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Reg. Docket File 50-220
LFMB Reactor File
LFMB R/F (2)
P. Polk, ORB-3
S. Sheppard, ORB-3
R. Diggs, LFMB
E. Coupe, LFMB

DOCKET NO. 50-220

OCT 30 1978

Niagara Mohawk Power Corporation
ATTN: Mr. Donald P. Dise
Vice President - Engineering
300 Erie Boulevard West
Syracuse, New York 13202

Gentlemen:

My letter to you dated September 18, 1978, advised that your September 1, 1978 application for amendment to Facility License No. DPR-63 requires a Class IV fee of $12,300 pursuant to 10 CFR 170.22. The application relates to the installation of a Radwaste Reduction System and associated waste handling equipment at Unit No. 1 of your Nine Mile Point Station. To date, we have not received the requested fee. The fee should be forwarded to this office within fifteen (15) days after your receipt of this letter.

Sincerely,

Original Signed by
Wm. O. Miller

William O. Miller, Chief
License Fee Management Branch
Office of Administration

DUPE

CERTIFIED MAIL
RETURN RECEIPT REQUESTED

7811150154

OFFICE>	LFMB:ADM	LFMB:ADM	LFMB:ADM			
SURNAME>	RWDiggs:slc	CJHolloway	WOMiller			
DATE>	10/27/78	10/27/78	10/27/78			

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ACRS (16)

L. BARRETT

OCTOBER 18 1978

Docket No. 50-220

Mr. Donald P. Dise
Vice President - Engineering
Niagara Mohawk Power Corporation
300 Erie Boulevard West
Syracuse, New York 13202

Dear Mr. Dise:

Your submittal of March 22, 1978, relating to expansion of the spent fuel and storage capacity at Nine Mile Point Nuclear Station, Unit No. 1, is being reviewed by our staff. In order to complete our review, you are requested to provide within 60 days of receipt of this letter, the additional information identified in the enclosure.

Sincerely,

D Vassallo for

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosure:
Request for Additional Information

cc w/enclosure:
See page 2

OFFICE	ORB #3	ORB #3				
SURNAME	PPolk:mjf	TIppolito	L. BARRETT			
DATE	10/16/78	10/18/78				

NRC FORM 318 (9-76) NRCM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

OCTOBER 1 1918

cc: Eugene B. Thomas, Jr., Esquire
LeBoeuf, Lamb, Leiby & MacRae
1757 N Street, N. W.
Washington, D. C. 20036

Anthony Z. Roisman
Natural Resources Defense Council
917 15th Street, N. W.
Washington, D. C. 20005

Oswego County Office Building
46 E. Bridge Street
Oswego, New York 13126

ENCLOSURE 1

REQUEST FOR ADDITIONAL INFORMATION

1. Provide a description of the modifications that have been made to the Nine Mile Point Unit 1 (NMP-1) spent fuel pool (SFP) after the pool was licensed to contain up to 1984 fuel assemblies. Discuss the effects of these modifications on operating the SFP. This should include the measured man-rem exposure to do the work, the change in radiation levels in the vicinity of the pool, the change in the amount of crud in the pool, and changes to the operation of the SFP purification system.

2. You stated in your letter to NRC dated September 29, 1977, in your response to Question 5, that the dose rates around the spent fuel pool are kept between 5 and 10 mrem/hour by controlling the frequency of changing the pool filter. Justify why the radiation fields will be as low as reasonably achievable (ALARA) during each phase of the proposed pool modification. Relevant experience at nuclear power plants show typical values, in the vicinity of spent fuel pools, of 1 to 2 mrem/hour. Your response should consider increased purification system operation or increased filter change out frequency.

3. You stated in your March 22, 1978 submittal that the additional spent fuel storage capacity you are requesting will be installed in steps as your storage needs dictate. Compare the man-rem exposure for the proposed stepwise pool modification with the exposure for

completing the modification in a single step. Show that your proposed course of action is consistent with the ALARA philosophy of 10 CFR Part 20.1(c).

4. Discuss the occupational exposure expected during each separate matrix listed in Table 1 of your March 22, 1978, submittal of this proposed SFP modification. Address the expected dose rates (from spent fuel pool water, spent fuel and the equipment to be disposed of), numbers of workers (including divers, if necessary) and occupancy times for each phase of the operation. Include removal and disposal of the present spent fuel racks and installation of the new higher density racks. Provide the estimated man-rem exposure. Compare the measured and estimated man-rem exposure for your 1978 SFP modification.

5. Provide the additional occupational exposure (in man-rem) from normal operation in the spent fuel pool area, including refueling, over the time period for the complete SFP modification proposed in your March 22, 1978, submittal. Include the expected exposure f · more frequent changing of SFP filters and demineralizers, from spent fuel pool water and from spent fuel.

6. Describe the method that will be used to dispose of the present racks (i.e., crating intact racks or cutting and packaging). If the racks are to be cut and packaged, show that the exposure received by this disposal method, as compared to crating the intact racks for disposal, will provide as low as is reasonably achievable (ALARA) exposure to personnel.

Your response should include a description of the method that was used to dispose of the racks and resultant exposures from the 1978 SFP modification.

7. Discuss the leakage of water from the spent fuel pool and the pool leak collection system. Where would the pool leakage be transferred to?

8. Discuss the effect of the fuel assembly movements during your 1978 SFP modification on the amount of crud in the pool water and the radiation levels in the vicinity of the pool. Your response should include measured radioactivity concentrations of SFP water and general radiation levels in the vicinity of the pool before, during and after the 1978 SFP modification.

9. Your March 22, 1978 submittal did not completely address the impact of the proposed SFP modification on the environment. Discuss in some detail the impact of the proposed SFP modification on the following:

   a) radioactive liquid effluents from the plant, including leakage of water from the pool and

   b) radioactive solid wastes from the plant, including the change in the frequency of replacing the SFP filter demineralizer resin.

10. Provide the estimated volume of contaminated material (e.g., spent fuel racks, seismic restraints) expected to be removed from the spent fuel pools during each step of the entire modification and shipped from the plant to a licensed burial site.

11. Provide a list of typical loads that might be carried near or over the spent fuel pool. Provide the weight and dimensions of each load. Discuss the load transfer path, including whether the load must be carried over the pool, the maximum height at which it could be carried and the expected height during transfer. Provide a description of any written procedures instructing crane operators about loads to be carried near the pool. Provide the number of spent fuel assemblies that could be damaged by dropping and/or tipping each typical load into the pool.

12. Discuss the instrumentation to indicate the spent fuel pool water level and water temperature. Include the capability of the instrumentation to alarm and the location of the alarms.

13. Propose a technical specification which prohibits carrying loads greater than the weight of a fuel assembly over spent fuel in the storage pool; or justify why this specification is not needed to limit the potential consequences of accidents involving dropping heavy loads, other than casks, onto spent fuel to those of the design basis fuel handling accident.

~~SEP 8 1978~~

SEP 18 1978

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E. Coupe, LFMB
LFMB R/F (2)

Docket No. 50-220

Niagara Mohawk Power Corporation
ATTN: Mr. Donald P. Dise
Vice President-Engineering
300 Erie Boulevard West
Syracuse, New York 13202

Gentlemen:

This office has received a copy of your September 1, 1978 application for amendment to Facility License No. DPR-63 which you filed with the Office of Nuclear Reactor Regulation. The application relates to the installation of a Radwaste Reduction System and associated waste handling equipment at Unit 1 of Nine Mile Point Nuclear Station. The application was not accompanied by an amendment fee as prescribed by Section 170.22 of 10 CFR Part 170 (copy enclosed). Section 170.12(c) requires that your company provide a proposed determination of the amendment class, state the basis therefor, and submit the fee with your application for amendment.

We have determined that the requested action falls in Class IV which requires an amendment fee of $12,300 because an extensive environmental impact appraisal is required. You should forward the Class IV fee of $12,300 promptly to this office. Fees are payable to the U.S. Nuclear Regulatory Commission by check, draft, or money order. If after the final evaluation of your application is completed it is determined that it was incorrectly classified, you will be refunded any overpayment or billed for any additional amount due.

If we can be of assistance to you, call 301/492-7225.

Sincerely,

Original Signed by
Wm. O. Miller

William O. Miller, Chief
License Fee Management Branch
Office of Administration

not received

Enclosure:
10 CFR 170 (Rev. 2/21/78)

App3
CCP

OFFICE	LFMB:ADM	LFMB:ADM	LFMB:ADM	LFMB:ADM		
SURNAME	ESCoupe:s	RMDiggs	CJHolloway	WOMiller		
DATE	9/15/78	9/15/78	9/15/78	9/18/78		

SEP 18 1978

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DEisenhut
TBAbernathy
JRBuchanan
ACRs (16)

Docket No. 50-220 AUGUST 2 9 1978

Niagara Mohawk Power Corporation
ATTN: Mr. Donald P. Dise
Vice President - Engineering
300 Erie Boulevard West
Syracuse, New York 13202

Gentlemen:

A number of instances of control rods failing to fully insert on scram have occurred over the past few years. The typical reported event involves a number of rods stopping short of the fully inserted position and then settling back to notch position "02" or six inches short of full insertion.

