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U. S. ATOMIC ENERGY COMMISSION

DIVISION OF REACTOR LICENSING

REPORT TO ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

IN THE MATTER OF

NIAGARA MOHAWK POWER CORPORATION

APPLICATION FOR CONSTRUCTION PERMIT

DRL 5/A 9/21/64

Note by the Director, Division of Reactor Licensing

The attached report has been prepared by members of the Division of Reactor Licensing for consideration by the Advisory Committee on Reactor Safeguards at its October, 1964 meeting.

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CONTENTS

		Page
I	Introduction	1
II	Summary	1
III	Safety Considerations	2
	A. Site	2
	1. Water Usage	3
	2. Meteorology	4
	3. Geology and Hydrology	4
	4. Seismology and Seismic Design	5
	5. Effect of Liquid Effluent on Aquatic Life	7
	6. Environmental Monitoring	7
	B. Containment	8
	C. Engineered Safeguards	11
	1. Isolation Valves	11
	2. Containment Heat Removal Equipment	11
	D Electrical Power	13
	E. Instrumentation	14
	F. Recirculation Flow Control	15
IV	Safety Analysis	16
	A. Routine Operation	16
	1. Radioactive liquid waste	16
	2. Radioactive gaseous waste	17
	3. Radioactive solid waste	17
	B. Accident Evaluation	18
	1. Steam Line Break Accident	18
	2. Loss of Coolant Accident	19
	3. Refueling Accident	21
	4. Control Rod Ejection Accident	22
	5. Zirconium-Water Reactions	23
V	Conclusion	25

I. Introduction

The Niagara Mohawk Power Corporation applied for a Construction Permit on April 1, 1964. In support of this application, a report entitled Preliminary Hazards Summary Report Nine Mile Point Nuclear Station was submitted for review. In this case, the applicant is Niagara Mohawk Power Corporation and the General Electric Company is the nuclear subcontractor. Niagara Mohawk will act as its own architect-engineer.

In addition to the staff technical review, meetings were held between the applicant and the Reactor Licensing staff on June 3 and 4, 1964. An ACRS Subcommittee meeting was held near the proposed site in Oswego, N. Y. on June 10, 1964. As a result of the discussions during these meetings, questions were sent to the applicant on July 10, 1964. Answers to these questions were received on August 17, 1964 and were incorporated into the First Supplement to the PHSR. A further meeting with the applicant was held on September 17, 1964, to discuss the answers to the July 10, 1964 questions as well as some additional technical problems.

II. Summary

The Niagara Mohawk Power Corporation proposes that the Nine Mile Point facility be located on an approximately 1500 acre site on the southeast shore of Lake Ontario, approximately 7 miles northeast of Oswego, New York.

The reactor is to be a 1538 Mw(t) boiling water direct cycle unit from which approximately 525 Mw(e) gross will be generated. The unit will be installed in a pressure suppression containment structure similar to that used in Humboldt and proposed for Bodega and Jersey Central. This structure will be designed to withstand the pressure which would result from a release of the thermal energy which is stored in the primary system. With active engineered safeguards, it will also control over an extended period of time the temperatures and pressures which would result from release of fission product decay energy. A reactor building will be built above the containment structure and leakage from the containment structure will be into the reactor building.

The applicant has considered the potential consequences to the environs of a number of accidents, including a main steam line break, a loss of coolant accident, a control rod ejection accident, and a refueling accident. Although we do not agree with every detail of the applicant's assumptions used in evaluation of the consequences of these accidents, we believe that the plant can be operated within the requirements of 10 CFR 20 and 10 CFR 100.

We have reviewed in some detail the requirements and design criteria for engineered safeguards which would maintain the facility in a safe condition in the event of a serious accident. We believe they are adequate and consistent in principle with those already considered acceptable for other facilities of a similar size with concrete containment structures, provided an additional drywell heat removal system is added. We understand that the applicant is considering such an addition.

We have identified the following problem areas which we believe should receive the attention of the Committee and which are discussed in following sections of this report:

1. Containment

   a. Criteria for time of ventilation isolation valve closure.
   b. Retest pressure.
   c. Combustion of hydrogen evolved from a potential zirconium-water reaction.

2. Adequacy of engineered safeguards to remove fission produced decay heat from containment.

3. Recirculation flow control.

4. Inclusion of a period trip in the nuclear instrumentation.

5. Means of limiting the consequences of a control rod ejection accident.

III. Safety Considerations

The proposed Nine Mile Point facility is similar or identical in major respects to both the Bodega Bay and Oyster Creek facilities which have been reviewed recently by the Committee. The same general problems concerning the facility design are in evidence in the proposed Nine Mile Point facility. However, since the applicant, Niagara Mohawk Power Corporation, has not previously appeared before the Committee, and since this application is being considered on its own merits, the principal safety considerations are discussed in this report even though they are similar to those considered for Bodega and Oyster Creek.

A. Site

The Nine Mile Point site occupies approximately 1500 acres of land in a rectangular plot on a broad promontory on the shore of Lake Ontario, seven miles northeast of the City of Oswego in Scriba township, Oswego County, New York, and 36 miles northwest of Syracuse, the nearest large city to the site. The site is generally flat with a gentle slope toward the lake. The station elevation will be 260 feet MSL, and the normal lake level is 246 feet. About 200 acres along the lake shore in the central portion of the site is cleared of trees, and all structures of the proposed facility will be erected in this area, with the reactor located approximately 400 feet from the lake shore. The shortest distance to the site boundary will be 4000 feet to the west, and the nearest residence will be on Lake View Road, approximately 5200 feet in the same direction. Distances to the other boundaries are over a mile to the east, and about a mile and half to the south. The land area surrounding the site is largely used for pasture and light agriculture. The following tabulation shows the population distribution within 20 miles of the site.

