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D. L. Ziemann, DRL
R. H. Vollmer, DRL
C. J. DeBevec, DRL
R. M. Diggs, DRL
D. Thompson, DRL

AUG 27 1970

Docket No. 50-220

Niagara Mohawk Power Corporation
ATTN: Mr. Minot H. Pratt
Vice President and
Executive Engineer
300 Erie Boulevard West
Syracuse, New York 13202

Gentlemen:

In the course of our review of your application dated April 20, 1970, pertaining to a proposed power increase for the Nine Mile Point Nuclear Station, we find that additional information is necessary to complete our evaluation. The specific information requested is described in the enclosure.

Your reply should be submitted as an amendment to your application for a power increase. Please contact us if you desire additional discussion or clarification of the material requested.

Sincerely,

Original Signed by
Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

Enclosure:
Request for Additional
Information

cc: Arvin E. Upton, Esquire
LeBoeuf, Lamb, Leiby & MacRae
1821 Jefferson Place, N. W.
Washington, D. C. 20036

Dispatched 8/27/70

Applied

OFFICE ▶	DRL	DRL	DRL	DRL	DRL	
SURNAME ▶	CJDeBevec DLZiemann:sjg	RMDiggs	DJSkovholt	FSchroeder	PAMorris	
DATE ▶	8/14/70	8/14/70	8/19/80	8/24/70	8/27/70	

Original Signed by
Peter A Morris

REQUEST FOR ADDITIONAL INFORMATION

PERTAINING TO APPLICATION FOR POWER INCREASE

OF NINE MILE POINT NUCLEAR STATION

1. Describe the changes, if any, in design fuel exposure with regard to maximum and average expected exposures. Include discussion of the planned refueling cycles. Describe your surveillance program for fuel which operates beyond current production fuel experience considering the proposed combinations of fuel linear power generation rates and exposures under both normal and anticipated transient modes of operation. Indicate the effect of the power increase on the predicted core reactivity requirements.

2. The FSAR states that the design lifetime neutron exposure of the reactor vessel is 5×10^{17} nvt (energies 1 mev). Describe the effect of the power increase on this estimated lifetime exposure and the effect on resultant NDTT considerations.

3. The Nine Mile Point FSAR and Supplements do not adequately describe the isolation valves in the instrument lines which penetrate primary containment. Describe these valves in terms of isolation capability, reliability and testability as compared with the isolation valves in other lines penetrating primary containment. Include a discussion of the testing schedule applicable to these valves.

4. On page II-15 of your "Technical Supplement to Petition to Increase Power Level" it is stated that core cooling analyses presented in the "Fifth Supplement to FSAR" were performed for a power rating of 1850 MWt. Please define in detail the applicability of the referenced information to the proposed 1850 MWt design rating.

5. Your safety analysis considers the inadvertent actuation of one bypass valve. Provide the results of an analysis of a transient associated with inadvertent actuation of all bypass valves.

6. Provide a summary description of the current status of the design modifications which were to be made during construction (page 109 of the Fifth Supplement to the FSAR).

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7. The proposed change in Technical Specification bases (fourth sentence of last paragraph of page 7), involving continued operation with a peaking factor greater than 3.06, lacks adequate technical support. Describe in detail the basis on which you propose to delay refueling while continuing to operate the old core under conditions of peaking factors greater than 3.06.

8. The proposed change in Technical Specifications (deletion of second sentence of first paragraph, page 9) is explained by stating that the analytical basis does not include the assumption of bypass failure in the determination of dwell time. Define what is meant by dwell time and state the basis for excluding consideration of expected transients (i.e., bypass valve malfunctions).

9. Your letter of September 24, 1969, reported on the status of the items identified by the ACRS for further action. Please describe the current status of all of these items except the items entitled: "Periodic Inspection of the High Pressure Coolant System", "Primary System Leak Detection Studies", and "Safety Review and Audit Board". In view of the proposed increase in power, describe in detail your plans for monitoring primary coolant system internal vibration during the escalation in power.

10. In light of the recent incident at Dresden-2 caused by inadvertent opening of the bypass valves, as described in the enclosed letter from Commonwealth Edison dated July 6, 1970, provide the following information as applicable to Nine Mile Point Nuclear Station:

    a. Evaluate the adequacy of the NMP primary containment design temperature of 310°F for the maximum temperatures and other environmental conditions which might be reached as a result of steam release inside the drywell. Include the basis of qualification of all safety and protective system components and instrument cables required to function following a LOCA to operate at the maximum temperatures achievable for the requisite period of time.

    b. Evaluate the safety valve installation to show that discharge from one or more valves would not result in loss of function or other undesirable reaction with the other safety valves.

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c. Discuss the adequacy of the control logic and trip settings of equipment which is designed to operate in the event of an expected transient or incident. The particular items of concern are the isolation condenser with respect to initiation and isolation conditions; the pump tripout and automatic restart capability of the feedwater system; and the availability of offsite and onsite power for ECCS component sequencing.

d. Describe the capability of drywell instrumentation to provide wide range temperature, pressure and radioactivity information directly to the console operator.

e. Evaluate the adequacy of plant emergency procedures and expected operator responses if a similar event were to occur at NMP. The evaluation should include a discussion of the criteria for ECCS component shutoff, for actuation of drywell sprays, and for breaking containment isolation including venting through the standby gas treatment system.

