressure Piping.

The maximum operating loads included the design pressure and temperature, weight of piping, contents and insulation, as well as the effect of supports and other sustained external loadings including seismic conditions. The stress limits used by the applicant for assumed load combinations are reasonable and in our judgment the recirculation loop piping will have adequate integrity to safely withstand these loads.

3.2.4 Emergency Condensers

The isolation condensers which are designed to Class I standards provide a natural circulation heat sink in case of reactor isolation from the main condenser. The emergency condensers are located outside of the primary containment, but inside the concrete and metal-sided reactor building. The secondary side of each condenser contains enough inventory to remove decay heat for the first 1-1/2 hours after reactor pressure vessel isolation. Makeup to the secondary side for continued heat removal is achieved either by elevated water storage tanks or by a diesel-driven fire pump. The tube sides of the condenser are exposed to reactor pressure vessel pressure during operation and are designed for a pressure of 1250 psig and a temperature of 575°F. We conclude that this system is acceptable.

3.2.5 Relief and Safety Valves

The reactor coolant system safety and relief valves are located inside the containment. The safety valves, mounted on the reactor vessel head, are designed and sized according to the ASME Boiler and Pressure Vessel Code, Section I. A total of 15 safety valves are provided and are capable of preventing the overpressurization of the system which would result from a turbine trip without benefit of a reactor scram (at 1538 Mwt). There are six relief valves provided in the design. Four of the relief valves are adequate to prevent actuation of the safety valves in the event of a turbine trip with a failure of the bypass system, but assuming the reactor does scram. Further aspects of the relief valves as they pertain to the emergency core cooling system are discussed in Section 3.5.1 of this report.

We conclude that these valves will prevent overpressurization of the primary coolant system.

3.3 Primary Containment

3.3.1 Design and Construction

The primary containment design consists of a drywell, a connecting vent system, and a pressure suppression chamber (torus). The reactor vessel, the reactor coolant recirculating loops, and other branch connections of the reactor primary system are located in the drywell.

The drywell has a "light bulb" configuration consisting of a spherical section, 70 feet in diameter, and a cylindrical section approximately 23 feet in length and 33 feet in diameter. The design pressure is 62 psig. The pressure absorption chamber is in the form of a torus with a major diameter of 123 feet and an inner diameter of 27 feet. The design pressure is 35 psig. A vent system connects the drywell to the torus and terminates below the water level in the torus, so that in the event of a reactor system pipe failure in the drywell, the released steam passes directly to the torus pool water where it is condensed. This transfer of energy to the water pool rapidly reduces the pressure in the drywell, and thereby limits the amount of leakage from the primary containment.

Provisions are included for the removal of heat from the primary containment to maintain integrity of the containment system following any accident up to and including the design basis loss-of-coolant accident.

The basis for the design pressure and dynamic response of the primary containment is the loss of coolant following the sudden and complete severance of the largest line connected to the reactor vessel, while the reactor is operating at its steady state ultimate power level (1779 Mwt). The design criteria for containment are as follows:

(a) To withstand the peak transient pressure (coincident with an earthquake) which could occur due to the postulated break of any pipe inside the drywell.

(b) To channel the flows from postulated pipe breaks to the pressure absorption chamber.

(c) To withstand the force caused by the impingement of the fluid from a break in the largest local pipe or connection, without containment failure.

(d) To limit primary containment leakage rate during and following a postulated break in the reactor primary system to substantially less than that which would result in offsite doses approaching the reference values in 10 CFR 100.

(e) To include provisions for periodic leak rate tests.

(f) To be capable of being flooded in a design basis accident to a height which permits unloading of the core.

The design basis loss-of-coolant accident causes calculated peak pressures of about 34 psig in the drywell and 22 psig in the suppression chamber following severance of the recirculation line. Analytical methods based upon experimental information obtained at Humboldt Bay and Bodega Bay test facilities (Moss Landing) were used to calculate these pressures. Because these pressures are substantially below the design values, we conclude that the primary containment will have a significant margin above the peak pressures calculated for the recirculation line break.

Containment piping penetrations are protected against overpressurization due to a rupture of a hot process line by a guard pipe between the process line and the penetration sleeve. Expansion bellows are also provided on hot process line penetrations. On all double-ended penetrations the inner penetration seal is designed to fail at a lower pressure than the outer seal on the pipe sleeve. The penetration sleeves are all capable of withstanding pressures greater than the containment design pressure. The penetrations are also designed to accommodate the effects of pipe rupture with consideration of the potential for thrust and impingement loads. We conclude that these design requirements will provide the requisite degree of functional integrity.

The design of the primary containment structure is based on the regulations of the American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III for Class "B" vessels, American Society for Testing and Materials Standards, and the American Institute of Steel Construction.

Considerations of accident pressure, jet loads, thermal load, dead load, external load, and seismic loads have been accounted for in the containment design. The various loadings have been considered together in logical and conservative combinations. Under these critical load combinations the stresses in main load-carrying elements will be within the applicable code requirements.