TID 1500 MW .88 have .76
elel 13.3 7
LPZ 17.7 (35)? to Syracuse
CD 7 to Oswego

POPULATION WITHIN TWENTY MILES OF THE REACTOR BUILDING

1960 Census Figures

Distance (Miles)	Population in increment	Total population within radius
1	0	0
2	300	300
3	500	800
5	1,100	1,900
10	28,900	30,800
20	44,000	74,800

As can be seen from this table, and scrutiny of the data presented in the PHSR, the population density of the area surrounding the site is low, being less than 100 per square mile in all sectors within five miles. The nearest village, Lycoming, is 2 miles from the site and has a population of 125. The next nearest significant population is the City of Oswego at a distance of seven miles, with a 1960 population of 22,155. Adjacent to the nearest site boundary (to the west), along the lake shore, is a 12 acre plot known as Lakeview, operated as a summer camp eight weeks per year. There are up to 500 people at the camp during the week, and 1500 on week-ends. There is a scattering of summer cottages along the lake shore, the nearest of which is about 8000 feet to the east.

? .88

A comparison of the 4000 feet exclusion radius available with the TID 14844 exclusion distance of approximately 0.92 miles indicates that engineered safeguards more effective than assumed in TID 14844 by a factor of approximately 7 will be required in order for this reactor to be within the guidelines set forth in Part 100. Similarly, a factor of 21 will be required for this reactor to be within the guidelines set forth in Part 100 regarding the city distance, if the City of Oswego (population 22,155) is used for a reference. Otherwise, the factor will about 2.

$\frac{17.7}{7} - (2.5)^{1.5} = 4$ Syracuse = 36 mi? TID = 17.7

1. Water Usage

Lake Ontario, one of the nation's major water resources, is used as a source of water supply by various municipalities and industries along the shore, for shipping, for commercial and sports fishing, and for extensive recreation. The nearest public water supply intake is for the city of Oswego, and is located eight miles west of the proposed facility. The mouth of the Oswego River empties into the lake between the plant and the Oswego water supply intake. As described later, the temperature and current patterns established by this river, as well as the breakwaters extending into the lake near the river mouth, would tend to provide protection to the water intake from reactor effluents that might be released into the lake in the condenser cooling water. It is anticipated that use of the Oswego intake

will be expanded to supply more of Onondaga County, including the City of Syracuse. By the time this has been accomplished, this intake may very well furnish an essential and irreplaceable source of water for a very large number of people. Other intakes from the lake are over 35 miles away, and do not represent any problem

OK?

2. Meteorology

A long period of record of wind direction and velocity is available from a station ten miles from the site in Oswego. In addition, one year of data has already been taken and analyzed at the site. It has been found that the average wind velocity at the 203 foot level is 14.7 mph, and the frequency of calm is zero, indicating that the wind velocity at this site is generally somewhat higher than it is in central parts of the country. The maximum expected wind velocity is estimated to be 110 mph. Due to the influence of the lake, the frequency of inversion varies drastically during the year, from a minimum of 6% in the winter to 75% in June, with an annual average of 35%, similar to many other sites. The frequent passage of vigorous low-pressure systems tends to prevent the occurrence of long periods with constant wind direction. During 1963, there was only one occasion when the wind persisted in one octant for more than 45 hours, and three occasions when it persisted for more than 36 hours. The data has been analyzed by the applicant with respect to wind direction, speed, and stability for one year, and appropriate diffusion parameters for one hour, five days, and thirty days derived therefrom.

The U. S. Weather Bureau has examined the meteorological data for this site and has advised that the diffusion parameters utilized by the applicant are appropriate. However, they also suggested that wind Sector D be reduced in size to examine the differences between over land and over water trajectories, as well as the possible marked differences in wind frequencies within the sector. This was done in the First Supplement to the PHSR in what the staff believes to be a satisfactory manner. The Weather Bureau also suggested the possible occurrence of other diffusion conditions not analyzed by the applicant which might have some effect on the annual average stack emission limit. However, preliminary estimates by the applicant and by the staff indicate that the stack limit will be large enough to accommodate any small adjustments which might result from such considerations in the future.

3. Geology and Hydrology

Bedrock at the site is Oswego sandstone. The overburden above bedrock averages about eleven feet at the boring positions. It is estimated that the Oswego sandstone extends down to about 185 feet below the ground surface in the general station area. The Oswego sandstone grades into the Lorraine group, the Trenton group, and the Cambrian sediments between approximately 85 and 1700 feet. All

major structures will be founded on the Oswego sandstone, which is a very substantial rock and is resistant to weathering.

A detailed study by the applicant of the surface drainage patterns in the area of the site has shown that all surface water will run off into Lake Ontario, or percolate down a short distance and then move toward the lake in the underlying soil and rock.

A detailed study of lake currents in the vicinity of the site has been made. This was accomplished by using both drift cards and the clorine laden water from the Oswego River. With these tracers, it was possible to relate the surface lake currents, which would carry the plant effluent, to surface wind conditions, and thereby predict the frequency of occurrence of various lake patterns. It was found that the predominant lake current is to the east from the plant, turning northward near the eastern end of the lake. The minimum yearly average dilution calculated at any beach due to this current is twenty-one. Most of the time it will be impossible for plant effluent to move across the mouth of the Oswego River to the Oswego water intake to the west. It is estimated that it will be carried to this point by the winds only four percent of the time during a year, and that the dilution in the lake will be a factor of twenty even when this happens. These two factors would give an average dilution factor of 500. In addition, it is estimated that the diffusion effect due to the submerged location of the intake will produce an additional factor of 100 dilution. Hence, there is a possibility that the concentration of radioactivity in the effluent could be permitted to exceed Part 20 without producing excessive concentrations in the nearest supply of drinking water. However, such a proposal has not been made by the applicant at this time.

The maximum measured level increase in Lake Ontario due to a seiche is 2.9 feet. Wave heights as great as twelve feet have been measured, and at this location these can come up to the shore before breaking. Thus, it appears that the site elevation of fourteen feet above the average lake elevation is just sufficient to prevent flooding during severe conditions.