It is the staff's understanding that while failure to fully insert to the "00" position has occurred at a number of plants, not all of these events have been reported, perhaps because the event was not considered to have sufficient safety significance. Therefore, it is difficult for the staff to determine if instances of control rods failing to insert are decreasing or are indicative of a generic problem with potential safety implications. To enable a better understanding of the number of facilities affected and any trend in the number of rods failing to insert on each scram, we are requesting all licensees with operating BWRs to review their records for calendar year 1977. We would appreciate receiving a letter within 30 days that states that either no such events occurred at your facility(s) or that within 90 days of this letter you will provide a summary tabulation of the events. For each such event, you should identify the number of rods not fully inserted, the position of the rods, the cause of failure to fully insert and any related maintenance activities. Reference to reports that already describe such events is acceptable.

We recommend that you maintain an ongoing tabulation of any additional such events that may occur during the calendar year 1978. Should our evaluation of the 1977 data warrant, we may make a similar request for data covering calendar year 1978.

App 3

OFFICE➤						
SURNAME➤						
DATE➤						

AUGUST 2 9 1978

This request for generic information was approved by GAO under a blanket clearance number B-180225(R0536); this clearance expires June 30, 1981.

Sincerely,

Original signed by

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

cc: See next page

OFFICE➤	ORB #3	ORB #3	ORB #3			
SURNAME➤	SSheppard	PPolk:mjf	TIppolito			
DATE➤	8/21/78	8/ /78	8/29/78			

cc: Eugene B. Thomas, Jr., Esquire
LeBoeuf, Lamb, Leiby & MacRae
1757 N Street, N. W.
Washington, D. C. 20036

Anthony Z. Roisman
Natural Resources Defense Council
917 15th Street, N. W.
Washington, D. C. 20005

Oswego County Office Building
46 E. Bridge Street
Oswego, New York 13126

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ACRS (16)
File

Docket No. 50-220

Niagara Mohawk Power Corporation
ATTN: Mr. Donald P. Dise
Vice President - Engineering
300 Erie Boulevard West
Syracuse, New York 13202

Dear Mr. Dise:

Reference is made to your submittal of March 22, 1978 for an additional increase in the storage capacity of the spent fuel pool for Nine Mile Point Unit No. 1. As a result of our review, we have determined that we need the additional information identified in the enclosure with respect to the poison material you propose to use in the storage racks.

Sincerely,

Richard J. Clark for

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosure:
Request for Additional
Information

cc: see next page

App3

ccp

OFFICE➤	ORB#3	ORB#3				
SURNAME➤	RClark:acr	TIppolito				
DATE➤	8/18/78	8/18/78				

AUGUST 21 1978

cc: Eugene B. Thomas, Jr., Esquire
LeBoeuf, Lamb, Leiby & MacRae
1757 N Street, N. W.
Washington, D. C. 20036

Anthony Z. Roisman
Natural Resources Defense Council
917 15th Street, N. W.
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Oswego County Office Building
46 E. Bridge Street
Oswego, New York 13126

REQUEST FOR ADDITIONAL INFORMATION
SPENT FUEL POOL CAPACITY EXPANSION
NINE MILE POINT, UNIT NO. 1
DOCKET NO. 50-220

1. Provide the details of the uranium-235 loading and distribution in the fuel assembly which were used for the spent fuel pool criticality calculations.

2. Provide the areal density of boron-10 in the Boraflex plates which was used in the criticality calculations and provide the "quality assured" minimum value of this areal density (i.e., atoms of boron-10 per square centimeter of Boraflex plate).

3. Provide the calculated change in the storage lattice k_∞ with a small change in the boron-10 concentration in the Boraflex plates.

4. Will there be any place in the pool under any of the proposed changes where it will be possible to place a fuel assembly very close to the outside of a rack which is filled with fuel assemblies? Such a place might be the space between the outer periphery of the rack modules and the walls of the pool. What is the maximum neutron multiplication factor that can be obtained in this situation?

5. Provide the nominal value of the k_∞ for the PDQ-7 cell model shown in Figure 7 of your submittal.

6. Provide the maximum value of the k_∞ that is obtained when the perturbations listed in Table 3 of your submittal are all assumed to be in the direction that gives an increased neutron multiplication factor.

7. Provide the change in the multiplication factor when axial neutron leakage is assumed.

8. NRC procedures require that an onsite test be performed to verify within 95 percent confidence limits, that a sufficient number of neutron absorbing plates (i.e., poison plates) in the installed racks contain the required boron content to maintain the $k_{eff} \leq 0.95$. When the poison plates are made an integral part of the racks and the condition of the poison plates is continually monitored by surveillance tests, the NRC finds that a single, initial neutron attenuation test on the racks is sufficient. However, a single, initial neutron attenuation test will not be sufficient for the proposed racks with the removable poison inserts. Describe how you propose to periodically perform tests to verify, within 95 percent confidence limits, that there will always be a sufficient number of poison plates which contain the required boron content to maintain the $k_{eff} \leq 0.95$ in the proposed storage racks with the removable poison inserts.

9. Will any sources of neutrons other than spent fuel assemblies be stored in the spent fuel pool? If so at what rate will they emit neutrons?

10. What will the maximum integrated neutron and gamma flux be in the Boraflex material over the lifetime of the racks? What spent fuel assembly power density and burnup, and what rack life were assumed in calculating these maximum integrated fluxes? What is the assumed energy spectrum for the gamma flux?

11. What will the maximum temperature be in the center of the Boraflex material, assuming the highest neutron and gamma flux and the worst accident conditions?

12. State the quantity and composition of the gas which will come out of the Boraflex material when it is being irradiated in the spent fuel pool.

13. On page 9 of your March 22, 1978 submittal it is indicated that there is a poison slab which is sealed in a stainless steel casing. Is the Boraflex going to be sealed inside of a stainless steel casing?

14. What will the chemical composition of the Boraflex be after receiving the design dose of irradiation?

15. What is the melting temperature of the Boraflex in the unirradiated condition?

16. Is the Boraflex going to be bonded to stainless steel? If so what will happen to this bond under the design dose of irradiation in conjunction with the design number of thermal cycles?

17. What will the physical properties such as the density, the modulus of rupture, and the compressive strength of the Boraflex be after it receives the design dose of irradiation in the spent fuel pool?

18. Provide a detailed description of and the documented results of a prototypical experiment, which includes all significant aspects of the spent fuel pool situation and environment, that shows that these Boraflex plates will not become so brittle from irradiation in the spent fuel pool that they could be broken up by the insertion and removal of fuel assemblies or by a safe shutdown earthquake at some time in the design life of the spent fuel racks.

19. If the Boraflex is to be exposed to the pool water, state the maximum percentage of boron oxide, B_2O_3, in the B_4C. Since B_2O_3 is soluble in water it will either be necessary to assume that this amount of boron is leached from the Boraflex plates or to experimentally show that this will not happen during the life of the racks.

20. Describe the instrumentation and alarms on the spent fuel pool water level and temperature or reference the location in the FSAR where this description can be found.

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SSheppard
Attorney, OELD
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DEisenhut
TBAbernathy
JRBuchanan
ACRS (16)

Docket No. [redacted] AUGUST 7 1978

50-220

Niagara Mohawk Power Corporation
ATTN: Mr. Donald P. Dise
Vice President - Engineering
300 Erie Boulevard West
Syracuse, New York 13202

Gentlemen:

During the review of the E. I. Hatch Unit 2 nuclear power plant (Docket No. 50-366), the NRC Staff identified certain specific deficiencies in the design of the voltage regulator system of the motor generator sets which supply power to the reactor protection system, as follows:

(1) there were potential undetectable single component failures which could adversely affect the operability of the reactor protection system; and

(2) there is a postulated sequence of component malfunctions initiated by an earthquake which could adversely affect the operability of the reactor protection system.

Both of these deficiencies are described in greater detail in attachment (2) to this letter, which is an extract from the Hatch 2 Safety Evaluation Report (NUREG-0411).

We determined in the course of the Hatch 2 review that the safety problems associated with the postulated single failure could be remedied by additional surveillance; specifically, by assuring that the output voltage of each reactor protection system motor-generator is checked to be within ± 10% of the nominal value, approximately every eight hours. Requirements for such surveillance were imposed as part of the Hatch 2 Technical Specifications (Attachment 3).

Accordingly, provided the surveillance set forth in Attachment 1 is carried out, there is reasonable assurance that a facility using a system such as that used at Hatch 2 can be operated without endangering public health and safety.

For Hatch 2, the licensee requested and was granted an exemption from seismic design requirements for the period necessary to obtain and install qualified components.

OFFICE➤						Appl B
SURNAME➤						CCP
DATE➤						

Our records show that your reactor protection system power supply system is of the same general design as that at Hatch 2. However, it is not clear from such information whether the components actually installed at your facility have the same qualification as those used at Hatch 2. Moreover, it is not completely clear that system interaction for your systems will have the same adverse characteristics as that identified at Hatch 2.

For these reasons, you are hereby requested, pursuant to 10 CFR § 50.54(f), to evaluate your reactor protection system power supply in light of the information set forth in Attachment 2 to determine: whether there is potential for undetected single failures to adversely affect the reactor protection system, and whether there is a potential for the postulated sequence of events initiated by an earthquake which could adversely affect the reactor protection system. Your report should be filed within 60 days of the date of this letter. If you identify any necessary or desirable facility modifications or Technical Specification changes, proposals to implement such modifications or changes should accompany your report.

In the interim, promptly upon receipt of this letter, you should commence surveillance of the reactor protection system power supply as set forth in Attachment 1 hereto. Such surveillance should be continued until otherwise directed or authorized by NRC.

