UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

JUL 10 1964

Niagara Mohawk Power Corporation
300 Erie Boulevard West
Syracuse 2, New York

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

Gentlemen:

As you know, your application for a Provisional Construction Permit for the proposed Nine Mile Point Nuclear Station is being evaluated by the Division of Reactor Licensing. As a part of this evaluation, you have met with members of the staff on June 3 and 4, 1964, and with an ACRS Subcommittee on June 10, 1964. Based upon our review of your application and the discussions during these meetings, the information requested in the attached list of questions will be necessary in order to complete our evaluation of your application. Further information may be found necessary as the review of the staff continues.

We understand that changes have been made in the design of the facility from that described in the Preliminary Hazards Summary Report. Your application should be modified accordingly.

The above information should be submitted as an amendment to your application in three copies signed under oath or affirmation and sixty additional copies.

Sincerely yours,

/S/

R. L. Doan, Director
Division of Reactor Licensing

Enclosure:
As stated above

cc: Mr. Arvin E. Upton
LeBoeuf, Lamb & Leiby

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SECTION I

INTRODUCTION

Several areas in the design of the Niagara Mohawk reactor may be considered to fall under the category of Research and Development as noted in paragraph 50.35(a)(3) of the Commission's Rules and Regulations. Provide information which assures that each of the following programs is reasonably designed to resolve the safety questions involved prior to completion of the facility:

a. Fuel development program.
b. In-core or in-vessel start-up range neutron detectors development.
c. Control system safeguards development.
d. Control by use of variable speed recirculation pumps.
e. Operation with 35% average void fraction.
f. Critical experiment program.

SECTION III

STATION DESCRIPTION

1. With reference to seismology and seismic design:

   a. Provide representative lists of Class I and Class II structures and components as defined in Section III A.2.0 of the PHSR. Include in these lists primary and auxiliary equipment which must continue to function during and after the earthquake to protect the facility.

   b. State the maximum allowable stresses in terms of a percentage of working or yield stress for Class I and Class II structures and components when under combined seismic and functional load from a 0.01g and a 0.11g earthquake.

   c. Provide a family of curves on a log-log plot of acceleration response spectrum as a function of period from 0.01 second to 1 second for various values of damping and state how these curves will be used in the seismic design of facility components and structures.

   d. Provide a table of damping factors for typical structures and components.

2. State the basis for the use of ASME Boiler and Pressure Vessel Code, Section VIII, in the design and fabrication of the containment structure instead of the current applicable Section III.

3. With reference to leak rate testing of the containment:

   a. State the design criteria or concepts which will be considered in providing individual leak testing capabilities for all potential sources of containment leakage.

   b. Define whether the leakage rate limit, specified as 0.5 per cent per day at design pressure, is a composite rate for the combined dry well and pressure suppression chamber at a specific pressure, or if the leakage rate limit is to be interpreted in some other manner.

   c. State the criteria upon which the retest pressure will be established which will be used to verify on a periodic basis the integral leakage rate of the pressure suppression system at design pressure. Include factors to be used to extrapolate from test pressure to design pressure.

   d. Define the criteria which will be employed in providing testing capabilities of the leak-tightness of the containment isolation valves and of those systems considered as an extension of the containment boundary (such as the post-incident pressure suppression chamber cooling system).

   e. Describe your plans, if any, to monitor the reactor building and the pressure absorption system leakage rates continuously to detect the development of any leaks which may occur.

f. Describe provisions for testing the leak tightness of the dry well subsequent to refueling.

4. State the design provisions under consideration, (a) to permit inspection of the internal components of the pressure suppression chamber, and (b) to incorporate baffles in the pool, if required.

5. Provide a list of type and approximate number of penetrations between the drywell and the turbine building. State a fraction of the total allowable leakage rate which could occur via these paths.

6. Describe how the Moss Landing test results presented in Appendix I to the PHSR have been applied to the proposed Niagara-Mohawk design by supplying, if applicable, a discussion of assumptions and calculation techniques.

7. State design and performance criteria for actuation of containment isolation valves in the event of an accident in terms of backup or redundancy of actuating sensing devices and power supplies.