The materials of construction have been selected in accordance with, and have been given a degree of attention in construction appropriate to, the critical nature of the structure. As part of the quality assurance program, the certified mill test reports were reviewed to assure their compliance with the material specifications. Shop and field fabrication techniques were closely controlled in order to ensure that a structure of the requisite quality had been achieved. Radiographic and magniflux techniques were used as required by the applicable sections of the ASME Code. We conclude that this structure has been designed and built to give satisfactory service over the design life of the facility.

A system for detecting leakage from the primary coolant system has also been incorporated into the primary containment. The system consists of an open drywell floor drain sump and closed equipment drain tank with associated pumps and piping. Leakage into these tanks is measured by monitoring the quantity of water which is automatically pumped from these tanks. As stated in the ACRS letter, during the first year of operation the applicant will study supplemental and potentially more sensitive methods for improving primary system leak detection capability. If the results of the study indicate that significant improvements in leakage monitoring capability can be achieved, such improvements will be implemented. The results of the applicant's program will be reviewed by the staff.

3.3.2 Testing and Surveillance

An overpressure test required by the ASME Code at 115% of the design pressures, 71.3 psig for the drywell and 46.3 psig for the suppression chamber, has verified that the primary containment meets its structural performance requirements. Integrated leak rate tests will be performed prior to initial plant operation at test pressures of 22 and 35 psig. To verify the plant's continued leaktight integrity, integrated leakage testing will be performed periodically at 22 psig. After the initial preoperational leakage test, additional tests will be performed on a schedule of 1, 2, 4 and every 4 years thereafter, provided the containment leakage remains within the allowable limit established in the Technical Specifications (i.e., a leak rate of 1.6% of the volume per day at a pressure of 35 psig). In addition, tests will be performed to assure that leakage from specified individual valves or penetrations do not exceed 5% of the allowable containment leakage limit. We conclude that the testing program is adequate to provide assurance of containment integrity throughout the service lifetime of the facility.

3.3.3 Containment Spray System

The containment heat removal spray system consists of two independent spray-cooling loops. Each loop will pump water from the containment absorption pool through heat exchangers cooled by the raw water system (service water system) into a pair of spray header systems located in the containment drywell atmosphere. Water from the spray headers flows by gravity back to the absorption pool.

Each of the containment spray loops has redundancy in active components (i.e., double pumps and valves) which provides protection against loss of any single active component. Since all automatic valves in the system will be kept open (except for testing) during plant operation, actuation of the containment spray depends only on operation of pumps. Single failures of passive components of the piping system could also be tolerated without reducing the capability of the system. On the basis of our review, we conclude that the containment spray system is acceptable.

3.3.4 Containment Inerting System

The containment atmosphere control system is designed to maintain an inert atmosphere within the primary containment to preclude possible combustion of hydrogen that may be evolved by a metal-water reaction as a consequence of a highly unlikely loss-of-coolant accident. The containment is purged with nitrogen gas before reactor operation and the oxygen concentration is maintained at less than 5%. Maintaining the oxygen concentration at this value assures that flammable mixtures of hydrogen and oxygen will not occur as a result of a metal-water reaction.

The system is located external to the drywell. Piping and component design, up to and including the first two isolation valves, will meet the requirements for Class I structures. The system also will be used to detect gross leakage paths in the primary containment boundary. This assures a continuous monitoring of containment integrity during plant operation. We conclude that the system as proposed by the applicant provides an adequate means for establishing and assuring an inert atmosphere within the containment and a means to continuously monitor containment integrity.

3.4 Secondary Containment

The secondary containment, or reactor building, encloses the primary containment structure (drywell and absorption chamber). It consists of reinforced concrete substructures to the elevation of the refueling floor, topped by a conventional steel building frame with insulated metal siding.

The building contains the reactor servicing facilities, new and spent fuel storage facilities, and reactor auxiliary systems including the isolation condenser system, demineralizers, standby liquid control system, control rod hydraulic system, and the standby gas treatment system.

The standby gas treatment system is designed to minimize the release of radioactive materials to the environment during a loss-of-coolant accident or whenever a high level of radioactivity exists in the reactor building. The system consists of two low capacity exhaust fans and two filtering trains of gas and particulate filters. Each train is capable of limiting the leak rate to 100% of the reactor building volume per day under neutral wind conditions. The fans are sized to maintain the reactor building pressure at a negative pressure of 0.25 inch of water.

A test program will be conducted to demonstrate the design capability of the secondary containment. Additional secondary containment capability tests will be conducted during various meteorological conditions and at each refueling outage. The charcoal filters of the standby gas treatment system will be tested to demonstrate a halogen removal efficiency of not less than 99%, using freon gas. The particulate filters will be tested using DOP to demonstrate a particulate removal efficiency of not less than 99% for particulate matter larger than 0.3 micron. We conclude that the design features and testing program for the reactor building and standby gas treatment system are adequate to demonstrate the capability to minimize the release of radioacticity to the environment.

3.5 Other Plant Systems

3.5.1 Emergency Core Cooling System (ECCS)

3.5.1.1 General

The principal subsystems that make up the ECCS are the auto-relief system and the two core spray systems.