We have requested that the U. S. Geological Survey comment on the geologic and hydrologic information for this site. Copies of these comments will be made available to the Committee when received.

4. Seismology and Seismic Design

There is no evidence of folding or faulting at the site or in the regional geology, and the region is stated to be one of the most stable areas in the county. Historically, the nearest earthquake occurred fifty miles to the east at Lowville, New York, and resulted in insignificant acceleration at the site. This is also the location of the nearest known fault. The next nearest quake occurred at Attica,

New York, 110 miles west of the site. Its magnitude of six would have produced an acceleration of only 0.01 g at the site. The applicant has estimated that the largest shock likely to occur in the region would be a Magnitude 7 shock, such as occurred in 1925 in the St. Lawrence River Valley at a distance of 280 miles northeast. If this size shock were placed at Lowville, the ground acceleration at the site would be approximately 0.11 g.

The intensity of ground motion which would be expected at the site was calculated by a method developed by Kanai. This procedure takes into account earthquake magnitude, epicentral distance, and the elastic properties of the soil and rock in the area. Using this method, the ground motion response spectrum for the Oswego sandstone, the material on which all principal facility structures will be founded, was developed and is given on Plate C-22 of the First Supplement to the PHSR. A maximum ground acceleration of 0.11 g was calculated.

The U. S. Coast and Geodetic Survey has been requested to examine the seismicity aspects of the site and has estimated that a ground acceleration of approximately 0.07 g is the maximum that has ever occurred in this vicinity. On the basis we believe the applicant's assumptions are adequately conservative.

The Class I structures, those whose failure could cause a significant release of radioactivity or which are vital to the safe shutdown and isolation of the reactor, will be designed in accordance with the following criteria:

1. Functional load stresses resulting from normal plant operation, when combined with stresses due to the seismic accelerations specified in the ground motion response spectrum, shall be within the working stress for the particular material involved.

2. In the case of the containment structure, primary load stresses plus those resulting from pressure, temperature and seismic accelerations specified in the ground motion response spectrum shall be within working values.

These criteria are in most important respects consistent with the seismic design criteria set forth for the other large power reactors now in some phase of the Commission's licensing procedure. Typically two earthquake accelerations have been specified; namely, an acceleration that has occurred in historical time, and also a higher acceleration that theoretically could occur. In all cases the facilities are being designed to ride through the "historical" earthquake with combined normal operating, accident and seismic stresses on all components, structures, and systems important to safety at or near the established working stress of the material under consideration. For the higher acceleration theoretical earthquake, no loss of function of all systems important to safety is permitted; however, this implies

Bldg code Richter VIII IX reasons

Zone 3 can have quakes up to well above 0.11g

? 0.5g & above

that in some cases stresses may reach to or beyond the yield stress of the material involved.

In the case of Niagara Mohawk, they have elected to specify only a design acceleration corresponding to the theoretical maximum earthquake (0.11g) but are designing the facility in accordance with the criteria normally specified for the "historical earthquake." This conservative approach is, of course, acceptable. Based on our evaluation of the advice from the U. S. Coast and Geodetic Survey concerning the maximum ground motions that have occurred in the past, we believe that the applicant's earthquake design criteria are adequate for the seismic conditions likely to be encountered at this site. ?

5. Effect of Liquid Effluent on Aquatic Life

The applicant has not established at this time a numerical limit on the amount or concentration of radioactivity to be released to Lake Ontario, but intends to base this limit on 10 CFR 20, taking into account the composition of the discharge and the dilution and concentration mechanisms in the Lake. The study of the dilution character of the Lake previously discussed, and the applicants proposal to carry out ecological and radiological studies on the marine life will be important in establishing this release limit. It is our understanding, based on discussions, that the applicant is maintaining liaison with officials of New York State on the radioactivity aspect of their proposed operations.

We have requested comments on the effects of liquid effluents on fish and shellfish from the U. S. Fish and Wildlife Service. Copies of these comments will be made available to the Committee, when received.

6. Environmental Monitoring

An environmental monitoring program will be initiated by the applicant during the construction phase prior to operation of the plant. This monitoring will continue into the operational phase of the plant. This program will delineate background radioactivity levels, prior to operation, and provide a basis for evaluating the effects of the operation of the plant, if any, on the environment.

B. Containment

The containment system proposed for the Nine Mile Point plant is a pressure suppression system similar to that installed at Humboldt Bay, and proposed for Bodega Bay and Oyster Creek. The proposed system consists of two chambers, the drywell in which the reactor vessel and recirculation equipment will be situated, and the suppression pool to which the drywell atmosphere will vent in case of a pressure rise.

The drywell will be a spherical vessel 70 feet in diameter with a 33-foot diameter cylindrical top. It will be designed and fabricated in accordance with Section III of the ASME Boiler and Pressure Vessel Code of ASTM A-212 Grade B firebox steel. The design pressure of the drywell will be 62 psig at a temperature of 280°F.

The suppression chamber is a toroid in form, the centerline diameter of which is 123 feet. The cross section is circular and 27 feet in diameter. It will be approximately one-half filled with water. As with the drywell, it will be designed and fabricated in accordance with Section III of the ASME Boiler and pressure vessel code. This vessel will be designed for a pressure of 35 psig.

Communication between the drywell and the suppression chamber will be by 10 large diameter vent pipes, a manifold in the suppression chamber, and finally a large number (approximately 120) of downcomers which are submerged in the suppression pool water.

The specification of the Nine Mile Point containment design pressures is based upon an analytical model developed from Moss Landing test results. The design pressures are based upon consideration of a double ended break of one of the recirculation lines which results in a blowdown of the fluid in the primary system. It has also been assumed that a relatively small steam leak has prepurged the drywell of air. This analysis yielded a drywell pressure of 31.6 psig (or 46.6 psig with prepurge) and a suppression chamber pressure of 25.3 psig.