8. Provide the following information relative to the reactor building:

   a. The method by which the proposed accident leakage rate of 100% per day at 1/4 inch of water will be tested.

   b. The design and performance criteria and a description of the emergency ventilation system.

   c. The provisions for periodic testing of the components of the emergency ventilation system.

   d. The ability of the filters in the emergency ventilation system to dissipate the fission product heat which would be present.

   e. State the design pressure rating of the reactor building.

9. State the safety valve flow rates and the pressure set points. Include the amounts of accumulation.

10. With reference to the reactor core, provide the following:

   a. The mechanical design criteria of the fuel rods in terms of maximum allowable stress, clad and pellet surface temperatures, internal pressure, burn-up and fuel integrity.

   b. The design and performance criteria and a description of the fuel assemblies.

c. The manner in which principal core components, include fuel assemblies, are to be assembled and supported in the core.

d. Provide design and performance criteria and a description of the control rods and fixed curtains, including materials and method of construction, inspection and testing, and experience to date.

11. Provide a summary description of the control rod drive system including experience to date and prototype testing program.

12. Is the scram logic scheme based on the coincident tripping of both logic channels, with each channel being subdivided into two subchannels?

13. With reference to the recirculation system:

a. Provide the design and performance criteria and a description of the recirculation system and its associated control system.

b. Provide a description of the proposed method of control of the reactor by recirculation flow in order to achieve the objective of load following within specified limits.

c. Give the minimum DNB ratio which may result from the worst combination of parameters such as flow rate, power level, and control rod pattern. Give the calculated DNB ratio for the case where the power level would rise to the highest scram set point with the recirculation flow rate at its minimum value. Include in your reply to the above, the data, such as flow rate and peaking factors, which were used in calculating the DNB ratio.

14. Provide a list of all equipment and the power requirement of each item of equipment that will be supplied from (1) the engine driven generator and (2) the station battery. State the design criteria of each of these sources in terms of available capacity and the length of time the power will be required.

15. Provide the design and performance criteria, a description, and provisions for testing the following systems:

a. Drywell ventilation system.

b. Emergency cooling system.

c. Core spray system.

d. Post incident cooling system.

16. Provide a list of all primary and auxiliary equipment that must function through the course of (1) the maximum credible accident, (2) the loss of coolant accident, and (3) the fueling accident. For each piece of equipment noted, provide the following:

    a. Description of devices (pressure transducer, radiation monitors, etc.) that will cause actuation of the equipment.

    b. Description of the electrical power supplies for all equipment that must function.

    c. The estimated length of time that each piece of equipment must continue to function.

    d. State capacity criteria of the equipment to handle the design function.

    e. If redundancy is used in equipment and systems to achieve reliability, evaluate the degree of independence of operation of the equipment or system.

    f. Analyze the consequences of individual failure of the various required pieces of equipment.

    g. State reliability criteria of all components considering pressure, temperature, and moisture conditions under which it must operate.

17. Submit the design and performance criteria which will ensure that all components and instrumentation of systems located within the primary containment and essential to safety will withstand the imposed leak rate tests and accident pressures, temperature, and steam without damage or impairment of operability.

18. State the design criteria for personnel access to the control room during the course of reactor accidents. Define "continuous occupancy" as used on page II-77.

19. Indicate whether any pipes or tubes, such as pneumatic or hydraulic lines, lead from the dry well area or the suppression pool tank to the control room. If they do, discuss the potential for the transmission of radioactivity through these to the control room, if there were large quantities of radioactive material present in the dry well.

20. Explain how the "vessel to basin seal" is maintained during refueling and the manner in which the area above the vessel is connected to the rest of the dry well when the reactor system is pressurized.

SECTION IV

RADIOACTIVE WASTE DISPOSAL

1. With reference to monitoring of gaseous radioactive effluents at the plant stack:

   a. Explain why it is considered acceptable to only periodically analyze samples for particulates and halogens.

   b. In view of the relative MPC's of noble gases and halogens, explain in terms of monitoring instrument sensitivities whether an unexpected increase in the halogen content in the stack off-gas would be detected before resulting in off-site overexposures.