In the event of a small break in the primary system and without offsite power sources, the auto-relief system would depressurize the reactor pressure vessel to permit operation of the low pressure core spray system before excessive fuel cladding heating occurs. The auto-relief system consists of six electromagnetic pressure relief valves located on the main steam lines inside the drywell of the primary containment vessel. Three valves are programmed to operate on initiation of the auto-relief system by high drywell pressure and low reactor water level. If a valve failed to operate, its mate would be automatically actuated; thus 100% redundancy is available.

Should a loss-of-coolant accident occur and offsite power remain available, the normal feedwater system could continue to deliver subcooled water to the reactor vessel. Addition of subcooled feedwater from the condenser hotwell causes the primary system to depressurize and thus allows the operation of the low pressure system before excessive cladding heating occurs.

The core spray subsystem is provided to circulate water from the suppression chamber to the reactor vessel. This water is distributed directly onto the fuel assemblies comprising the reactor core. The distribution is accomplished via the core spray spargers mounted inside the reactor vessel above the reactor core.

The core spray subsystem consists of two independent loops, each loop has redundancy of active components (i.e., double pumps and valves). Either loop is adequate to provide effective core cooling for the complete range of pipe break sizes considered as design basis loss-of-coolant accidents.

3.5.1.2 ECCS Functional Performance

The ECCS is provided to mitigate the consequences of loss-of-coolant accidents resulting from any size rupture of the primary system piping or equipment. The break spectrum considered included breaks equivalent to those resulting from pump and valve seals leakage as well as from double-ended pipe failures. The largest break considered during our evaluation was the double-ended rupture of a 26-inch recirculation line.

The applicant stated that the ECCS design criterion was that no fuel rod cladding melting would result for any postulated primary system rupture up to and including the double-ended rupture of a recirculation pipe. We did not accept this as the sole criterion because in our view the peak fuel rod cladding temperature should be limited to a temperature such that reasonable assurance is provided that the ECCS would terminate the temperature transient and assure an intact core geometry for effective long-term cooling. Based on our review of the available data in this regard, we concluded that peak fuel rod cladding temperatures should not exceed about 2000°F. Furthermore, the functional aspects of the core spray cooling are sufficiently well determined by tests and analysis to give reasonable assurance of its efficacy when clad temperatures are held to less than 2000°F. The results of the applicant's analysis of either onsite or offsite powered ECC subsystems indicate that the maximum predicted fuel clad temperatures do not exceed 2000°F for the entire spectrum of break sizes and locations that could occur in the design basis accidents. Consequently, we conclude that there is reasonable assurance that the core spray system would be effective in the unlikely event of a loss-of-coolant accident.

3.5.1.3 Mechanical Design of the ECCS

Core spray system piping external to the reactor vessel is designed to the stress limits set forth in the ASA B31.1 1955 Piping Code for maximum operating loads in combination with the design earthquake. Analyses of the piping system to determine the location of seismic snubbers and restraints have been reviewed by our seismic consultant, Dr. Newmark. He concluded, and we agree, that the design of the piping system is adequate to withstand the seismic conditions applicable to this facility.

The core spray spargers are located inside the reactor pressure vessel. Each sparger consists of two segments which form a ring header. Each segment is attached to the internal shroud at the inlet piping connection and is supported along the inner periphery of the shroud by saddle brackets. The applicant has indicated that the stresses are within Section III of the ASME Code allowables for all loading conditions including accident loads in combination with seismic loads even though they were not originally designed for combined accident and seismic loads. We conclude that this design produces an acceptable margin of safety for this facility.

The supply of water for the core spray is taken directly from the torus via connecting pipes. Excessive leakage of this water could flood the lower part of the reactor building. In the design as originally proposed by the applicant, this could lead to a loss of function of the core spray and containment spray systems. To prevent such a loss, the plant design has been modified by (a) connecting the raw water system to the core and containment spray systems to provide an alternate source of water, (b) sealing all penetrations into the pump compartments, and (c) providing water-tight doors at the entrances (from the torus or center room) to the pump compartments. We conclude that these changes provide assurance that sufficient water for emergency core cooling would be available in the highly unlikely event of excessive leakage from the piping systems.

3.5.2 Fuel Handling and Storage

Fuel handling operations are carried out using facilities provided for unloading and storing of new fuel in the reactor building, transferring and unloading of new assemblies into the reactor core, underwater removal of spent fuel assemblies from the reactor core, transfer of spent fuel assemblies from within the reactor containment to storage in the spent fuel pool, and offsite shipment of spent fuel assemblies for reprocessing in a specially designed cask.

During refueling, transport to the spent fuel storage pool, and during storage, spent fuel will be continuously submerged in water. The spent fuel storage racks in the pit are arranged to ensure a subcritical array. During refueling and storage, personnel will be protected by water and/or concrete shielding. Systems are provided to monitor spent fuel pool water temperatures and activity. Interlocks, as described in Section 5.3, have been provided to prevent manipulations which could result in fuel damage during the refueling operation.

On the basis of our review, we have concluded that the provisions for fuel handling and storage are acceptable.

3.5.3 Control Room

The control room is located in the southeast corner of the turbine building and contains all necessary controls and instrumentation for operation of the reactor, turbine-generator and auxiliary systems. The control room is designed to be occupied during design basis accident conditions as well as during normal operation. Although specific provisions were not made in the design, the equipment necessary to conduct safe shutdown can be operated remotely from outside the control room.