Sincerely,

Original signed by
Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosures:
1. Surveillance Program
2. Extract from Safety Evaluation Report
3. Extract from Hatch 2 Technical Specifications

cc w/enclosures:
See next page

OFFICE➤	ORB #3	ORB #3				
SURNAME➤	RClark:mjf	TIppolito				
DATE➤	8/ /78	8/ /78				

cc: Eugene B. Thomas, Jr., Esquire
LeBoeuf, Lamb, Leiby & MacRae
1757 N Street, N. W.
Washington, D. C. 20036

Anthony Z. Roisman
Natural Resources Defense Council
917 15th Street, N. W.
Washington, D. C. 20005

Oswego County Office Building
46 E. Bridge Street
Oswego, New York 13126

Attachment 1

Surveillance Program

(1) The output voltage and current of each reactor protection system motor-generator set shall be logged once per shift;

(2) A motor-generator set shall be removed from service if the output voltage is not within ±10% of its nominal valve and cannot be adjusted to fall within this band;

(3) The protective over-voltage and under-voltage relays and the under-frequence relay shall be calibrated initially at least once every six months, and after an operating basis earthquake. The tripping logic and the generator output breaker shall be functionally tested as a part of the calibration of these relays. The voltage setpoints shall be within the range specified in Requirement (2) above and the frequency setpoint shall be greater than or equal to 57 Hertz; and

(4) A protection system functional test shall be conducted upon discovery of a condition beyond the limits of Requirement (2) above. This test shall include all Class IE loads which are connected to the buses.

Attachment 2

Extract From Safety Evaluation Report:

Related to Operation of E. I. Hatch Nuclear Plant,

Unit 2, Docket 50-366, June 1978

The design of the Hatch Unit 2 reactor protection system power supply is essentially the same as that of previously-licensed BWR reactors. The protection system power supply consists of two high-inertia alternating current motor-generator sets.

During our review of the Hatch Unit 2 operating license application, we questioned the adequacy of protection afforded Class IE reactor protection system against possible sustained over-voltage or under-voltage conditions from the non-Class IE reactor protection system power supply. Specifically, we questioned the capability of the reactor protection system power supply to accommodate (1) postulated single failures and (2) the effects of earthquakes without jeopardizing the capability of the reactor protection system to perform its intended safety function.

Criterion 21 of the General Design Criteria requires in part that the redundancy and independence designed into the reactor protection system be sufficient to assure that no single failure results in loss of the protection function. In applying the single failure criterion to a specific design, we assume that all potential undetectable failures are in their failed mode (Appendix 7A of the Standard Review Plan) before the occurrance of the postulated detectable single failure which (in a system meeting the single criterion) will not disable the protection function. For the Hatch Unit 2 reactor protection system power supply, a single undetected failure of an output voltage sensor for either motor-generator set could be postulated that would allow the generator output voltage to remain outside the voltage rating (range) of the connected Class IE loads. Such an abnormal voltage, resulting from a possible failure in the motor-generator set voltage regulating circuitry, if persisting for a sufficient time, could result in damage to the reactor protection system components with the attendant potential loss of capability to scram the plant.

IEEE Standard 379-1977, "IEEE Standard Application of the Single-Failure Criterion to Nuclear Power Generating Station IE Systems," provides that an otherwise undetectable failure may be deemed detectable by means of appropriate surveillance and/or testing. To ensure that failure of the non-Class IE reactor protection system power supply will not cause adverse interaction to the Class IE reactor protection system, the following requirements will be included in the Technical Specifications to ensure the timely detection of failures due to sustained over-voltage or under-voltage conditions:

(1) The output voltage and current of each reactor protection system motor-generator set shall be logged once per shift;

(2) A motor-generator set shall be removed from service if the output voltage exceeds 132 volts AC or is less than 108 volts and cannot be adjusted to fall within this band;

(3) The protective over-voltage and under-voltage relays and the under-frequency relay shall be calibrated before initial plant startup, at least once every six months, and after an operating basis earthquake. The tripping logic and the generator output breaker shall be functionally tested as a part of the calibration of these relays. The voltage setpoints shall be within the range specified in Requirement (2) above and the frequency setpoint shall be greater than or equal to 57 Hertz; and

(4) A protection system functional test shall be conducted upon discovery of a condition beyond the limits of Requirement (2) above. This test shall include all Class IE loads which are connected to the buses.

We conclude that these Technical Specification requirements will ensure the timely detection of failures due to sustained over-voltage or under-voltage conditions. We also conclude that with these Technical Specification requirements, the reactor protection system power supply conforms to the provisions of IEEE Standard 379-1977 and, therefore, satisfies the applicable requirements of Criterion 21 of the General Design Criteria.

Criterion 2 of the General Design Criteria requires in part that systems important to safety, such as the reactor protection system, be designed to withstand the effects of earthquakes. The Hatch Unit 2 reactor protection system is a Class IE system, hence it is seismic Category I. The reactor protection system power supply, however, is not seismically qualified. We have determined that a sequence of events initiated by an earthquake can be postulated which could result in damage to the reactor protection system components with the attendant potential loss of capability to scram the plant. This sequence of events includes (a) the occurrence of an earthquake that would cause the undetected failure of a voltage sensor, (b) the failure of the motor-generator set resulting in abnormal output voltage, (c) persistence of the abnormal output voltage undetected by visual observation and surveillance testing for a time sufficient to damage reactor protection system components, and (d) failure of these components in such a manner that results in loss of scram capability (instead of in the fail-safe mode).

Therefore, we require that, prior to startup following the first scheduled refueling outage, the applicant install a Class IE system approved by us capable of de-energizing the reactor protection system power supply when its output voltage exceeds or falls below limits within which the equipment being powered from the power supply has been designed and qualified to operate continuously and without degradation. With such a system, the reactor protection system power supply design will be in conformance with the applicable requirements of Criterion 2 of Appendix A to 10 CFR Part 50. The operating license will be conditioned accordingly.

Extract from Hatch 2 Technical Specifications

ELECTRICAL POWER SYSTEMS

3/4.8.2 ONSITE POWER DISTRIBUTION SYSTEMS

A.C. DISTRIBUTION - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 The following A.C. distribution system buses, inverters and motor-generator (MG) sets shall be OPERABLE with breakers open between redundant buses:

a. 4160 volt Essential Buses 2E, 2F and 2G,

b. 600 volt Essential Buses 2C and 2D,

c. 120/208 volt Essential Cabinets 2A and 2B,

d. 120/208 volt Instrument Buses 2A and 2B,

e. A.C. inverters 2R44-S002 and 2R44-S003, and

f. If in service, Reactor Protection System instrumentation MG sets 2A and 2B.

APPLICABILITY: CONDITIONS 1, 2 and 3.

ACTION:

a. With one of the above required A.C. distribution system buses or inverters inoperable, restore the inoperable bus or inverter to OPERABLE status within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With two or more of the above required A.C. distribution system buses or inverters inoperable, restore at least all except one of the inoperable buses and inverters to OPERABLE status within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With RPS instrumentation MG set 2A and/or 2B voltage outside the range of 108 to 132 VAC, demonstrate the OPERABILITY of all equipment which could have been subjected to the abnormal voltage for all Class IE loads connected to the associated bus(es) by performance of a CHANNEL FUNCTIONAL TEST or CHANNEL CALIBRATION, as required, within 24 hours.

d. With RPS instrumentation MG set 2A and/or 2B inoperable, restore the inoperable MG set(s) to OPERABLE status within 30 minutes or remove the inoperable MG set(s) from service.

SURVEILLANCE REQUIREMENTS

4.8.2.1.1 The above required A.C. distribution system buses and inverters shall be determined OPERABLE:

a. At least once per 7 days by verifying correct breaker alignment and indicated power availability, and

b. At least once per 31 days by determining that the 250 volt DC/600 volt AC inverters 2R44-S002 and 2R44-S003 are OPERABLE by verifying inverter output voltage of 600 volts ± 5% while supplying their respective buses.

4.8.2.1.2 The above specified RPS instrumentation MG sets 2A and 2B shall be determined OPERABLE:

a. At least once per 8 hours by verifying;

1. RPS instrumentation MG sets 2A and 2B voltage to be between 108 and 132 VAC, and

2. No unexplained change in RPS instrumentation MG set 2A and/or 2B current in excess of 5% from the value observed during the Startup Test Program.

b. At least once per 6 months and prior to resetting the Reactor Protection System trips following a seismic event of Operational Basis Earthquake intensity, by demonstrating the OPERABILITY of RPS instrumentation MG set 2A and 2B over-voltage, under-voltage and under-frequency protective instrumentation by performance of a CHANNEL CALIBRATION including simulated automatic actuation of the protective relays, tripping logic and output circuit breakers and verifying the following setpoints;

1. Over-voltage ≤ 132 VAC,

2. Under- voltage ≥ 96 VAC, and

3. Under- frequency ≥ 57 Hz.

NOTE: 4.8.2.1.2 b.2 is in error; it should read "Undervoltage ≥108 VAC, and"

A.C. DISTRIBUTION - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.8.2.2 As a minimum, the following A.C. distribution system buses, inverters and motor-generator (MG) sets shall be OPERABLE:

a. Two 4160 volt Essential Buses, 2E, 2F and/or 2G,

b. One 600 volt Essential Bus, 2C or 2D,

c. One 120/208 volt Essential Cabinet, 2A or 2B,

d. One 120/208 volt Instrument Bus, 2A or 2B,

e. A.C. inverters 2R44-S002 and 2R44-S003*, and

f. If in service, Reactor Protection System instrumentation MG sets 2A and 2B.