Figure VII-1 in the PHSR is a graph of the calculated drywell pressure as a function of time after the accident. The suppression chamber should be at approximately the same pressure since the two chambers are connected by vacuum breakers. In addition, the suppression pool will be connected to the reactor building by vacuum breakers.

Based upon our review of the design of the containment system as well as the assumptions used in specifying the design pressures, we believe the containment system will be able to withstand the immediate consequences of all predictable accidents.

Two airlocks for personnel access will be provided in the drywell. The doors of each airlock will be mechanically interlocked so that

Bellows valve disch into drywell?

only one door may be open at a time. No plans are made for personnel access during power operation; however, limited access will be permitted during hot shutdown.

There will be three personnel access openings to the suppression chamber. These will be sealed with double gasketed bolted covers.

Although the drywell ventilation system will be normally closed during operation, the ventilation ducts will be equipped with dual isolation valves in series which will be closed by a high radiation signal. The applicant is now considering closing the suppression pool ventilation isolation valves at a set time delay after receipt of a high drywell pressure signal. The advantage of such a scheme would be that the containment system could blow down to the atmosphere thus lowering pressures. The disadvantage of this scheme is that the containment atmosphere is allowed to blow down freely to the atmosphere concurrent with an accident, a time when nothing is predictable. In view of the near impossibility of guaranteeing that there would not be a concurrent release of fission products with overpressure in the suppression pool, we believe that all ventilation isolation valves to the containment should be closed immediately by the same signals which initiate such emergency functions as closure of main steam lines isolation valves and closure of other isolation valves. Typically such signals would come from such sources as reactor water level, condenser vacuum, or containment pressure.

All fluid lines penetrating the containment barrier will have one or two isolation or check valves, depending upon the function of the line, which will close upon receipt of such signals as high drywell pressure, low reactor water level, and low condenser vacuum. Signals will be received from two duplicate sensors in each of two independent channels. The function will be energized by either of the sensors in both channels. Power for the channels will be provided by independent power sources. Power to close the isolation valves will be supplied by the station battery or stored mechanical energy. We believe that this isolation arrangement is acceptable.

At the present time, the applicant states that they are considering providing the capability of testing of individual penetrations on some regular interval. This would include capability for pressurizing between the airlock doors and providing double gasketed seals with testing capability on the manhole covers in the suppression pool, the drywell head, and the electrical and mechanical penetrations. Taps for pressure testing between the ventilation isolation valves will also be provided. We believe that such testing capability would be highly desirable and should be installed.

The specified leakage rate for the combined drywell and suppression pool is 0.5% per day at the suppression pool design pressure (35 psig). This is to be demonstrated during an initial leakage rate test with all penetrations installed at a series of pressures up to 35 psig so as to determine leakage characteristics as a function of pressure.

Retests are planned at a pressure of at least 5 psig. We believe that unless a retest pressure higher than 5 psig is used, it will be difficult to extrapolate the measured leakage rate to the specified design pressure of 35 psig; however, this point need not be settled at this time.

As with Jersey Central, the possibility of a metal-water reaction as a consequence of an accident cannot be completely discounted. If this reaction were to occur, the evolution of hydrogen would commence at some time after the initial pressure peak in the drywell and would not alter the design pressure. However, the pressure would not decay to zero as shown in Fig VII-1 but would approach some higher value due to hydrogen evolution. Another consideration is the possibility of burning or explosion from the excess hydrogen in the containment. We believe that it is not necessary to solve these problems at this stage of the licensing proceedings; however, study should continue through construction to assure that appropriate design or equipment considerations are installed to prevent a hydrogen-oxygen reaction which would violate containment integrity.

The drywell for this proposed facility is relatively small in size and surrounded by an insulating shell of concrete. As such, it shares the problem of all power reactors which are situated in concrete containment systems. That is, means must be provided to remove the fission product decay heat energy through all phases of operation, and principally after accidents. If this capability does not exist, the integrity of the containment system after an accident cannot be guaranteed. This problem will be discussed in greater detail in the "Engineered Safeguards" portion of this report.

The reactor building will be a reinforced concrete or steel frame structure with reinforced concrete walls. It will be situated directly above the containment system. The majority of piping, electrical and personnel penetrations will communicate with this building. The exceptions are 2 main steam lines, 2 feedwater lines, and 4 extra penetrations which communicate with the turbine building.

In the event of an accident, an emergency ventilation system capable of one air change per day will serve the reactor building and will provide a controlled path of release from the reactor building, via filters, to the stack. Two complete full-capacity sets of equipment will be installed. The building is being designed for an inleakage of 100% per day with a pressure differential of 1/4 inch of water. Thus, all radioactive materials leaking from the containment system, except those which may leak via the penetrations to the turbine building, will enter the reactor building and subsequently be exhausted through charcoal and absolute filters to the stack.

Except for those areas of concern noted earlier, we believe that the design of the containment system, including the reactor building, is adequate to fulfill its intended purpose.

C. Engineered Safeguards

In the event of an accident, and as noted previously, in order for the containment integrity to be maintained, two active engineered safeguards must function. These are isolation valves and containment heat removal equipment. The fact that the proposed drywell is relatively small and surrounded by concrete makes it mandatory that means be provided for removal of fission product decay heat. Specifically, the two active engineered safeguards are designed as follows:

1. Isolation Valves

Isolation valves will be installed in all fluid lines which penetrate the drywell and suppression chamber. Two valves in series which will close automatically, or by manual actuation, will be installed on lines which communicate directly with the nuclear steam supply system or which open to the containment atmosphere. On inlet lines, one of the two valves may be a check valve. Lines which form a closed loop inside the containment but which could discharge radioactive material outside as a result of a pipe failure, will be equipped with one isolation valve. [illegible] lines connected to the nuclear steam supply system will [illegible] isolation valve which will be closed by manual actuation. [illegible] to these criteria are as follows:

a. No isolation will be provided on the control rod drive hydraulic system lines.

b. No isolation valves will be provided in the post-incident cooling system and in the lines between the suppression pool and core spray.