The control room has adequate instrumentation and controls for controlling the reactor plant in a safe manner. While all reactor protection and engineered safety features are automatic, facilities for manual operation of the safety features are also provided in the control room.

We have evaluated the design of the reactor control room with respect to the adequacy of the shielding for continuous occupancy in the event of the design basis accident, and with respect to the potential doses during ingress and egress subsequent to an accident. Our calculations show that adequate shielding has been provided to limit the whole body doses to an operator to less than 5 rem. We conclude that adequate control room shielding has been provided.

3.5.4 Radwaste Systems

The purpose of the radwaste system is to treat and dispose of all types of solid, liquid, and gaseous wastes accumulated during operation of the facility.

The solid radwaste system serves to collect, process, and package items such as filter sludge, spent resins, and equipment originating in the primary system for offsite disposal. The material is dewatered in a centrifuge, compressed into 55-gallon drums, or mixed with concrete in preparation for shipment, depending on the quantity and activity level.

The gaseous radioactive waste control system is designed to process noncondensible gaseous products from the main condenser to limit fission product release to the environment. A 30-minute holdup capability is provided to allow radioactive decay of short lived products prior to stack release. The stack gas is continually monitored.

The liquid radioactive waste system collects, treats, and disposes of all liquid wastes generated within the facility. All liquid wastes are collected, sampled and discharged on a batch basis, so that inadvertent discharge of high activity waste is unlikely.

We conclude that these systems are adequate to assure that the 10 CFR Part 20 limits will not be exceeded.

4.0 ELECTRICAL POWER

The onsite AC electrical power system will utilize two redundant 3125 KVA (2560 kw) diesel generator units arranged in a split-bus configuration. Each generator is rated at 3125 KVA continuous, and 3360 KVA for 2000 hours per year. Maximum emergency loads are 2740 KVA. Thus, a 12% margin is available, even if one diesel generator were to fail. There are two internal busses, each of which is directly energized by one of the diesel generators. The separation extends through the downstream 480 volt sections. The generators will not be connected in parallel. No crossties are provided between the two essential busses. Two separate and independent sources of DC electrical power have also been provided.

Offsite electrical power is available from two 115 kV transmission lines and is fed into the emergency busses by two 115/4.16 kV startup transformers. Each startup transformer energizes one of the essential busses.

Based on our evaluation, we have concluded that the electrical power system for NMP, including the DC portions, is adequate.

5.0 INSTRUMENTATION AND CONTROLS

5.1 Protection System

Much of the instrumentation and control circuitry for the Nine Mile Point Nuclear Station is functionally identical to that provided for the Oyster Creek station.

The Nuclear Instrument system consists of Source Range Monitors (SRM), Intermediate Range Monitors (IRM) and Local Power Range Monitors (LPRM). The power range monitors provide individual continuous measurements of local power level throughout the core as well as average power level in the core quadrants. The SRM system uses pulse counting techniques and derives period information which is displayed. There is no period scram. The IRM system uses the "Campbell" measurement technique and consists of eight channels of instrumentation feeding eight variable range amplifiers. Reactor scram is initiated when (at least) one IRM channel in each of the two protection channels is driven to the upscale trip point. The LPRM system consists of 120 independent channels which utilize miniature fission chambers as sensors. The outputs of 64 LPRM channels are combined (averaged) as eight distinct Average Power Range Monitor (APRM) channels, each APRM channel being fed from eight LPRM channels located in a particular quadrant. There are two APRM channels in each quadrant. Each is connected to a different channel of the dual channel protection system. Upscale tripping provides scram (1/2 x 2 logic) and rod-block (1/8 logic).

Power/flow protection (rod-block) is provided by flow signals which continuously adjust the upscale trip points of the APRM channels.

A traversing In-Core Probe (TIP) is inserted into the core to obtain flux distributions, and to calibrate the LPRM system.

Radiation monitors on the main steam lines have sufficient sensitivity to detect gross failure of fuel elements. As indicated in the ACRS letter, the applicant will study methods to improve the use of these instruments and instruments in the off-gas system to provide the operator with an early indication of fuel failures as operating experience is gained with the facility. We plan further review of this matter as part of the full term operating license review.

Five sets of instrument channels monitor the following process system parameters and provide scram capability:

High Reactor Pressure

High Primary Containment (drywell) Pressure

Low Reactor Water Level

Low Condenser Vacuum

High Radiation, Main Steam Lines

Each is monitored by four independent channels connected in 1/2 x 2 logic. Scram is also initiated upon loss of voltage to the protection system, upon main steam line isolation (both lines), and manually. Each channel consists of two independent subchannels made of relay contacts controlled by the various channels of the protection system instrumentation. A subchannel, in turn, controls one relay. The tripping of a subchannel is equivalent to tripping the respective channel. Tripping both channels of the dual system scrams the reactor.

We have reviewed the design of the dual channel protection system, including the containment isolation system, and have concluded that it conforms to all applicable criteria. We have also independently verified the applicant's analyses that the Intermediate Range Monitors obviate the need for period scram.

Based on the foregoing, we conclude that the design of the NMP protection system is acceptable.