APPLICABILITY: CONDITIONS 4 and 5.

ACTION:

a. With less than the above required A.C. distribution system buses and inverters OPERABLE, suspend all operations involving CORE ALTERATIONS, irradiated fuel handling, positive reactivity changes or operations that have the potential of draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

b. With RPS instrumentation MG set 2A and/or 2B voltage outside the range of 108 to 132 VAC, demonstrate the OPERABILITY of all equipment which could have been subjected to the abnormal voltage for all Class IE loads connected to the associated bus(es) by performance of a CHANNEL FUNCTIONAL TEST or CHANNEL CALIBRATION, as required, within 24 hours.

c. With RPS instrumentation MG set 2A and/or 2B inoperable, restore the inoperable MG set(s) to OPERABLE status within 30 minutes or remove the inoperable MG set(s) from service.

SURVEILLANCE REQUIREMENTS

4.8.2.2 At least the above required A.C. distribution system buses, inverters and MG sets shall be determined OPERABLE per Specifications 4.8.2.1.1 and 4.8.2.1.2.

Docket
50-220

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

July 28, 1978

ALL BWR POWER REACTOR LICENSEES

Gentlemen:

We have discovered that pages 3/4 3-51 and B 3/4 11-5 were inadvertently omitted from the Draft Radiological Effluent Technical Specifications for BWR's (BWR-STS-I) which were forwarded to you by our letter dated July 11, 1978. Attached are copies of the omitted pages for your use.

If you have any questions on this matter, please contact us.

Sincerely,

Brian Grimes

Brian Grimes, Assistant Director
for Engineering and Projects
Division of Operating Reactors

Enclosures:
1. Page 3/4 3-51
2. Page B 3/4 11-5

Appy3

cop

TABLE 4.3-11 (Continued)

TABLE NOTATION

* - During releases via this pathway.

** - During liquid additions to the tank.

(1) - The CHANNEL FUNCTIONAL TEST shall also demonstrate that automatic isolation of this pathway and control room alarm annunciation occurs if any of the following conditions exist:

1. Instrument indicates measured levels above the alarm/trip setpoint.

2. Circuit failure.

3. Instrument indicates a downscale failure.

4. Instrument controls not set in operate mode.

(2) - The CHANNEL FUNCTIONAL TEST shall also demonstrate that control room alarm annunciation occurs if any of the following conditions exist:

1. Instrument indicates measured levels above the alarm/trip setpoint.

2. Circuit failure.

3. Instrument indicates a downscale failure.

4. Instrument controls not set in operate mode.

(3) - The CHANNEL CALIBRATION shall include the use of a known (traceable to the National Bureau of Standards radiation measurement system) liquid radioactive source positioned in a reproducible geometry with respect to the sensor and emitting beta and gamma radiation with the fluences and energies in the ranges measured by the channel during normal operation.

3/4.11.2.5 EXPLOSIVE GAS MIXTURE

This specification is provided to ensure that the concentration of potentially explosive gas mixtures contained in the waste gas holdup system is maintained below the flammability limits of hydrogen and oxygen. (Automatic control features are included in the system to prevent the hydrogen and oxygen concentrations from reaching these flammability limits. These automatic control features include isolation of the source of hydrogen and/or oxygen, automatic diversion to recombiners, or injection of dilutants to reduce the concentration below the flammability limits.) Maintaining the concentration of hydrogen and oxygen below their flammability limits provides assurance that the releases of radioactive materials will be controlled in conformance with the requirements of General Design Criterion 60 of Appendix A to 10 CFR Part 50.

3/4.11.2.6 MAIN CONDENSER

Restricting the gross radioactivity rate of noble gases from the main condenser provides reasonable assurance that the total body exposure to an individual at the exclusion area boundary will not exceed a small fraction of the limits of 10 CFR Part 100 in the event this effluent is inadvertently discharged directly to the environment without treatment. This specification implements the requirements of General Design Criteria 60 and 64 of Appendix A to 10 CFR Part 50.

3/4.11.2.7 MARK I CONTAINMENT (OPTIONAL)

This specification provides reasonable assurance that releases from drywell purging operations will not exceed the annual dose limits of 10 CFR Part 20 for unrestricted areas.

3/4.11.3 SOLID RADIOACTIVE WASTE

The OPERABILITY of the solid radwaste system ensures that the system will be available for use whenever solid radwastes require processing and packaging prior to being shipped offsite. This specification implements the requirements of 10 CFR Part 50.36a and General Design Criteria 60 of Appendix A to 10 CFR Part 50. The process parameters included in establishing the PROCESS CONTROL PROGRAM may include, but are not limited to waste type, waste pH, waste/liquid/solidification agent/catalyst ratios, waste oil content, waste principal chemical constituents, mixing and curing times.

Niagara Mohawk Power Corporation

cc: Eugene B. Thomas, Jr., Esquire
LeBoeuf, Lamb, Leiby & MacRae
1757 N Street, N. W.
Washington, D. C. 20036

Anthony Z. Roisman
Natural Resources Defense Council
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Docket No. 50-220

JULY 1 0 1978

Niagara Mohawk Power Corporation
ATTN: Mr. Donald P. Dise
Vice President - Engineering
300 Erie Boulevard West
Syracuse, New York 13202

Gentlemen:

We have completed our initial review of the Nine Mile Point, Unit 1 fire protection program. Additional information is required to complete our evaluation. The specific information we require is identified in Enclosure 1.

We have taken several positions to resolve certain issues. The staff's positions are identified in Enclosure 2.

The site visit by the fire protection review team is scheduled for the week of September 18, 1978. We request that you be prepared to discuss the information and positions at that time.

Sincerely,

Original signed by

Thomas A. Ippolito, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosure:
Request for Additional Information

cc w/enclosure:
See page 2

A A/3

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DATE	7/10/78	7/10/78				

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2

ENCLOSURE 1

REQUEST FOR ADDITIONAL INFORMATION

NINE MILE POINT UNIT 1

1. Fire Protection Program-Organizational Chart (A.1)*

Provide an organizational chart of the offsite and onsite personnel involved in the fire protection program.

2. Combined Fire and Security Emergency (A.1)

Describe the responsibilities of key plant personnel in the event of a combined fire and security emergency.

3. Equipment Required For Safe Shutdown (A-2)

The information provided in the Fire Protection Program report is not sufficiently detailed. The information does not indicate whether there are any fire areas in the plant containing equipment that performs a function required for safe shutdown that cannot be performed by equipment located in another fire area. For each fire area, provide a list of the equipment (including cable runs) located in the area that can be used to perform functions required for safe shutdown. For each item on the list, indicate whether its function can be performed by equipment located in another fire area and identify the area. In preparing the list for each fire area, the following functions should be considered to be required for safe shutdown:

1. Placing the reactor in a subcritical condition and maintaining the reactor subcritical idefinitely.

2. Bringing the reactor to hot shutdown conditions and maintaining it at hot shutdown for an extended period of time (i.e., longer than 72 hours) using only normal sources of cooling water.

3. Maintaining the reactor coolant system inventory indefinitely using only normal sources of makeup water.

4. Bringing the reactor to cold shutdown conditions within 72 hours.

If all the redundant equipment available to perform any of the above functions (assuming a loss of offsite electrical power) is located in a single fire area, identify the specific separation that exists and any combustible material between the redundant equipment.

4. Fire Induced Spurious Equipment Operation (A.2)

Identify any equipment required for safe shutdown (see item 3 above) that is subject to spurious operation as a result of a fire. Particular attention is directed to valves and valve position indicators. Discuss the effects on safe shutdown of such spurious operation.

*Referenced section in Appendix A to BTP 9.5-1

3

5. Instrument and Station Air System (A.2)

Describe the function of the instrument and station air system in achieving and maintaining both hot shutdown and cold shutdown conditions. Identify any fire areas which contain components or piping of the air system and air operated valves whose position must change for shutdown. Verify that the loss of the air system will not prevent shutdown operations.

6. Safe Shutdown Systems-Valves (A.2)

a. Provide a list of remotely-operated valves, with their fail positions, in safe shutdown systems identified in item 3 above.

b. Describe the provision and accessibility to manually operate these valves, if necessary, during the shutdown operations following fires which prevent remote operation of the valves.

7. Failure Analysis (A.4)

Provide a failure analysis which verifies that a single failure, other than a failure of the fire main discharge header, does not impair the primary and backup fire suppression capabilities. The analysis should include consideration of failures in the suppression system, the fire detection system or the power sources for such systems.

8. Lightning Effects (A.5)

Describe the means provided to prevent lightning from initiating fires which could damage safety-related equipment. Describe the means provided to prevent lightning from damaging the fire protection system.

9. Effects of Extinguishing Agents (A.5)

Provide the results of an analysis which shows that rupture or inadvertent operation of a fire fighting system will not subsequently cause damage or failure of safety-related equipment required for safe shutdown.

10. Safety-related Systems Interlocked with Fire Fighting Systems (A.5)

Identify any safety-related systems or their auxiliaries which are interlocked to and could be disabled by operation of a fire fighting system.

11. Fire Brigarde Equipment (B.3)

Describe the equipment provided for the fire brigade. Describe means that will be used to either override door locking mechanisms or breach a barrier to provide fire brigade access and personnel egress in the event of a locking mechanism failure. Describe the training and tools provided for this purpose.