Function of the ventilation system isolation [illegible] discussed in the containment section of this [illegible] report has maintained the criteria that the [illegible] individual items of equipment [illegible] isolation valves will be redundant. Power for [illegible] isolation valves will be supplied by the plant battery [illegible] stored mechanical energy.

We believe that the criteria for installation and actuation of the isolation valves, except as discussed in the Containment Section of this report for ventilation system valves, are adequate and consistent with previously approved reactors of similar nature.

2. Containment Heat Removal Equipment

The containment heat removal equipment consists of the core spray system and the suppression pool cooler.

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a. No isolation will be provided on the control rod drive hydraulic system lines.

b. No isolation valves will be provided in the post-incident cooling system and in the lines between the suppression pool and core spray.

Function of the ventilation system isolation valves is discussed in the containment section of this report. The applicant has maintained the criteria that the sensing devices and other individual items of equipment required for actuation of the isolation valves will be redundant. Power for closure of the isolation valves will be supplied by the plant battery or by stored mechanical energy.

We believe that the criteria for installation and actuation of the isolation valves, except as discussed in the Containment Section of this report for ventilation system valves, are adequate and consistent with previously approved reactors of similar nature.

2. Containment Heat Removal Equipment

The containment heat removal equipment consists of the core spray system and the suppression pool cooler.

The core spray system as now visualized by the applicant will consist of two independent full capacity units. In each unit, water will be drawn from the suppression pool by a core spray pump which will discharge to a spray header in the reactor vessel. Power for each pump will be supplied by an independent bus. One pump will be automatically switched to emergency power upon loss of outside power. The other may be manually switched to emergency power. The system will be actuated automatically by a low reactor water level and will function when the system pressure is below 150 psig.

The design criterion for the core spray system is that sufficient cooling be provided by either unit to limit the fuel temperature to prevent rapid oxidation of the zirconium fuel cladding. A minimum flow of approximately 1850 gpm will be required to meet this criterion.

The suppression pool cooling system will consist of two independent full capacity units. Each will consist of a post incident cooling pump of about 2000 gpm capacity and a post incident heat exchanger. Each heat exchanger will be capable of removing from between about 3 and 13 Mw of decay heat energy depending upon the suppression pool temperature. This appears to be barely adequate, since the decay heat energy will be in the order of 13 Mw within 3 hours of shutdown. The heat sink for the post incident heat exchangers will either be the service water system or the raw water cooling system. Power will be available from the facility diesel generator to operate this post incident cooling.

With reference to the interrelation between the core spray and suppression pool cooling systems, it should be noted that both of these engineered safeguards must function to maintain the integrity of the containment system in the event of a steam line break accident or a loss of coolant accident. This results from the fact that the only method of rejecting heat from the containment system is through use of the suppression pool coolers. To underline the importance of these systems to public health and safety, we requested in Question 1 under Section VII, an evaluation of containment system pressure as a function of time assuming that certain combinations of active engineered safeguards (i.e. core spray and post incident cooling loops) do not function. The principal result of this evaluation is given in Figure VII-10 in the First Supplement. Note by reference to this figure and the text that in order to guarantee the integrity of the containment beyond 40,000 sec (about 12 hours) it will be necessary that both the core spray and the post incident cooling loops function. There is no back-up to these two systems. Both double

For comparison, consider Connecticut Yankee. Three active engineered safeguards were specified by Yankee, namely: (1) safety injection, (2) air recirculation, and (3) containment spray. Any one of these three, each of which is independent, and of a completely different

concept, would protect the containment. In addition, each of these systems has redundancy provided in major components within the system. Even if the safety injection system is discounted, since it isn't operated by emergency power, one independent redundant safeguard system out of two would protect the Connecticut-Yankee containment.

In view of the potential consequences of a failed containment vessel, we believe that the position which we have already taken with Connecticut Yankee with reference to the independence and redundancy of engineered safeguards to protect containment integrity should be maintained. The position is that two independent and differently designed systems should be provided, and that each of these systems should be redundant in major components and in cooling capacity.

In the case of Niagara-Mohawk, both of the engineered safeguards which act to protect containment integrity must function, and thus the design proposed does not meet this criterion. We believe that there should be some additional provision for removing heat from the drywell. The additional system should be redundant and should be independently capable of preventing containment damage. The applicant is considering such an addition.

D. Electrical Power

Electrical power to the facility will be supplied from the following sources:

1. During normal facility operation, power will be supplied from the station generator via the normal auxiliary power transformer. Two 345 kv transmission lines will deliver the electrical power from the facility to Clay Station, about 30 miles away.

2. During shutdown, or startup when the station generator is not in operation, 115 kv transmission lines from the Lighthouse Hill Hydro Station and from the Oswego Steam Station will supply power to the two reserve buses at the facility. In the event of failure of the normal auxiliary power from the station generator, automatic switching equipment will bring the reserve system into operation. Both reserve buses will have ample capacity to allow the facility to operate at full power. Any one reserve bus will allow an orderly shutdown.

3. Upon failure of normal and reserve systems, a station diesel generator which will have sufficient capacity to supply power for all engineered safeguards and other emergency functions will be available.

4. A final source of power will be the station battery which will supply power to emergency lighting, rod position indication, some instrumentation, and critical controls.

We have considered the adequacy of the station electrical power in terms of reliability and ability to protect the facility under all potential accident situations. Based on these considerations, we believe that the auxiliary and emergency power provisions for this facility are adequate and consistent with that already approved.

E. Instrumentation

The startup range instrumentation consists of three fission counter channels which cover the source range upward to approximately 10^6 counts per minute (10^{-4} percent of full power). Log count rate information is derived and recorded continuously. Period information is derived and indicated continuously and will alarm at a preset short period. None of the channels is connected to the automatic safety system.

One channel of intermediate range instrumentation is provided and consists of compensated ion chamber whose output is fed to a Log N amplifier. The output of the Log N amplifier is recorded continuously and covers the range from 10^{-5} percent to full power. A time derivative of the output is also indicated on a period meter and will alarm at a preset short period. The intermediate range channel is not connected to the automatic safety system.