2. State the criteria for utilization of effluent monitoring equipment in terms of sources of radioactivity, frequency of monitoring, levels monitored, and alarm points.

3. Describe those design features that will assure that the spent fuel pit water cannot drain thus exposing fuel. What provisions will be available to alarm excessive leakage and add water should a leak occur.

4. State criteria governing the height of the facility stack.

5. State the design and performance criteria for the liquid and the gaseous waste disposal systems in terms of:

   a. Maximum allowable off-site personnel doses.

   b. Assumed sources of radioactive effluent material.

   c. Hold-up capacity.

   d. Routine emission rate.

   e. Instantaneous emission rate.

   f. Expected levels of noble gases and halogens in off-gas stream.

6. List those points at which airborne radioactive material will be discharged to the environment without the material passing through the reactor stack. Describe the precautions which will be taken to reduce the possibility of accidental release of radioactive isotopes from such areas and the manner in which significant releases will be monitored.

7. By assuming various periods of waste gas hold-up, (for example, one minute to one month) state the facility stack emission rates for the principal noble gas and halogen effluents for normal operating conditions.

SECTION V

REACTOR CHARACTERISTICS

1. With reference to hydraulic and nuclear stability of the reactor:

   a. State the stability margin criterion for the reactor under various operating conditions as used in Section V C.5.0 of the PHSR.

   b. Describe the means to be employed, in addition to the proposed analog studies, to determine the margin of reactor hydraulic and nuclear stability at the design operating conditions.

   c. Provide data or experience obtained with operation at average voids of 35% and average exit voids of 58%. What is the expected maximum void fraction at the exit of the hot channel under minimum subcooling conditions?

   d. State the acceptable range of void coefficients as a function of core life that will provide the required stability.

   e. Indicate the manner in which a higher average exit void fraction (58%) than used on other boiling water reactors may effect the reliability of the analog computations of stability.

   f. Discuss the possibility of the occurrence of Xenion oscillations, their effects on peaking factors, and how the oscillations, if they occur, will be controlled.

2. Discuss the variation of the temperature and Doppler coefficients of reactivity as a function of core life. In particular, give design criteria regarding limiting values of the reactivity coefficients throughout core life, and for those cases when the core is only partially loaded.

3. Describe the parameters that will be measured during the contemplated critical experiment work.

4. State the basis for determining that 3% shutdown margin is adequate.

SECTION VI

STATION SITE AND ENVIRONMENT

1. Discuss diffusion conditions over the lake as compared to those over land.

2. State the width of the sector for which the wind persistence data is given.

3. State diffusion parameters representing the worst cases for times between one day and thirty days.

4. Since section D includes both land and water trajectories, divide it into at least two cases.

5. State the criteria which was used to select the "worst case" for thirty day average meteorological conditions.

6. In Table A-5.4, state the total hours, and for what month the data was taken.

7. Provide a breakdown of wind speed data in the range of zero to ten miles per hour for the data given in Figures A-5 and A-14 of Volume II of the PHSR.

8. In Figure A-7 of Volume II of the PHSR state why the end points do not agree with each other.

9. State the distance from the site to the nearest known fault.

10. State the distance and location of the nearest permanent residence.

11. State the distance to the nearest land where dairy cows graze or are likely to graze in the future.

12. Re-evaluate the dilution factors which may be expected at the Oswego City intake. The factor of 50,000 appears to be in error.

SECTION VII

SAFETY ANALYSIS

1. For the loss of coolant accident, provide a graph of drywell and suppression pool pressure as a function of time after the accident assuming:

   a. Fission product inventory and time of shut down equivalent to paragraphs b and c of Section VII F.2.2.1 of the PHSR.

   b. Core spray does not function.

   c. One post incident cooling loop does not function.

   d. Both post incident cooling loops do not function.

   e. Core spray and post incident cooling loops do not function.

   Provide a list of all other assumptions which were used to establish these curves and the reason for each.

2. Submit an analysis of the "critical" intermediate sized rupture, which is larger than the capacity of all "make-up" water supplies, but results in the reactor vessel pressure remaining above the cut off point of the core spray system. Assume the feedwater pumps are supplying water at the rate governed by the high water level indicator in the water column which results from boilout in this water column. (Critical rupture means the rupture resulting in the largest dose under TID 14844 criteria.)