4

12. Shared Emergency Equipment (B.3)

List the emergency equipment that is shared or proposed to be used by both the fire brigade and the security team.

13. Supplemental Fire Department (B.4)

Describe the procedures and required authorization for entry, command and supervision of off-site fire department.

14. Fire Brigade Organizational Chart (B.5)

Provide an organizational chart of the fire brigade for each shift which shows the brigade size, composition and chain of command and which designates the normal duty position of the brigade leader, i.e., operator, electrician, maintenance man, etc.

15. Fire Brigade Physical Examination (B.5)

Confirm that all fire brigade members are provided with a periodic physical examination to screen out personnel with heart or respiratory disorders, or provide justification for any exceptions.

16. Fire Drills (B.5)

Confirm that the assessment of fire drills include subsections 3.0a, 3.0b, 3.0d and 3.0f of Attachment No. 2 to "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance", or provide justification for any exceptions.

17. Fire Fighting Procedures (B.5)

Confirm that the following have been done or provide justification for any exceptions:

1. fire fighting procedures are documented and include strategies established for fighting fires in all safety-related and areas presenting a hazard to safety-related equipment; and

2. fire fighting strategies include the requirements of subsections d(1) through d(10) of Attachment No. 5 to "Nuclear Plant Fire Protection Functional Responsibilities, Administrative Controls and Quality Assurance."

18. Removal from Service Procedure (C.6)

Provide a summary of the procedures established to control the disarming of any automatic or manually actuated fire protection system. Identify the management position responsible for authorizing the disarmament and the means used to assure the system is returned to normal.

5

19. Drains (D.1)

a. Provide the results of an analysis which shows that drains have sufficient capacity, and/or equipment pedestals have sufficient height to prevent standing water from sprinklers and fire hoses from damaging safety-related equipment or supporting sysems necessary for safe shutdown of the plant. As an alternate, show that the standing water does not damage such equipment.

b. Identify the areas containing safe shutdown equipment that are not provided with floor drains. Describe the drainage path for those areas without drains.

c. Identify the areas containing combustible liquids that are not provided with floor drains. Describe the drainage path and provisions for containing or diverting the combustible liquid in those areas without drains. In those areas with drains, state the capacity and location of the drain reservoirs and describe the provisions to prevent the spread of flammable liquid fires via the drain system to safety-related areas or to other areas containing combustible liquids.

20. Curbed Areas (D.1)

Provide the results of an analysis that shows that curbed areas surrounding combustible liquid tanks have sufficient capacity to contain the full contents of the tanks plus the quantity of water required for extinguishment of a fire involving the combustible liquid.

21. Pipe and Ventilation Duct Penetrations (D.1)

Provide the results of an analysis which shows that the existing or proposed fire barrier penetration seals for pipe and ventilation duct penetrations are adequate to prevent the spread of smoke and fire through the barrier considering the combustible loading and possible air pressure differential.

22. Piping Containing Combustibles (D.2)

Identify all piping containing flammable gas or combustible liquid which is routed through areas containing safety-related equipment, safety-related cables or through which personnel must pass to reach safety-related equipment for local operation. Provide an analysis to show that a fire involving the liquid or gas will not prevent safe shutdown or result in the loss of function of a safety-related system. Describe the provisions, if any, to piping systems which would reduce the likelihood or magnitude of a flammable gas or combustible liquid fire.

23. Diesel Fuel Transfer Shut-off (D.2)

Describe the means provided to automatically and/or manually stop the transfer of diesel oil from the storage tanks to the diesel generator day tanks in the event of a fire in the area housing the day tank, or through which the fuel oil transfer piping is routed.

6

24. Separation Criteria (D.3)

Describe the separation criteria used for the routing of electrical cables. Certain cables electrically connected to equipment necessary for safe shutdown may be used for functions designated as non-safety-related and therefore classified as non-safety-related. Examples of these might be remote indicating lights for valves, breakers, etc. Describe whether such cables are kept with the safety division to which they were originally connected and if not, describe the effects on the safe shutdown equipment due to shorts to these cables as a result of fire.

25. Fire Stops (D.3)

Provide a detailed description of existing and proposed fire breaks and fire stops. Include sketches, identification of materials of construction, and description of test results which demonstrate the effectiveness of fire stops used on electrical cubicles and vertical cable trays; and for intersection between horizontal and vertical cable runs. Provide the criteria that were used in the design of the fire breaks and fire stops.

26. Cable Insulation Materials (D.3)

Identify all types of cable used in all areas of the cable tray system. For each type of cable, identify the materials used for insulation and jacketing. State the combustion and toxic characteristics of each type of material. Identify whether flame tests were performed on single and jacketed assemblies. Provide the acceptance criteria and results of the flame tests. Identify the flame temperature used, the exposed area, and the heat rate. Provide a comparison between these test procedures and the IEEE 383 flame test procedures.

27. Method of Heat and Smoke Venting (D.4)

In all the areas where manual fire fighting is proposed as either primary or backup means of suppression, describe the methods which would be used for heat and smoke removal using either fixed or portable air handling equipment. If the plant HVAC systems are proposed for such service, provide design data to show that these systems are rated for the conditions (temperature and capacity) required when used for this service.

28. Prevention of Fire and Smoke Spread (D.4)

Describe the manner in which fire and smoke are prevented from spreading from area to area via the normal and emergency ventilation systems in all parts of the plant areas. Describe the location, actuation method and fire rating of dampers used for fire and smoke control in both air supply and return air systems. Describe the details of interlocks for ventilation system shutdown or mode change that can be utilized for fire and smoke control.

1

29. Ventilation System Power and Control (D.4)

Identify all areas where ventilation systems power supply or controls are located within the area they serve. Provide the basis for leaving ventilation systems power and control cables within the area they serve.

30. Preventing Recirculation of Ventilation Air (D.4)

Describe the separation between the air intakes and exhausts for normal and emergency ventilation systems and the current and proposed provisions which prevent smoke from being drawn back into the plant.

31. Operation of Fire Dampers (D.4)

Discuss the provision for remotely re-opening fire dampers (including dampers actuated by carbon dioxide suppression system operation) for post-fire smoke venting.

32. Combustible Filters (D.4)

Identify the location of all combustible filters used at the plant and discuss the potential fire hazard involved at each location. Describe the fire detection and suppression capability and fire prevention measures for all such combustible filters. Provide the results of an analysis on the effects of combustion of the filters in terms of heat, smoke generation, radiation release, and damage to safety-related equipment.

33. Emergency Breathing Air (D.4)

Describe the total capacity of the presently available self-contained breathing units including reserve supply. Describe how this capability will be upgraded by the addition of the air compressor.

34. Proximity of Regular and Emergency Lighting Wiring (D.5)

Provide the results of an evaluation of the potential for a fire in a safety-related area to cause damage to electrical wiring which would result in the loss of both regular and emergency lighting to areas providing access to the fire or egress from the area. State the number of portable lights designated for emergency use at a central location.

35. Communication Systems (D.5)

Verify that emergency communications can be maintained for any fire area by using equipment located onsite that is not subject to damage from a fire in the area.

8

36. Fire Detection System Design (E.1)

Provide design data for the automatic fire detection system in each fire area, including such items as type, number and location of the detectors; and signaling, power supply and supervision of the system. Identify any deviation(s) from NEPA 72D.

37. Fire Suppression System Design (E.3)

Provide the design data for all automatic suppression systems (both existing and proposed) including such items as design densities, soak times, power supplies, and associated alarms. Identify areas of non-compliance with appropriate NFPA Standards.

38. Remote Shutdown Panels (F.6)

Provide an analysis to demonstrate that no fire which could impair control from the control room could also prevent control from remote shutdown panels. Identify any interconnections between the remote shutdown panel and the control room and discuss the means of isolating the two control stations in the event of a fire.

39. Radiological Consequences of a Fire (F.14)

Evaluate the radiological consequences of a fire in radwaste areas and areas containing contaiminated materials such as filter cartridge, spent resin, etc.

9

PF-1 Fire Door Supervision (D.1)*

Fire doors to safety-related areas or areas posing a fire hazard to safety-related areas should be normally closed and locked or electrically supervised with delayed alarm and annunciation in the control room.

PF-2 Electrical Cable Penetration Qualification (D.3)

The cable penetration fire barriers should be tested to demonstrate a three-hour rating, as is required for fire barriers. The test should be performed or witnessed by a representative of a qualified independent testing laboratory, and should include the following:

(1) The tests should be performed in accordance with ASTM E-119 and the following conditions.

(2) The cables used in the test should include the cable insulation materials used in the facility.

(3) The test sample should be representative of the worst case configuration of cable loading, cable tray arrangement, anchoring and penetration fire barrier size and design. The test sample should also be representative of the cable sizes in the facility. Testing of the penetration fire barrier in the floor configuration will qualify the fire stop for use in the wall configuration also.

(4) Cables penetrating the fire barrier should extend at least three feet on the unexposed side and at least one foot on the exposed side.

(5) The fire barrier should be tested in both directions unless the fire barrier is symmetrical.

(6) The fire barrier should be tested with a pressure differential across it that is equivalent to the maximum pressure differential a fire barrier in the plant is expected to experience.

(7) The temperature levels of the cable insulation, cable conductor, cable tray, conduit, and fire stop material should be recorded for the unexposed side of the fire barrier.

*Referenced section in Appendix A to BTP 9.5-1.