5.2 Rod Block

The rod block function serves to protect the core from local transients induced by improper control rod withdrawal. At a given flow rate and reactor power, an increase in reactor power to about 107% will generate a signal from one of the eight APRM channels to automatically actuate rod block circuitry. Our review indicates that the rod block system is redundant and testable, and is acceptable.

5.3 Refueling Interlock

The Refueling Interlock system is essentially an arrangement of electrical interlocks between the fuel hoist mechanisms and the control rod drives such that a loaded hoist cannot be over the core when more than one rod is in a withdrawn condition. Our analysis shows that, with the mode switch in the "Refuel" position, the system meets the single failure criterion, and is fail-safe upon voltage loss. If, during refueling operations, the mode switch is placed in the "Run" or "Shutdown" position, a scram will occur. If the switch is in the "Startup" position, as occasionally required during refueling, a portion of the total interlock arrangement is bypassed in order to allow the withdrawal of more than one rod. We find this design feature to be satisfactory in view of the brief duration of such operation and the additional administrative controls which would be imposed during such operation.

Based on the foregoing, we conclude that the design of the Refueling Interlock system is acceptable.

5.4 Containment Spray System

There are two independent systems, each with its own spray header. Within each system there are two spray pumps and two raw water pumps. Each system is energized from one of the two emergency busses, and can provide full safety feature action.

The containment spray system is actuated by instrument channels which monitor reactor water level and drywell pressure. Each of these two parameters is monitored by four independent instrument channels. The eight instrument channels provide signals to a dual channel logic system (1/2 x 2). Upon receipt of a reactor water level (LO-LO) signal and a drywell pressure signal, a subchannel of the logic channel is tripped. Tripping both channels of the dual system will activate the pump-start circuitry of both systems. Only the pump-start timers of the containment spray pumps are energized. The containment spray raw water pumps are started manually as they are not required until approximately 30 minutes into the accident.

On the basis of our review, we conclude that the instrumentation and controls for the containment spray system are acceptable.

5.5 Core Spray System

There are two separate and independent core spray systems, each system consisting of two parallel supply loops. Each supply loop contains a core spray pump, a core spray topping pump and the necessary system valves, piping and instrumentation. Two sets of pumps, each consisting of a core spray and a core spray booster pump are connected to each of the 4.16 kV busses. One set of pumps associated with a given essential bus provides flow for one core spray system while the second set of pumps on that same bus provides flow for the second core spray system. The two supply loops associated with the second essential bus are connected in the same way. Each set of pumps can provide 100% of required system design capacity, thus 400% capacity is provided.

The core spray system is actuated by instrumentation channels which monitor reactor water level, drywell pressure and reactor pressure. Each of these three parameters is monitored by four independent instrument channels. The twelve instrument channels provide signals to a dual channel logic system as previously discussed. The inputs to each sub-channel consist of reactor water level, drywell pressure and reactor pressure.

Upon receipt of a reactor water level (LO-LO) signal or a drywell pressure (HI-HI) signal a logic subchannel is tripped. Tripping both channels of the dual system will actuate the pump-start circuitry. The discharge valves in each core spray loop are opened by a signal derived from primary system pressure. When a reactor water level (LO-LO) signal or a drywell pressure (HI-HI) signal is received in coincidence with a reactor pressure (LO) signal, the pump-start circuitry and discharge valves are actuated simultaneously.

Under conditions of automatic initiation, all pump-start timers are energized simultaneously. Four timing circuits, two for each of the 4.16 kV essential busses, sequence the starting of the core spray pumps. The first timer for each bus times out in 5 seconds and completes the first core spray pump-start circuit for the associated bus. The second timer for each bus times out in 13 seconds and completes the second core spray pump-start circuit for the associated bus. These same timers provide additional signals to complete the starting circuits of the core spray topping pumps after the core spray pumps are running.

Based on our analysis we have concluded that the instrumentation and controls for the core spray system are acceptable.

5.6 Auto-Relief System

The automatic depressurization system consists of six solenoid-actuated relief valves, three of which are required for system operation.

The system is actuated by instrumentation which monitors reactor water level and drywell pressure. Each of these two parameters is monitored by four independent instrument channels. The eight instrument channels provide signals to two redundant matrices, either of which can operate the valves.

Within a matrix the logic is as follows: 1 of 2 reactor water level (LO-LO-LO) and 1 of 2 drywell pressure (HI-HI) in coincidence with a second logic section of 1 of 2 reactor water level (LO-LO-LO) and 1 of 2 drywell pressure (HI-HI).

Upon receipt of coincident reactor water level and drywell pressure signals the matrix actuates a timer circuit. This timing circuit times out in 120 seconds thus providing time for the operator to intervene. The operator can interrupt the operation and reset the timer for an additional 120-second period or interrupt the operation for 120-second periods after the blowdown is in progress. Completion of the timing cycle, a seal-in function, completes the actuation circuitry of three, preferred electromagnetic relief valves. The timer continues to time out and at 125 seconds provides a signal to the second set of three redundant valves. If at this time one of the three preferred valves has not left its fully closed position, a backup valve will operate.

A second matrix, redundant to the first, can also actuate the valves. This matrix receives signals from the same eight instrument channels. A second timer, identical to the first timer, provides redundant signals to the actuation circuitry of the preferred valves. A signal is also provided to the backup valves at 125 seconds.