3. Analyze the probability and consequences of a rod or group to rods moving slowly into the core and the recirculating flow rate or control rod position being altered to compensate for the reactivity change.

4. Discuss the ability of the main steam line isolation valves to function properly with the flow of steam and slugs of water which might occur in the event of severence of a main steam line at a point external to the vapor suppression system.

5. Assuming the occurrence of a nuclear excursion violent enough to release fission products from the fuel, discuss the containment of the fission products reaching the condenser and off-gas holdup system and the possibility that the turbine exhaust hood diaphragm might be blown out as a result of the excursion.

6. In the event of a main steam line break provide the following information:

   a. The basis for assuming a fission product inventory proportional to that measured in Dresden two hours after sampling when the dose to the off-site receptor would be received within about 15 minutes.

b. The basis for assuming 60% of the discharged coolant would remain as water and rain out in the pipe tunnel.

c. The basis for assuming the specific volume of 1000 cu.ft/lb of cloud. Evaluate off-site doses as a function of cloud size.

d. The basis for excluding the contribution of noble gases from the off-site dose calculations.

e. Explain why the assumptions made in the analysis of this accident would be the most severe in terms of off-site doses. Would, for example, different assumed meteorological parameters provide significantly different doses?

7. Provide the basis for selection of 2.5% for the maximum worth of an individual control rod. In support of the acceptability of this maximum worth, analyze the consequences of a break in a control rod thimble and the consequent control rod ejection. Consider the effects of direct transfer of heat from fuel to water by dispersion of UO_2 from failed fuel elements and the effects of mechanical reactive forces on reactor fuel (including possible positive reactivity input) and control elements. In addition, the energy input from the zirconium-water reaction and the hydrogen gas released from this reaction should be taken into account. Interest should be centered on both the good and poor heat transfer cases. The starting point for this accident should be full power and end of core life. In each case analyzed, the assumptions and results of the analyses should be provided, including the following:

a. Nuclear constants and other variables chosen conservatively within the range permitted by the design criteria.

b. Heat transfer coefficients.

c. Fuel temperature history.

d. Reactor power history.

e. Reactor pressure history.

8. Discuss the magnitude of the energy and pressure contribution of any zirconium-water reaction to the drywell pressure as a function of time for the 100% core melt case and its effect on the drywell leakage rate. Analyze the effect on the energy and pressure contribution of a zirconium-water reaction due to delayed effectiveness of the core spray system.

9. With reference to the loss of coolant accident, re-evaluate consequences of this accident assuming (1) a contribution from fission products contained in the gap of the fuel rods, (2) amount of fission products released would be greater than the assumed 10% due to varying burn-up in the core, and (3) exfiltration from the reactor building at various wind speeds.

10. Analyze the off-site consequences of a failure of reactor auxiliary system piping (such as an emergency condenser) in the reactor building. Include an evaluation of the temperature and pressure reached in the reactor building considering the adequacy of sensors which initiate system isolation.

11. As a result of a loss of coolant accident, it would appear that the rapid depressurization during blowdown introduces the possibility of clad failure and carryover of fission product activity into the pressure suppression chamber during blowdown. An additional carryover of fission products would be due to the activity contained in the coolant at the time of the failure. Provide an estimate of the quantities involved, and their effect on the resulting doses.

12. Analyze the consequences of the maximum credible accident and the loss of coolant accident if they occurred concurrent with rain.

13. With reference to the Refueling Accident:

    a. List the important physics and thermal parameters used establishing the 2700 MW-sec power transient.

    b. Provide the bases for your conclusions that 300 lbs of UO_2 will reach temperatures in excess of 8000°F, 2500 lbs of UO_2 will reach temperatures in excess of 5000°F, and about 400 fuel rods will reach temperatures in excess of the estimated threshold for fuel rod burnout.

    c. Define in greater detail the assumptions needed to establish the source of the assumed fission products released to the reactor water.

    d. Justify the assumed 10^{-4} ratio of air to water concentration of halogens and state why it is justifiable to assume an air-borne halogen concentration significantly different from TID-14844 assumptions.