10

(8) Acceptance Criteria - The test is successful is:

(a) The cable penetration fire barrier has withstood the fire endurance test without passage of flame or ignition of cables on the unexposed side for a period of three hours, and

(b) The temperature levels recorded for the unexposed side are analyzed and demonstrate that the maximum temperature is sufficiently below the cable insulation ignition temperature, and

(c) The fire barrier remains intact and does not allow projection of water beyond the unexposed surface during the hose stream test.

If previous tests can be shown to meet the above position, the licensee should provide the results of the tests to show that the above position is met.

PF-3 Smoke Detection Systems Tests (E.1)

In situ tests should be conducted with a suitable smoke generation device to verify that the products of combustion from a fire would be promptly detected by installed smoke detectors and that ventilation air flow pattern in the area do not significantly reduce or prevent detection response. Bench tests should be conducted to verify that smoke detectors will provide prompt response and have adequate sensitivity to the products of combustion for the combustibles in the area where smoke detectors are installed. If any fire detection systems are found to be inadequate, appropriate modifications should be made to provide adequate detection system performance.

PF-4 Battery Room Ventilation Air Flow Monitor

If not presently provided, a ventilation air flow monitor should be installed in each of the station battery rooms to alarm and annunciate, in the control room, the loss of ventilation air flow.

MAY 26 1978

Docket No. 50-220

Niagara Mohawk Power Corporation
ATTN: Mr. Donald P. Dise
Vice President - Engineering
300 Erie Boulevard West
Syracuse, New York 13202

Gentlemen:

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RE: MANPOWER REQUIREMENTS FOR OPERATING REACTORS

We are enclosing a document entitled, "Manpower Requirements for Operating Reactors." We are using the bases given in this document for allowing the sharing of duties to meet minimum staffing requirements for fire brigades at nuclear power plants. This is being provided for your guidance in meeting NRC requirements in this area.

Sincerely,

Original Signed by
Victor L. Stello

Victor Stello, Jr., Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

Enclosure:
Manpower Requirements for
Operating Reactors

cc w/encl:
See next page

OFFICE	ORB#1	ORB#3	ORB#3	D/DOR		
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DATE	5/23/78	5/25/78	5/26/78	5/26/78		

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MANPOWER REQUIREMENTS FOR OPERATING REACTORS

The NRC has established requirements for personnel at operating reactors for purposes of plant operation, industrial security, and fire fighting. The following discussion considers the extent to which plant personnel assigned to either plant operation or security may also be temporarily allowed to man a fire brigade in the event of a fire for a single unit facility and sets forth an acceptable sharing scheme for operating reactors.

Summary of Manpower Requirements

1. Fire Brigade: The staff has concluded that the minimum size of the fire brigade shift should be five persons unless a specific site evaluation has been completed and some other number justified. The five-man team would consist of one leader and four fire fighters and would be expected to provide defense against the fire for an initial 30-minute period. See Attachment A for the basis for the need for a five-man fire brigade.

2. Plant Operation: Standard Review Plan Section 13.1.2 requires that for a station having one licensed unit, each shift crew should have at least three persons at all times, plus two additional persons when the unit is operating. For ease of reference, Attachment B contains a copy of this SRP.

3. Plant Security: The requirements for a guard force are outlined in 10 CFR Part 73.55. In the course of the staff's review of proposed security plans, a required minimum security response force will be established for each specific site. In addition to the response team, two additional members of the security force will be required to continuously man the Central Alarm Station (CAS) and Secondary Alarm Station (SAS). It is expected that many facilities will have a security organization with greater numbers of personnel than the minimum number assumed for purposes of discussion in this paper.

The NRC staff has given consideration to the appropriateness of permitting a limited degree of sharing to satisfy the requirements of plant operation, security and fire protection and has concluded that, (1) subject to certain site and plant specific conditions, the fire brigade staffing could generally be provided through operations and security personnel, and (2) the requirements for operators and the security force should remain uncompromised. Until a site specific review is completed, the following indicates the interim distribution and justification for these dual assignments, and therefore our interim minimum requirements for a typical presently operating commercial single unit facility. The staff believes that manpower for the fire brigade for multi-unit facilities is not now a problem because of the larger numbers of people generally present at the sites. Situations which do pose problems will be reviewed on a case-by-case basis.

1. Plant Operation: The staff has concluded that for most events at a single unit nuclear facility, a minimum of three operators should be available to place the reactor in a safe condition. The two additional operators required to be available at the nuclear facility are generally required to be present to perform routine jobs which can be interrupted to accomodate unusual situations that may arise. That is, there is the potential for the remaining two members of the operating crew to assume other short-term duties such as fire fighting. In light of the original rationale for providing extra plant operators to cope with off-normal conditions, it appears justified to rely on these personnel for this function. The staff recommends that one of the two operators assigned to the fire brigade should be designated as leader of the fire brigade in view of his background in plant operations and overall familiarity with the plant. In this regard, the shift supervisor should not be the fire brigade leader because his presence is necessary elsewhere if fires occur in certain critical areas of the plant.

2. Plant Security: In the event of a fire, a contingency plan and procedures will be used in deploying the security organization to assure that an appropriate level of physical protection is maintained during the event. The staff has determined that it is possible in the planning for site response to a fire, to assign a maximum of three members of the security organization to serve on the fire brigade and still provide an acceptable level of physical protection. While certain security posts must be manned continuously (e.g., CAS, SAS), the personnel in other assignments, including the response force, could be temporarily (i.e., 30 minutes) assigned to the fire brigade. In judging the merits of this allowance the underlying question is whether the minimum security force strength must be maintained continuously in the event of a plant emergency such as a fire. Further examination of this issue leads to two potential rationales for reaching an affirmative decision. First, could there be a causal connection between a fire and the security threat? Second, are there compelling policy reasons to postulate a simultaneous threat and fire?

   The first potential rationale would only be credible if, (1) the insider (posed as part of the threat definition) was an active participant in an assault and started a fire coincident with the attack on the plant or, (2) a diversionary fire was started by an attack force somewhere external to the plant itself where no equipment required for safe shutdown is located. The role of the insider will be discussed first. While 73.55 assigns an active status to the insider, the rule also requires that measures be implemented to contain his activities and thereby reduce his

effectiveness. At present, these measures include background checks on plant employees, limited access to vital plant areas, badging systems and the two-man rule. Here, limited access means that only designated employees are allowed in vital areas and that their entry is controlled by either conventional locks or card-key systems. Also, if separate trains of safety equipment are involved, then either compartmentalization or the two-man rule is required. These measures to contain the insider are presently being implemented and will provide assurance that people of questionable reliability would not be able to gain employee status at a nuclear plant and should they become an employee with unescorted access, significant restraints would be interposed on the ability of such a person to carry out extensive damage to plant vital areas. Recognizing that additional safeguards may still be appropriate, the staff has recommended to the Commission that plant personnel also be required to obtain an NRC security clearance. The staff believes that the attendant background investigation associated with a clearance, in conjunction with the other 73.55 measures, will provide a high degree of assurance that plant personnel will not attempt to take an active sabotage role. If the clearance rule is adopted the staff believes some of the measures, such as the two-man rule, designed to contain the insider can be relaxed. Thus, there does not now appear to be a reasonably credible causative relationship between a fire intentionally set by an insider and the postulated external security threat. For the case of diversionary fires set external to the plant itself, adequate security forces can still be maintained by allowing only part of the fire brigade to respond while both fire fighters and security force armed responders maintain a high degree of alertness for a possible real attack somewhere else on the plant. Thus, the effective number of armed responders required by 73.55 can be maintained for external diversionary fires.

The second potential rationale concerns whether a serious, spontaneous fire should be postulated coincident with an external security threat as a design basis. In evaluating such a requirement it is useful to consider the likelihood of occurrence of this combination of events. While it is difficult to quantify the probability of the 73.55 threat, it is generally accepted that it is small, comparable probably to other design basis type events. The probability of a fire which is spontaneous and located in or in close proximity to a vital area of the plant and is serious enough to pose a significant safety concern is also small. It would appear, therefore, that the random coincidence of these two unlikely events would be sufficiently small to not

require protection against their simultaneous occurrence. In addition, it should be noted that the short time period (30 minutes) for which several members of the security force would be dedicated to the fire brigade would further reduce the likelihood of coincidence.

As neither of the two potential rationales appear to preclude the use of members of the security force in the event of a fire the staff has concluded that the short assignment of security personnel from the armed response force or other available security personnel to the fire brigade under these conditions would be acceptable.

To ensure a timely and effective response to a fire, while still preserving a flexible security response, the staff believes that the fire brigade should operate in the following manner. In the event of an internal fire, all five members of the fire brigade should be dispatched to the scene of the fire to assess the nature and seriousness of the fire. Simultaneously, the plant security force should be actively evaluating the possibility of any security threat to the plant and taking any actions which are necessary to counter that threat. For external fires, a lesser number than the five-man brigade should respond for assessment and fire fighting. As the overall plant situation becomes apparent it would be expected that the most effective distribution of manpower between plant operations, security and fire protection would be made, allowing a balanced utilization of manpower resources until offsite assistance becomes available. The manpower pool provided by the plant operations personnel and security force are adequate to respond to the occurrence of a design basis fire or a security threat equivalent to the 73.55 performance requirements. It is also recognized that other, more likely combinations of postulated fires and security threats of a lesser magnitude than the design basis, could be considered. While the probabilities of these higher likelihood events may be sufficient to warrant protecting against them in combination, the manpower requirements required to cope with each event would be similarly reduced thereby allowing adequate coverage by plant personnel.