11. We understand, as result of meeting on August 20, 1970, that current considerations in the design of the Fitzpatrick plant include changes in Nine Mile Point circuit interconnections which provide offsite power. Please describe in detail these changes and your evaluation of the effect on safety resulting from these changes.

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Insert in lieu

4. On page II-15 of your "Technical Supplement To Petition To Increase Power Level" it is stated that (core cooling) analyses presented in the "Fifth Supplement to FSAR" were performed for a power rating of 1850 Mwt. Please define in detail the applicability of the referenced information to the proposed 1850 Mwt design rating.

Insert

11. We understand (as result of meeting on August 20, 1970,) that current considerations in the design of the Fitzpatrick plant include changes in [redacted] Nine Mile Point circuit [redacted] interconnections which providing offsite power. Please describe in detail these changes and your evaluation of the effect on safety resulting from these changes.

REQUEST FOR ADDITIONAL INFORMATION

PERTAINING TO APPLICATION FOR POWER INCREASE

OF NINE MILE POINT NUCLEAR STATION

expected

1. Describe the changes, if any, in design fuel exposure with regard to maximum and average ~~target~~ exposures. Include discussion ~~on~~ of the planned refueling cycles. Describe your surveillance program for fuel which operates beyond current production fuel experience considering the proposed combinations of fuel linear power generation rates and exposures under both normal and anticipated transient modes of operation. Indicate the effect of the power increase on the predicted core reactivity requirements.

2. The FSAR states that the design lifetime neutron exposure of the reactor vessel is 5×10^{17} nvt (energies > 1 mev). Describe the effect of the power increase on this estimated lifetime exposure and the effect on resultant NDTT considerations.

3. The Nine Mile Point FSAR and Supplements do not adequately describe the isolation valves in the instrument lines which penetrate primary containment. Describe these valves in terms of isolation capability, reliability and testability as compared with the isolation valves in other lines penetrating primary containment. Include a discussion of the testing schedule applicable to these valves.

Consider whether multi-node analysis should be done re recent concern on ECCS performance. Discuss with DCV?

4. Regarding the ECCS, your application makes reference to earlier analyses (Fifth Supplement) purporting to show that these core cooling analyses were performed for a power rating of 1850 MWt. Please define in detail the applicability of the referenced information to the proposed design rating.

5. Your safety analysis considers the inadvertent actuation of one bypass valve. Provide the results of a transient associated with inadvertent actuation of all bypass valves.

6. ~~A summary of the design modifications previously committed for construction is on page 109 of the Fifth Supplement to the FSAR.~~ Provide a summary description of the current status of these ~~items.~~ design modifications which were to be made during construction (page 109 ...)

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7. The proposed change in Technical Specification bases (fourth sentence of last paragraph of page 7), involving continued operation with a peaking factor greater than 3.06, lacks adequate technical support. Describe in detail the basis on which you propose to delay refueling while continuing to operate the old core under conditions of peaking factors greater than 3.06.

8. The proposed change in Technical Specifications (deletion of second sentence of first paragraph, page 9) is explained by stating that the analytical basis does not include the assumption of bypass failure in the determination of dwell time. Define what is meant by dwell time and ~~the bases on which~~ consideration of expected transients (i.e., bypass valve malfunctions). ~~are excluded.~~ state the basis for excluding

9. Your letter of September 24, 1969, reported on the status of the items identified by the ACRS for further action. Please describe the current status of all of these items except the items entitled, (1) "Periodic Inspection of the High Pressure Coolant System", (2) "Primary System Leak Detection Studies", and (3) "Safety Review and Audit Board". In view of the proposed increase in power, describe in detail your plans for monitoring primary coolant system internal vibration during the escalation in power.

Why not these? (1,2) Covered by Assumption of Power approval dated 6/19/70

(3) Satisfactorily implemented.

10. In light of the recent incident at Dresden-2 caused by inadvertent opening of the bypass valves, as described in the enclosed letter from Commonwealth Edison dated July 6, 1970, provide the following information as applicable to Nine Mile Point Nuclear Station:

a. Evaluate ~~An evaluation of~~ the adequacy of the NMP primary containment design temperature of 310°F for the maximum temperatures and other environmental conditions which might be reached as a result of steam release inside the drywell. Include the basis of qualification of all safety and protective system components and instrument cables required to function following a LOCA to operate at the maximum temperatures achievable for the requisite period of time.

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c. Discuss ~~The~~ adequacy ~~should be shown for~~ of the control logic and trip settings of equipment which is designed to operate in the event of an expected transient or incident. The particular ~~equipment~~ items of concern ~~is~~ are the isolation condenser with respect to initiation and isolation conditions; the pump tripout and automatic restart capability of the feedwater system; and the availability of offsite and onsite power for ECCS component sequencing.

d. Describe ~~An evaluation of~~ the capability of drywell instrumentation to provide wide range temperature, pressure and radioactivity information directly to the console operator.

e. Evaluate ~~An evaluation of~~ the adequacy of plant emergency procedures and expected operator responses if a similar event ~~occurred~~ were to occur at NMP. The evaluation should include a discussion of the criteria for ECCS component shutoff, for actuation of drywell sprays, and for breaking containment isolation including venting through the standby gas treatment system.

11.

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DATE ▶						

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

AUG 27 1970

Docket No. 50-220

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1821 Jefferson Place, N. W.
Washington, D. C. 20036

REQUEST FOR ADDITIONAL INFORMATION

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OF NINE MILE POINT NUCLEAR STATION

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