Based on our review, we have concluded that the design of the auto-relief system is acceptable.

6.0 ANALYSES OF DESIGN BASIS ACCIDENTS

Four major postulated accident situations were examined to explore possible means by which fission products might escape from the facility. The rod-drop accident could release fission products from the fuel through the the steam lines to the condenser where they would be released to the environs. The refueling accident would release fission products from the fuel through the refueling pool water to the reactor building. The steam-line-break accident would release fission products in the primary coolant directly to the environs. A loss-of-coolant accident would release fission products from the fuel to the primary containment. This latter accident was assumed for this facility to determine compliance with the guidelines established in 10 CFR Part 100.

The results of our analyses for these accidents are summarized in the following sections and the doses which we have calculated using conservative assumptions are summarized in the following table. We have assumed only 90 percent efficiency for halogen removal as compared with the 99 percent which the applicant believes will be achieved. Credit for release of activity from the 350-foot stack was given except for the steamline-break and control-rod-drop accidents.

TABLE 6.0

DOSE SUMMARY

Accident	Two Hour @ 0.78 Mile (rem) Exclusion Area Radius		Course of Accident @ 4 Miles (rem) Low Population Zone Radius	
	Thyroid	Whole Body	Thyroid	Whole Body
Loss of Coolant	150	6	70	1
Refueling	65	2	30	1
Control Rod Drop	20	<1	10	<1
Steamline Break	30	<1	5	<1

The meteorology used in our calculation of the consequences of the refueling and loss-of-coolant accidents was as follows: fumigation conditions were assumed for the two-hour exposure at the site boundary. In calculating the doses at the low population distance, we assumed the dilution factor that results from the use of an envelope over the curves for the various Pasquill types of atmospheric diffusion parameters for a 350-foot release height, which maximizes dose with distance for the first 8 hours. For 8 to 24 hours this condition was assumed to continue, but the plume was spread uniformly in a 22-1/2 degree sector. For the next three days, the wind was assumed to continue blowing into the same sector, but diffusion conditions varied so that the location of the maximum concentration changed and the wind speed was allowed to increase. After four days, similar diffusion conditions continued, but the wind was assumed to blow into the sector only 1/3 of the time.

For the steam-line-break and control-rod-drop accidents, ground release was assumed with Pasquill Type F conditions and a wind speed of 1 m/sec for calculations of the two-hour doses at the site boundary. At the low population distance, these conditions were assumed to continue for 8 hours. During the next 8 to 24 hours a wind speed of 1 m/sec and spreading of the plume into a 22-1/2 degree sector were assumed. For these accidents the 24-hour time interval is the full course of the accident.

As seen from the data in the above table, the doses resulting from these postulated accidents are well within 10 CFR Part 100 guideline values.

6.1 Loss of Coolant Inside the Drywell

In calculating the consequences of the loss-of-coolant accident associated with 100% fuel perforation, we have assumed fission product release fractions released from the core as suggested in Technical Information Document 14844, "Calculations of Distance Factors for Power and Test Reactor Sites"; i.e., 100% of the noble gases, 50% of the halogens, and 1% of the solids.

A primary containment leak rate of 2.0 percent of the containment volume per day was assumed to remain constant for the duration of the accident.

We have assumed a 90% halogen removal efficiency of the charcoal adsorbers of the standby gas treatment system in the secondary containment building. In our analysis, we took the conservative approach of assuming that leakage from the drywell goes directly to the standby gas treatment system without mixing and then out the stack at 350 feet above ground level.

In addition to the radiological consequences of an assumed loss-of-coolant accident, the potential consequences of radiolytic decomposition of water have been considered. The significance of the matter is not completely understood or known at this time. However, such decomposition would result in the production of gaseous hydrogen and oxygen in the containment atmosphere. If sufficient hydrogen and oxygen are produced by such a reaction, it is possible that a flammable mixture could be attained that if ignited would introduce an additional source of energy into the containment system. Preliminary studies by the applicant suggest that the extent of the decomposition reaction may be limited by back-reaction rates. As noted in the ACRS letter, we will evaluate further information as it becomes available and will take action as deemed necessary. This matter is undergoing thorough review by industry, Oak Ridge National Laboratory, Battelle Memorial Institute, and the Commission's Division of Reactor Licensing. We conclude that the outcome of these efforts will be available to reevaluate the matter within the 18-month term of the provisional operating license.

6.2 Control-Rod-Drop

In the control-rod-drop accident it is assumed that a bottom-entry rod has been fully inserted and has stuck in this position unknown to the reactor operator. It is then assumed that the drive becomes uncoupled and withdrawn from the rod. Subsequently, it is assumed that the rod falls out of the core inserting an amount of reactivity corresponding to the worth of the rod.

Hot standby is the worst operating condition at which the accident could happen both because a higher energy release is calculated for this condition and because a path for the unfiltered release of fission products could exist through the mechanical vacuum pump on the condenser. A rod reactivity worth of 2.5% $\Delta k/k$, the highest worth rod permitted by operating procedures, was assumed in the analysis. This reactivity addition would produce an excursion with a minimum reactor period of 8.5 milliseconds and a total energy generation of 4000 Mw-sec, resulting in a peak fuel energy density of about 200 cal/gm (average across the peak fuel pellet). Perforation of about 330 fuel rods is predicted.