Niagara Mohawk proposes not to incorporate a reactor scram on short period. We have doubts as to the suitability of a safety system which does not provide period protection. General Electric believes that period scrams are not necessary provided the six high neutron flux channels are adequately sensitive. Analysis indicates that with sufficient sensitivity the proposed linear safety channels can provide protection essentially equivalent to the "log-N" period channels ordinarily incorporated into the reactor instrumentation design. This matter will be considered during the detailed design and construction phase of the project.

Six channels of instrumentation are provided to monitor reactor neutron levels from 10^{-6} percent to 125 percent of full power. Each channel consists of a gamma compensated ion chamber whose output is fed to a linear, variable range picoammeter. The picoammeters can be stepped nineteen ranges, each range change being the square root of ten. Three of the picoammeters are connected to one logic channel of the dual logic channel automatic safety system, and the remaining three are connected to the other logic channel. A warning will be annunciated if any one of the picoammeters should exceed a preset percentage of scale reading and a scram will occur if at least one picoammeter in each of the two logic channels exceeds the preset percentage of scale reading. In addition, miniature ion chambers are to be installed in the reactor core to provide information concerning the flux distribution. In-core chambers will also be provided and connected to scram circuitry during reactor fuel loading.

The automatic safety system contains two logic channels, both of which must de-energize to produce a reactor scram. Each channel is subdivided into two independent subchannels (except for voltage source). Tripping of a subchannel by elements of the nuclear and process instrumentation is tantamount to tripping the entire channel, and tripping of the second sub-channel within a channel constitutes a back-up action. A subchannel trip de-energizes two parallel-connected relay coils contained in the subchannel, thereby de-energizing one of the two solenoid operated pilot valves associated with each rod. Operation of only those pilot valves associated with one channel will not initiate a scram. When both channels are tripped, both solenoid pilot valves are actuated and remove air pressure from the scram valves associated with each rod. The scram valves then admit high pressure water to the rods and vent the water which is driven from the system to the dump tank. Solenoid operated "back-up" valves will initiate rod motion in the event of failure of one or more pilot valves.

The logic scheme is classed by General Electric as being dual channel requiring the coincident tripping of two out of two logic channels in order to initiate a scram. It can, however, be considered to be a two out of four or a three out of four scheme depending on which subchannels trip. This scheme protects the reactor against a spurious scram signal originating in one channel and possesses sufficient redundancy to tolerate a single "non-safe" failure within one subchannel.

In our review of the proposed safety system a number of questions have arisen which have not as yet been resolved with the applicant. These involve such items as (1) the physical separation which will be maintained between subchannels to assure that failure in one subchannel will not effect another subchannel, (2) adequate methods of testing the logic scheme including testing of the pilot valves during operation, (3) provision for independence of the manual scram from the automatic scram system, (4) provisions for voltage and signal loss monitoring and (5) adequacy of the variable range picoammeters in lieu of period channels connected to the automatic safety system. It is, however, our opinion that these can be resolved with the applicant during final design and construction of the facility.

F. Recirculation Flow Control

The reactor power level is to be adjusted by positioning control rods and by varying the recirculation flow. The rods will be positioned manually in accordance to a program.

The recirculation system as now contemplated will consist of 5 separate loops. Each will take suction from the downcomer region and will discharge to the inlet plenum below the core. The flow in each loop may be varied from 30% to 100% full flow. This will be accomplished by using induction motors to drive the pumps. The

frequency of the AC power used to drive the pump motor will be varied by use of a drive motor connected to an AC generator by a variable hydraulic coupler. Flow control will be manual. The applicant is also considering, in the event of a severe decrease in distribution system frequency, use of an under frequency signal which would increase flow.

For this application, we have the same problem which was related in the Jersey Central report. That is, during periods of operation at reduced recirculation flow, the overpower scram set points are not correspondingly reduced. Thus, if through operator error in control rod positioning the power level were increased to the scram set point of 120% rated power while the recirculation flow remained low, some burnout of fuel would occur. In response to Question III-13, the applicant has calculated that under these circumstances, some 7% of the core would experience burnout. The applicant's present position is that if the BOR under these conditions (the currently accepted value is 1.5) is calculated to be below the operational limit, they will provide other than procedural controls. We agree that this is a resonable position, and that a more definitive solution can be established later in the licensing procedure.

IV. Safety Analysis

For convenience, the effect of operation of the proposed facility on the public health and safety is divided into two categories, namely routine operation and accidents.

A. Routine Operation

1. Radioactive liquid waste

All potentially radioactive liquid waste will be discharged to the lake after dilution with the main condenser cooling water stream of 240,000 gpm. A continuous sampling device will collect a side stream of the effluent for periodic analysis.

The criteria for discharge to the lake is that the concentrations will either be limited to 10^{-7} uc/cc at the point of discharge, or the effluent will be analyzed to assure that the concentrations of limiting isotopes are below those specified in 10 CFR 20.

The liquid waste holdup time will vary depending upon the particular operation in progress; however, the applicant has specified that on the average a one day holdup time would be available.

2. Radioactive gaseous waste

The majority of the potentially radioactive gaseous waste will be discharged through the facility stack which will be at least 300 feet tall. The criterion for release of these materials is that off-site doses will be within the limits specified in 10 CFR 20.

The principal sources of gaseous activity during operation in the absence of failed fuel is N-13 and A-41. The activity of other short lived isotopes of oxygen and nitrogen will decay to essentially zero, because a half hour delay line between the air ejector and the stack will be installed. If some failed fuel is in the reactor, isotopes of xenon, krypton, and iodine may also be released.

Figure IV-3 in the First Supplement is a graph of relative gas activity as a function of holdup time. This graph demonstrates for noble gases that the half hour holdup decreases the release level from the stack by a factor of about 50. Significant further reduction in noble gas activity level or in halogen level is not achieved unless a holdup time in the order of days is considered.