Conclusion

The staff believes that it would be reasonable to allow a limited amount of sharing of plant personnel in satisfying the requirements of plant operation, security, and fire protection. An acceptable sharing scheme would entail reliance on two plant operators and three members of the security organization to constitute the fire brigade. Since availability of the full fire brigade would only

be required for fires with potential for serious damage, actual distribution of plant personnel during a plant emergency would be governed by the exigencies of the situation. Of course, all personnel assigned to the fire brigade would have to fulfill all applicable training requirements. It should also be recognized that the diversion of personnel to the fire brigade would be of short duration and that substantial additional offsite assistance would be forthcoming in accordance with the emergency and contingency plan developed for each facility. In evaluating licensee proposals for manpower sharing due consideration will also have to be made of unique facility characteristics, such as terrain and plant lay-out, as well as the overall strengths of the licensee's fire and security plans. Minimum protection levels in either area could preclude the sharing of manpower.

Attachment A

Staff Position

Minimum Fire Brigade Shift Size

INTRODUCTION

Nuclear power plants depend on the response of an onsite fire brigade for defense against the effects of fire on plant safe shutdown capabilities. In some areas, actions by the fire brigade are the only means of fire suppression. In other areas, that are protected by correctly designed automatic detection and suppression systems, manual fire fighting efforts are used to extinguish: (1) fires too small to actuate the automatic system; (2) well developed fires if the automatic system fails to function; and (3) fires that are not completely controlled by the automatic system. Thus, an adequate fire brigade is essential to fulfill the defense in depth requirements which protect safe shutdown systems from the effects of fires and their related combustion by-products.

DISCUSSION

There are a number of factors that should be considered in establishing the minimum fire brigade shift size. They include:

1) plant geometry and size;
2) quantity and quality of detection and suppression systems;
3) fire fighting strategies for postulated fires;
4) fire brigade training;
5) fire brigade equipment; and
6) fire brigade supplements by plant personnel and local fire department(s).

In all plants, the majority of postulated fires are in enclosed windowless structures. In such areas, the working environment of the brigade created by the heat and smoke buildup within the enclosure, will require the use of self-contained breathing apparatus, smoke ventilation equipment, and a personnel replacement capability.

Certain functions must be performed for all fires, i.e., command brigade actions, inform plant management, fire suppression, ventilation control, provide extra equipment, and account for possible injuries. Until a site specific review can be completed, an interim minimum fire brigade size of five persons has been established. This brigade size should provide a minimum working number of personnel to deal with those postulated fires in a typical presently operating commercial nuclear power station.

If the brigade is composed of a smaller number of personnel, the fire attack may be stopped whenever new equipment is needed or a person is injured or fatigued. We note that in the career fire service, the minimum engine company manning considered to be effective for an initial attack on a fire is also five, including one officer and four team members.

It is assumed for the purposes of this position that brigade training and equipment is adequate and that a backup capability of trained individuals exist whether through plant personnel call back or from the local fire department.

POSITION

1. The minimum fire brigade shift size should be justified by an analysis of the plant specific factors stated above for the plant, after modifications are complete.

2. In the interim, the minimum fire brigade shift size shall be five persons. These persons shall be fully qualified to perform their assigned responsibility, and shall include:

   One Supervisor - This individual must have fire tactics training. He will assume all command responsibilities for fighting the fire. During plant emergencies, the brigade supervisor should not have other responsibilities that would detract from his full attention being devoted to the fire. This supervisor should not be actively engaged in the fighting of the fire. His total function should be to survey the fire area, command the brigade, and keep the upper levels of plant management informed.

   Two Hose Men - A 1.5 inch fire hose being handled within a windowless enclosure would require two trained individuals. The two team members are required to physically handle the active hose line and to protect each other while in the adverse environment of the fire.

   Two Additional Team Members - One of these individuals would be required to supply filled air cylinders to the fire fighting members of the brigade and the second to establish smoke ventilation and aid in filling the air cylinder. These two individuals would also act as the first backup to the engaged team.

ATTACHMENT B

4. a. Assignments of personnel meeting ANSI N18.1-1971 qualifications, Section 4.3.1 or Section 4.5.1, should be made to onsite shift operating crews in numbers not less than the following:

For a station having one licensed unit, each shift crew should have at least three persons at all times, plus two additional persons when the unit is operating. For a multi-unit station, each shift crew should have at least three persons per licensed unit at all times, plus one additional person per operating unit.

b. Operator license qualifications of persons assigned to operating shift crews should be as follows:

(1) A licensed senior operator who is also a member of the station supervisory staff should be onsite at all times when at least one unit is loaded with fuel.

(2) For any station with more than one reactor containing fuel, (1) the number of licensed senior operators onsite at all times should not be less than the number of control rooms from which the fueled units are monitored, and (2) the number of licensed senior operators should not be less than the number of reactors operating.

(3) For each reactor containing fuel, there should be at least one licensed operator in the control room at all times. Shift crew compositions should be specified such that this condition can be satisfied independently of licensed senior operators assigned to shift crews to meet the criteria of (1) and (2) above.

(4) For each control room from which one or more reactors are in operation, an additional operator should be onsite and available to serve as relief operator for that control room. Shift crew compositions should be specified such that this condition can be satisfied independently of (1), (2), and (3), and for each such control room.

c. Radiation protection qualifications of at least one person on each operating shift should be as follows:

The management of each station having one or more units containing fuel should either, (1) qualify and designate at least one member of each shift operating crew to implement radiation protection procedures, including routine or special radiation surveys using portable radiation detectors, use of protective barriers and signs, use of protective clothing and breathing apparatus, performance of contamination surveys, checks on radiation monitors, and limits of exposure rates and accumulated dose, or (2) assign a health physics technician to each shift, such assignment to be in addition to those assigned to shift operating crews in accordance with (a) and (b) above.

III. REVIEW PROCEDURES

Selection and emphasis of various aspects of the areas covered by this review plan will be made by the reviewer on each case. The judgment on the areas to be given attention during

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

Docket No. 50-220

April 21, 1978

All Power Reactor Licensees and Applicants

Gentlemen:

This letter and enclosed NUREG 0219, Draft 2 are being sent to all licensees authorized to operate a nuclear power reactor and to all applicants with applications for a license to operate or construct a power reactor.

NUREG 0219 outlines the current position of the NRR staff on implementation of the requirement to upgrade the qualification, training and equipping of security personnel. Since this is a draft, I would appreciate your comments by May 19, 1978 so that actual plant operating experience with security personnel can be factored into the final document.

Sincerely,

James R. Miller

James R. Miller, Assistant Director
for Reactor Safeguards
Division of Operating Reactors

Enclosure:
NUREG 0219

Niagara Mohawk Power Corporation

cc: Eugene B. Thomas, Jr., Esquire
LeBoeuf, Lamb, Leiby & MacRae
1757 N Street, N. W.
Washington, D. C. 20036

Anthony Z. Roisman
Natural Resources Defense Council
917 15th Street, N. W.
Washington, D. C. 20005

Oswego County Office Building
46 E. Bridge Street
Oswego, New York 13126

Docket No. 50-220

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

April 14, 1978

All Power Reactor Licensees

Gentlemen:

This letter is being sent to all licensees authorized to operate a nuclear power reactor and to all applicants with applications for a license to operate a power reactor (FSAR docketed) to advise you that the Nuclear Regulatory Commission has forwarded to the FEDERAL REGISTER a notice of meeting, subject "Implementation of 10 CFR 73.55 Requirements and Status of Research for Physical Protection of Licensed Activities in Nuclear Power Reactors against Industrial Sabotage" to be held May 11-12, 1978 in Albuquerque, New Mexico. This meeting will be open to the public.

The first day of the meeting will be taken up with presentations by Sandia Laboratories and Los Alamos Scientific Laboratories (LASL) covering their physical security research efforts. Sandia has extensive experience in the analysis of barriers, intrusion alarms, vital area identification and analysis, and security system design, modeling and evaluation. Additionally, Sandia provides consultant services to DOE and NRC in most facets of safeguards. LASL has considerable experience in explosives and detection systems research. LASL also provides consultant services to the NRC and is currently assisting the NRC in the review of nuclear power plants physical security plans.

The second day will begin with an address by Mr. Victor Stello, Jr., Director, Division of Operating Reactors, Office of Nuclear Reactor Regulation. The theme for this day will be safeguards issues and answers in the implementation of 10 CFR 73.55. Discussions of the major issues that have surfaced during physical security plan reviews will be moderated by Mr. James R. Miller, Assistant Director for Reactor Safeguards, Division of Operating Reactors, Office of Nuclear Reactor Regulation. These discussions will provide for an exchange between licensee representatives and NRC Safeguards personnel on questions related to implementation of the requirements of 10 CFR 73.55.

If you have any safeguards questions that you would like discussed at the meeting, please let us know and we will modify the enclosed agenda if time permits. For any further information or comments, please contact Frank Pagano of my staff (301-492-7846).