We have evaluated the consequences of the control-rod-drop accident assuming that 3300 fuel rods fail, releasing 100 percent of the noble gases and 50 percent of the halogens from the affected rods to the primary system. Of the halogens released from the affected rods, 90 percent are assumed to be retained in the primary system and one-half of the remaining halogens are assumed to be removed by plate-out. All of the noble gases and 2.5% of the halogens would be released from the primary system through the condenser vacuum pump system to the atmosphere via the stack.

An automatic isolation valve has been installed on the discharge side of the condenser vacuum pump which would be closed by a high radiation signal from the steamline monitor to confine fission products released from the fuel to the primary system. The pump would also be tripped by those signals, thus providing a second barrier to the release of fission products. These features were considered in the calculations, and the resulting doses are well within the 10 CFR 100 guidelines.

6.3 Refueling Accident

The refueling accident is assumed to occur 24 hours after shutdown. During fuel handling operation, a fuel bundle is assumed to fall onto the core with sufficient force to physically damage (perforate) 445 fuel rods with consequent release of 20% of the noble gases and 10% of the halogens from the damaged rods into the reactor building. Ninety percent of the halogens released from the perforated fuel rods are assumed to remain in the refueling water. The remaining airborne fission products (20% of the noble gases and 1% of the halogens contained in the fuel) within the building are assumed to be discharged to the atmosphere through the standby gas treatment system (with an iodine filter removal efficiency of 90%) and through the stack over a 2-hour period.

6.4 Steamline Break Outside Containment

The break of a main steamline outside of both the drywell and the reactor building represents a potential escape route for reactor coolant from the vessel to the atmosphere without passage through the primary containment or the reactor building.

The steamline break would be sensed by either increased pressure drop across the steam line venturi or increased temperature in the pipe tunnel. The steamline isolation valves would start to close within 0.5 second after the steamline break. We have assumed an isolation valve closure time of 10 seconds. With this closure time no fuel clad-perforations would be expected to occur to release additional fission products to the environs.

The primary coolant activity used in the calculations corresponds to the total iodine activity limit of 25 μCi/cc, given in the Technical Specifications.

6.5 Conclusion

On the basis of our evaluation, the radiological doses that could result from any of the design basis accidents are well within the guideline values given in 10 CFR Part 100.

7.0 EMERGENCY PLANNING

The applicant has described a comprehensive plan for coping with the unlikely event 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 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 of 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.

Niagara Mohawk has contracted with a physician who has training in the field of radiation medicine to provide medical consultant services and continuing professional training for the staffs of two local hospitals which have agreed to provide medical support to the Nine Mile Point facility, 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.

8.0 CONDUCT OF OPERATIONS AND TECHNICAL QUALIFICATIONS

Responsibility for safe operation of the plant is vested in the Plant Superintendent. He reports to the General Superintendent of the Central Division. This is one of three regional divisions within the Niagara Mohawk grid. The Division Superintendents report to the Vice President for Operations.

Within the onsite operating organization, responsibility for day-to-day operation of the facility rests with the Operations Supervisor, reporting to the Plant Superintendent. The Operations Supervisor will be a licensed senior reactor operator, as will each Station Shift Supervisor. The normal shift complement shall consist of at least four persons including one licensed senior reactor operator and one licensed reactor operator. After completion of the startup test program and demonstration of plant electrical output, the normal shift complement will remain at four members but shall include an additional licensed reactor operator.

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 have concluded that the professional staff is technically competent to operate the facility.

Engineering support to the Nine Mile Point station will be provided by the Niagara Mohawk engineering group, under the Vice President for Engineering, as well as by General Electric and specialist consultant firms. The Niagara Mohawk engineering staff is familiar with the plant and is capable of handling the preparation and review of design changes and plant modifications originating at the Nine Mile Point site.

General Electric will participate in the startup and initial operation of the plant and will continue to make available technical support to the Niagara Mohawk staff throughout the operating lifetime of the facility. On these bases, we conclude that adequate engineering capability will be available 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 Station 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 Safety Review and Audit Board, at least one member of which will be from outside the Niagara Mohawk organization for the first few years of operation. 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.

9.0 TECHNICAL SPECIFICATIONS

The applicant's proposed Technical Specifications to the license for NMP are presented in Amendment No. 3. 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. Modifications to the proposed Technical Specifications submitted by the applicant were made to more clearly describe the allowed conditions for plant operation. The finally approved Technical Specifications are appended to the proposed provisional operating license. Based upon our review, we conclude that normal plant operation within the limits of the Technical Specifications will not result in potential offsite exposures 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.

10.0 REPORT OF ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS)

As noted previously, the ACRS has reviewed the application for a provisional operating license for the Nine Mile Point Nuclear Station. The Committee completed its review of the facility during its 108th meeting held April 10-12, 1969. A copy of the ACRS letter, dated April 17, 1969, is attached.