Other principal sources of gaseous effluents to the stack will be from the turbine gland seal condenser, the reactor building vent air and the turbine building vent air.

Continuously operated instruments will monitor the stack effluent. In addition, provision will be made to collect samples for particulate and halogen analysis. The holdup line to the stack from the air ejector will be monitored and may be valved off if a high level is measured.

Some equipment such as the emergency condensers which may, under accident conditions, become highly contaminated, will be normally vented to the atmosphere. However, these vents will be monitored and will be closed automatically upon a high radiation signal.

3. Radioactive solid waste

Radioactive solid waste will be shipped off-site for disposal.

We have reviewed the design provisions for handling radioactive waste materials resulting from normal operation and believe that they are suitable.

B. Accident Evaluation

If all facets of the Niagara Mohawk pressure suppression scheme operate as assumed and concurrent fission product release and high containment pressures is considered to be incredible, one is hard put to find an accident which would result in a significant health hazard. This is illustrated in Table VII-1 in the PHSR which gives maximum off-site doses calculated by the applicant for various accidents. Rather than accept the applicant's version of the consequences of accidents, we have evaluated the consequences of each of the major accidents by using less conservative assumptions. We have determined, in each case, that the off site doses are within those suggested in 10 CFR 100 provided that some combination of engineered safeguards function. We believe that these safeguards as now proposed by the applicant can be designed to function to such an extent that the 10 CFR 100 guideline doses will not be exceeded.

1. Steam Line Break Accident

This accident is a rupture of one of the 24-inch main steam lines in the pipe tunnel outside the drywell. (In the application it is called the maximum credible accident.) The following assumptions are made to compute the off-site consequences:

1. Isolation valves start closing 2 seconds after break and are completely closed 32 seconds later.

2. 200,000 lbs of primary coolant are released.

3. Outside atmosphere is at 80°F, 40% relative humidity.

4. Radioactivity in the water is as given in Table VII-13 in the First Supplement.

5. Hemispherical shaped cloud 100 meters in radius is formed.

6. Wind speed is 1 m/sec.

With the exception of the cloud shape, these are, in our opinion, appropriately conservative assumptions. A significant increase in the off-site dose may be achieved if a cylindrical or cigar shaped rather than a hemispherical cloud is assumed. The longer cloud would serve to increase the time of passage over the site boundary. Depending upon the cross section assumed, an increase in dose of perhaps a factor of 10 can be computed. Even in this event, the

thyroid and whole body doses (approximately 100 and 3 rem respectively) would be below those given in 10 CFR 100 for serious accidents. The above is not necessarily meant to criticize the applicants assumption, but to illustrate how sensitive the calculation is to changes in this assumption.

We believe the applicant's evaluation of the consequences of this accident, with the exception noted above, is a reasonable appraisal of probable consequences.

For the consequences of the steam line break accident to remain within the limits prescribed above, it will be necessary for the following active "engineered safeguards" to function:

1. The control rods.

2. The main steam line isolation valves.

3. The emergency condenser isolation valves.

4. Instrumentation required for actuation and monitoring the above three functions.

5. Condensate surge and storage systems.

6. A control rod drive feed pump.

7. A power supply.

8. Containment isolation.

We have reviewed these required functions in terms of redundancy, reliability, capacities, and adequacy of emergency power, and believe that in the necessary respects the safeguards should function as required.

2. Loss of Coolant Accident

This accident is a rupture of one of the five recirculation loops. The following assumptions were made:

1. Reactor has been at a power level of 1538 MW(t) for 500 days.

2. Safety system closes containment isolation valves and initiates core spray.

3. Core spray limits core meltdown to 10%.

4. Containment system leaks at a rate as specified in Figure VII-1 in the PHSR (Rate is scaled from 0.5%/day as a function of pressure decay).

5. Post incident cooling functions.

The following active engineered safeguards are assumed to function:

1. Containment isolation valves.

2. Core spray.

3. Controls and instrumentation necessary to actuate and monitor the above.

4. Post incident cooling.

5. Reactor building cooling water.

6. Service water.

7. A power supply.

To place the potential consequences of this accident in perspective, the following table starts with the TID assumptions and consequences, and show, as the various active engineered safeguards are added, the reduction of off-site doses.

Condition	Site Boundary 2 Hr. Thyroid Dose
1. TID 14844 assumptions 0.5%/day leakage	2000 R
2. TID 14844 assumptions, 10% core meltdown 0.5%/day leakage	200 R
3. TID 14844 assumptions, 10% core meltdown, leakage decreases as a function of pressure	170 R
4. TID 14844 assumptions, 10% core meltdown, leakage decrease as a function of pressure, 70% per day exfiltration	10 R
5. Applicant's assumptions	Less than 0.1 R

The above indicates that a factor of 7 to 10 is needed to bring consequences down to those suggested in 10 CFR 100. To do this there is available (1) core spray, (2) reactor building, and (3) the emergency ventilation system. We believe that the combination of these three safeguards will accomplish this.

3. Refueling Accident

The refueling accident is the dropping of a new fuel assembly into a near-critical core. The ensuing reactivity input would be 0.023. The resulting power transient was calculated by the applicant to release 2950 Mw-sec of which 60 Mw-sec is due to zirconium-water reaction. Assuming that the reactor had been shutdown for 24 hours prior to the accident 0.93×10^6 curies of noble gases and 0.70×10^6 curies of halogens were calculated to be released to the water. We find no reason to disagree with these assumptions or with the methods of calculation of the power transient and consequent fission product release.

The following assumptions were then made concerning the amount of noble gas and halogen fission products released to the reactor building:

1. All of the noble gases are released to the reactor building.

2. The halogen concentration in the reactor building is in equilibrium with that dissolved in the water. A ratio of air to water concentration of 10^{-4} was assumed.

3. 100% per day of the reactor building volume is discharged through the emergency ventilation system which has a halogen removal efficiency of 95%.