Sincerely,

Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation

Enclosure:
Agenda

INDUSTRY MEETING - ALBUQUERQUE MAY 11-12

Status of Research

Meeting Agenda - May 11, 1978

8:30 - 8:45	- Introduction to the conference - Opening Remarks by James R. Miller, Assistant Director for Reactor Safeguards, DOR, NRR, NRC
8:45 - 9:00	- Overview of effectiveness assessment methodology - William M. Murphey, Chief, Technical Support Branch, DSFER, RES, NRC
9:00 - 9:45	- Facility characterization and target identification - Sandia Laboratories
9:45 - 10:00	- Break
10:00 - 10:45	- Detection of unauthorized access by stealth or deceit - Lawrence Livermore Laboratories
10:45 - 11:30	- Encounter probability - Sandia Laboratories
11:30 - 12:15	- Engagement outcome probability - Sandia Laboratories
12:15 - 2:15	- Lunch
2:15 - 3:00	- Design to protect against reactor sabotage - Sandia Laboratories
3:00 - 3:30	- Catalog of physical protection equipment - H. Michael Hawkins, Chief, Operational Support Branch, DSFER, RES, NRC
3:30 - 3:34	- Break
3:45 - 4:30	- DOE safeguards R&D program - Division of Safeguards and Security, DOE/Sandia Laboratories
4:30 - 5:00	- Explosion detection - Los Alamos Scientific Laboratory

Implementation of 10 CFR 73.55

Meeting Agenda - May 12, 1978

9:00 - 9:30	- NRC Safeguards responsibilities - An opening address by Victor Stello, Jr., DOR, NRR, NRC
9:30 - 9:50	- Actions to be taken by NRC if an unsatisfactory SPER results
9:50 - 10:10	- 10 CFR 21/10 CFR 73.55 relationships
10:10 - 10:30	- Equipment acceptance/Alarms/Locks/etc.
10:30 - 10:50	- Break
10:50 - 11:10	- Vital Areas/Activity control
11:10 - 11:30	- Threat/Insider/General performance requirements
11:30 - 12:00	- Screening/Search/Package control/Escorts
12:00 - 2:00	- Lunch
2:00 - 2:20	- Use of deadly force/Guard qualification/Response force
2:20 - 2:50	- Overall physical security program performance
2:50 - 3:10	- Review procedure: Security plan - Security Workbooks - SPER
3:10 - 3:30	- August 24 date - extension of time/10 CFR 50.54(p) changes/Licensee amendment fees
3:30 - 3:50	- Break
3:50 - 4:10	- New & Proposed rules/Contingency Planning/Guard Training/ Clearance/Upgrade
4:10 - 4:30	- I&E/NRR relationship with respect to inspection after August 24
4:30 - 5:00	- Closing remarks - NRC expectations, etc. - Victor Stello, Jr.

FEBRUARY 3 1978

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Docket No. 50-220

Niagara Mohawk Power Corporation
ATTN: Mr. Donald P. Dise
Vice President - Engineering
300 Erie Boulevard West
Syracuse, New York 13202

Gentlemen:

We have completed a preliminary review of your January 24, 1977 request for Technical Specification changes to Appendix B for Nine Mile Point Unit No. 1. We have concluded that we need additional information to complete our review.

Please provide responses to the items of information identified in the enclosure within 30 days from receipt of this letter. If you have any questions, please contact us.

Sincerely,

Original signed by

George Lear, Chief
Operating Reactors Branch #3
Division of Operating Reactors

Enclosure:
Request for Additional Information

cc w/enclosure:
see next page

OFFICE	ORB#3	ORB#3				
SURNAME	SNowicki:acr	GLear				
DATE	2/2/78	2/2/78				

cc: Eugene B. Thomas, Jr., Esquire
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ENCLOSURE

REQUEST FOR ADDITIONAL INFORMATION

NINE MILE POINT UNIT NO. 1

ENVIRONMENTAL TECHNICAL SPECIFICATIONS

1. Water Quality

The following changes have been proposed to Specification 3.1.1.a(1) concerning monitoring of water quality parameters:

Reduction in sampling frequency from monthly to bimonthly

Reduction in number of sampling stations

Change in location of sampling stations

Deletion of requirement to monitor Chromium and Radioactivity (gross alpha, beta, gamma and tritium)

Deletion of requirement of sample during November and December.

Provide a basis and justification for each of the above proposed changes, particularly where the proposed change is different from the recommendation in Section 6-1 of the FES. Provide an evaluation of how these proposed changes will affect the capability of this monitoring program to detect changes in the water quality of Lake Ontario caused by plant operation.

2. Zooplankton and Phytoplankton

The following changes have been proposed to Specification 3.1.2a(1) concerning monitoring of zooplankton and phytoplankton:

Reduction in sampling frequency

Reduction in number of replicate samples required

Deletion of requirement to sample during November and December

Reduction in the number of stations sampled

Change in depth of phytoplankton samples from 1m to 1/2m depth

Deletion of chlorophyll a and ^{14}C determination for phytoplankton

Deletion of requirement to sample Mysis, Pontoporeia and Gammarus with additional gear if these species are not sampled properly with the other zooplankton in oblique tows

Deletion of requirement to identify organisms to the lowest possible taxon

Deletion of requirement to sample at 3 different depths at one station

Deletion of requirement for density determiniation by species.

Provide a basis and justification for each of the above changes with particular attention as to why the proposed program should differ from that outlined in Section 6-1 of the FES. Provide an evaluation of the effect these changes would have on the monitoring program's ability to detect plant-induced changes in the distribution, abundance and species composition of the zooplankton and phytoplankton communities.

3. Periphyton

Provide a basis and justification for deleting Specification 3.1.2.a(1)(iii), which requires periphyton sampling, from the proposed environmental monitoring program, particularly in view of the emphasis that Section 6-1 of the FES places on periphyton monitoring during the operating phase in order to assess any changes that may occur. Provide an evaluation of the effect that this change will have on the ability of the environmental monitoring program to detect plant-induced changes in the aquatic ecosystem.

4. Benthos

The following changes have been proposed to the benthos sampling program required by Specification 3.1.2.a(1)(iv):

Reduction in number of replicate samples required

Reduction in the number of stations sampled

Deletion of requirement to sample in November and December

Deletion of requirement to enumerate organisms collected and identify to the lowest possible taxon

Deletion of requirement for determination of total biomass and number of organisms per unit area of substrate at each station.

Provide a basis and justification for each of the above proposed changes with particular emphasis on those changes that would make the new program deviate from the recommendations in Section 6-1 of the FES. Provide an evaluation of how these changes will affect the capability of the monitoring program to detect changes in the species composition and distribution of benthic organisms caused by plant operation.

5. Fish

The following changes have been proposed to Specification 3.1.2.a(1)(v) concerning far-field monitoring of fish:

Deletion of requirement to sample with trawls, seines and trap nets

Deletion of requirement to sample in November and December

Reduction in sampling frequency

Deletion of requirement for age and growth studies on yellow perch, white perch, and smallmouth bass

Deletion of requirement for monthly coefficient of maturity by sex

Deletion of requirement to determine fecundity or spawning periodicity of selected species collected in the vicinity of Nine Mile Point.

Provide a basis and justification for each of the above proposed changes with particular emphasis on how the proposed sampling program will be compatible with the recommendations in Section 6-1 of the FES. Provide information demonstrating that the requirements of ETS Section 3.1.2.a(1)(v) concerning age and growth studies, coefficient of maturity, fecundity and spawning periodicity have been adequately met. Provide an evaluation of how these proposed changes will affect the capability of the monitoring program to detect changes in fish populations caused by plant operation.

6. Ichthyoplankton

The following changes have been proposed to Section 3.1.2.a(1)(vi) concerning ichthyoplankton sampling:

Reduction in number of stations sampled

Change in station location

Deletion of requirement to sample at mid-depth

Reduction in sampling frequency

Deletion of requirement to sample in September through December

Deletion of requirement for night sampling.

Provide a basis and justification for each of the above changes with special emphasis in addressing why the program has been much reduced from that recommended in Section 6-1 of the FES. Also provide an evaluation of the effect that these proposed changes would have on the ability of the monitoring

program to detect and document adverse impacts on the fish populations near Nine Mile Point caused by operation of the plant.

7. Impingement

The following changes have been proposed to Specification 3.1.2.a(2) concerning impingement monitoring:

Reduction in sampling frequency

Change in number of fish of each species analyzed for weight and length

Establishment of method of estimating large numbers of fish impinged by using mean weight and volume.

Provide a basis and justification for each of the above changes in the form of a report analyzing and evaluating the results of impingement monitoring since Nine Mile Point Unit 1 began operation. The report should include: (a) an analysis of the environmental impact of impingement at the plant; (b) proposed limiting conditions and report levels for fish impingement; (c) appropriate substantiated recommendations for modifications or discontinuance of the various portions of the study; (d) how the proposed changes will meet the recommendations of Sections 5.5.2.a, 6.1 and the Summary and Conclusions of the FES; and (e) how well the modified program will detect and document adverse impacts on fish populations near Nine Mile Point caused by impingement.

8. Entrainment

Provide a basis and justification for deleting Specification 3.1.2(a)(3), which requires entrainment monitoring, in the form of a report analyzing and

evaluating the results of entrainment monitoring since the plant began operation. The report should include: (a) proposed final values of protection limits and report levels and/or appropriate substantiated recommendations for modifications or discontinuance of the various portions of this study; (b) identification of those biological parameters affected by entrainment that require monitoring throughout the life of the plant; (c) an analysis of the environmental impact of entrainment of aquatic organisms during operation of the plant; (d) how the proposed changes will meet the recommendations of the Summary and Conclusions of the FES and the recommendations in Section 6-1 of the FES, and (e) how well the modified program will detect and document adverse impact on aquatic organisms near Nine Mile Point caused by entrainment.