The ACRS, in its letter, made several recommendations to be followed during operation of the facility. These matters have been considered in our evaluation. They include: inspection schedule for corrosion surveillance (Section 3.2.1), periodic inspection of the reactor high pressure coolant system including reactor pressure vessel welds and main steam lines (Section 3.2), primary system leak detection methods (Section 3.3.1), possible effects of radiolysis of water in the unlikely event of a loss-of-coolant accident (Section 6.1), methods to monitor for vibration during plant operation (Section 3.2.1), safety review and audit functions (Section 8.0), and improvement of the capability of instrumentation of the type used on the main steam lines to detect early signs of gross failure of fuel elements (Section 5.1).

The applicant has agreed to see that the recommendations of the ACRS are carried out. We will follow the implementation of 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 to given to its comments, the Nine Mile Point Nuclear Station can be operated at power levels up to 1538 Mwt without undue hazard to the health and safety of the public.

11.0 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 requirements of Part 50. 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 from 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.

12.0 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 Nine Mile Point Nuclear Station can be operated as proposed without endangering the health and safety of the public.

Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Date: May 26, 1969

APPENDIX

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

APR 17 1969

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

Subject: REPORT ON NINE MILE POINT NUCLEAR STATION

Dear Dr. Seaborg:

During its 108th meeting, April 10-12, 1969, the Advisory Committee on Reactor Safeguards completed its review of the application by the Niagara Mohawk Power Corporation for a license to operate the Nine Mile Point Nuclear Station at power levels up to 1538 MW(t). During this review, the project was considered at Subcommittee meetings held on February 27, 1969 (at the site), and on April 8, 1969. In the course of these meetings, the Committee had the benefit of discussions with representatives and consultants of Niagara Mohawk Power Corporation, General Electric Company, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed. The Committee previously discussed this project in a construction permit report dated October 15, 1964.

The Nine Mile Point Nuclear Station employs a boiling water reactor. Power level, core design, and other principal features of the nuclear steam supply system are generally similar to those for the Oyster Creek Nuclear Power Plant Unit No. 1, previously discussed in the Committee's report to you dated December 12, 1968.

As in Oyster Creek Unit No. 1, type 304 stainless steel utilized at a number of places in the reactor vessel was furnace-sensitized during fabrication. Careful examination of these parts for evidence of corrosion has been made by the applicant, and none has been found. Although the likelihood of occurrence of significant corrosion (intergranular attack) during the service life of the plant appears small, the applicant plans to install appropriate corrosion test specimens within the vessel for future examination. The Committee believes that the applicant should resolve with the AEC Regulatory Staff, prior to the start of operation, a satisfactory schedule and inspection procedure for at least the initial portion of this corrosion surveillance program.

The Committee wishes to emphasize the importance of periodic inspection of the high pressure coolant system in this and other reactors. The in-service inspection requirements for this reactor as described, and to be stated in the Technical Specifications, appear adequate for initial operation. The Committee agrees with the applicant's intention to review his inspection program after about five years of operation. Because of the difficulties inherent in direct inspection of the bulk of the welds in the reactor pressure vessel after the reactor is in service, it is strongly recommended that alternative means for assuring continued pressure vessel integrity be studied and implemented to the degree practical. In addition, the applicant should develop more specific plans for in-service inspection of the main steam lines beyond the second isolation valve.

The applicant plans to study supplemental and potentially more sensitive methods of primary system leak detection and to implement methods which provide significant improvements in measurement of leak rate, in the time needed to measure leak rate, or in distinguishing the nature of the leak. The applicant should report to the Regulatory Staff his progress in this area within a year after start of power operation.

Studies are continuing on the possible effects of radiolysis of water in the unlikely event of a loss-of-coolant accident. These studies should be evaluated by the Regulatory Staff and appropriate measures taken as deemed necessary. Such measures should make allowance for effects of hydrogen generated by metal-water reactions if the effectiveness of the emergency core cooling system should be less than that predicted by the applicant.

The applicant has stated that he plans to study possible means of instrumenting and monitoring for vibration or for the presence of loose parts in the reactor pressure vessel as well as in other portions of the primary system and, by the time of the first refueling outage, to implement such means as are found practical and appropriate.

The safety review and audit function proposed by the applicant appears to be satisfactory. However, the Committee recommends that membership of the Safety Review and Audit Board include one or more experts from outside the applicant's organization, at least for the first few years of operation, to aid in effecting sufficiently independent review.

The applicant indicates that instrumentation which senses radioactivity from the steam system can be used to provide early signs of gross failure of fuel elements. As operating experience is gained, he intends to improve the utilization of this type of instrumentation for this purpose. The Committee strongly endorses this effort.

The Advisory Committee on Reactor Safeguards believes that, if due regard is given to the items mentioned above, the Nine Mile Point Nuclear Station can be operated at power levels up to 1538 MW(t) without undue risk to the health and safety of the public.

Sincerely yours,

Joseph M. Hendrie

Joseph M. Hendrie
Acting Chairman

References - Nine Mile Point Nuclear Station

1. Volumes I - IV, Final Safety Analysis Report.
2. First - Seventh Supplement to Final Safety Analysis Report.
3. Amendments 2 - 13, to Application for Licenses.
4. Final Safety Analysis Report - Nine Mile Point Nuclear Station - Technical Specifications (Revised), Draft - dated April 1969.