Redundant system

We believe that the second assumption above is subject to doubt. It would appear that the factor of 10^{-4} would be valid if equilibrium were reached between the iodine concentrations in the water and air and if the released iodine did not come out of the core and through the 50 ft. head above the core in a steam bubble. We believe that it cannot be guaranteed that equilibrium would be reached, nor that a steam bubble would not occur.

The removal of 100% per day of the reactor building volume via the emergency ventilation system requires (1) integrity of the reactor building not be violated and (2) the emergency ventilation system continue to function through the course of the accident.

In order to provide an estimate of an upper limit of consequences to the public without the various mitigating assumptions, we have calculated off-site thyroid doses for a 2 hour exposure assuming a release of 0.70×10^6 curies of halogens and TID 14844 assumptions. These are tabulated as follows:

Assumptions	Dose (Rem, Thyroid)
1. Ground level release from reactor building in short time	30,000 Rem
2. Ground level release from reactor building at rate of 70% per day	2,000 Rem
3. Release from reactor building at rate of 100% per day via stack and filters which are 95% efficient	50 Rem

The above illustrates that unless the integrity of the reactor building is maintained during refueling and the emergency ventilation is operable, doses in excess of those suggested in 10 CFR 100 may be received. We believe that both of these conditions can be met with detailed consideration of design.

4. Control Rod Ejection Accident

During normal operation, to remain within the thermal design criteria stated for this core requires low individual rod worth patterns to achieve allowable peaking factors. Thus, the permissible maximum rod reactivity worth decreases from 0.025 at low power levels to 0.015 at full power, and operating procedures will be necessary to ensure that thermal criteria are met. Because of these limitations, the applicant has stated that the maximum insertion of reactivity resulting from a control rod ejection is 0.015 at full power. The applicant has stated that some other means, in addition to procedural control, such as a rod worth minimizer, or other suitable alternative will be used to limit the reactivity available for insertion by any rod.

Two cases were analyzed, the first being a 0.015 reactivity insertion at full power. Using conservative nuclear parameters, the energy released during the excursion was calculated to be 3820 Mw-sec. In addition, a 0.23% zirconium water reaction yielded 650 Mw-sec and 2000 SCF of hydrogen. This results in a reactor vessel pressure peak of 1130 psia, approximately two seconds after the initiation of the accident. Detailed calculations for the fission product release were not made, but the applicant stated that the consequences would be much less than those for the loss of coolant accident with 10% core meltdown.

The Staff has reviewed the above accident and believes a realistic evaluation has been made. The energy input from this transient was impossible to validate without the use of a computer code; however, the calculated pressure peak was found to agree with the input energy. We also agree that the consequences would be less than the loss of coolant accident.

The second case analyzed was a reactivity insertion of 0.025 at 5 percent rated power. The excursion energy release was not given, but it was stated that the safety valves are open for about 3 seconds at a maximum flow of about 50 percent of their total capacity. The resultant UO_2 dispersion and momentum transfer to core components could not be predicted due to lack of confirmed analytical models and/or direct experimental data describing the modes and efficiency of energy transfer from the fuel to coolant. After completion of the destructive SPERT UO_2 tests and the TREAT capsule tests, it should become possible to provide more detailed answers regarding the core and vessel damage.

LOST

We cannot fully evaluate this accident, because a detailed analysis was not presented. In addition, the UO_2 dispersion and momentum transfer could not predicted. The applicant has stated that if the results of this accident are not acceptable, the maximum amount of reactivity that could be inserted would be reduced by restricting rod patterns. Because there is a means to reduce the amount of inserted reactivity, it is our opinion that the applicant can proceed with the design and fabrication of the core. This matter of whether the worths of individual control rods are excessive will receive further review prior to issuance of the operating license

5. Zirconium-Water Reactions

In response to our request for additional information, the applicant discussed the magnitude of the energy and pressure contribution of any zirconium-water reaction to the drywell pressure as function of time for the 100% core melt case and its effect on the drywell leakage rate.

Three analytical cases were chosen:

1. Core spray initiated normally.
2. Core spray delayed 15 minutes.
3. Perfectly insulated reactor core with 100 percent meltdown.

In our opinion the model used in these three cases for the calculation of the percent zirconium water reaction yielded results consistent with data in the current literature. The model, also defines the relationship between volume of hydrogen evolved, time, and absolute temperature, which agrees well with available data.

Case 1

There was no reaction because the peak clad temperature did not exceed 1600°F.

Case 2

This assumed one core spray pump inoperative and the other had a 15 minute delay before actuation. A total of 0.46% of the clad was calculated to react. The increase in pressure is less than 0.5 psi due to hydrogen formation.

Case 3

In this case only enough steam to supply the zirconium-water reaction was assumed present. Calculations were performed to obtain fuel temperatures as a function of time after the accident. The reaction was considered to have a threshold clad temperature of 1800°F. A total of 23% of the zirconium in the core was calculated to react.

The maximum increase in pressure due to the hydrogen is 22 psi at 3.5 hours, which when added to the transient pressure at that time falls below the design pressure of the suppression chamber - 35 psig. After condensation of water vapor in the drywell, the resultant equilibrium pressure due to air and hydrogen is 10 psig. The maximum percent of hydrogen free volume during the accident was stated to be 54%. If all of the oxygen in the free volume were combined with part of the hydrogen evolved (either burn or explode) the pressure generated would exceed 100 psig. Containment integrity could not be maintained under these conditions. Since the containment integrity cannot be guaranteed for Case 3, we believe that some provision should be included in the design to preclude a hydrogen oxygen reaction.

What?

Can't inert the ~~[illegible]~~ gas pool to ventilated ∴ O_2 available

V. Conclusion

Based on the foregoing considerations, and assuming that the reservations outlined in the summary portion of this report are adequately answered, we believe that the proposed facility can be constructed and operated at the Nine Mile Point site without undue risk to the health and safety of the public.