Docket No. 50-333

OCT 24

Power Authority of the State of
New York
ATTN: Mr. George T. Berry
General Manager and
Chief Engineer
10 Columbus Circle
New York, New York 10019

Gentlemen:

The Commission has issued the enclosed Amendment No. 3 to Facility License No. DPR-59, for the James A. FitzPatrick Nuclear Power Plant. This amendment includes Change No. 3 to the Technical Specifications, and is in response to your request of March 13, 1975, and staff discussions.

This amendment authorizes ~~24~~ 25 miscellaneous Appendix A and two Appendix B Technical Specification changes. The Appendix A changes relate to the correction of typographical errors, clarification of the specifications, and make the specifications consistent with the Final Safety Analysis Report (FSAR). The Appendix B changes establish consistency in the Site Radiological Environmental Monitoring Program for the two plants on the site (Facility License No. DPR-59 and License No. DPR-63), and achieve consistency with Appendix A in the administrative controls of the Technical Specifications.

We have evaluated the potential for environmental impact of operating the plant with the proposed changes to Appendix B. The changes apply only to administrative details and to modifications in the Site Radiological Environmental Monitoring Program. The proposed changes will not cause a change in effluent types or amounts, do not result in a change in authorized power level, and will not result in any significant environmental impact. Having made this determination, the Commission has further concluded that pursuant to Section 10 CFR Part 51.5(d)(4), an environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

DR

OFFICE➤						
SURNAME➤						
DATE➤						

Power Authority of the State of New York

As described in our enclosed Safety Evaluation two of the proposed Technical Specification changes were withdrawn and one was modified pursuant to discussions with members of your staff. We will conclude that these modifications to your request are acceptable to you unless you inform us in writing to the contrary.

Copies of the related Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

/S/

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Reactor Licensing

Enclosures:
1. Amendment No. 3
2. Safety Evaluation
3. Federal Register Notice

cc: See next page

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POWER AUTHORITY OF THE STATE OF NEW YORK

AND

NIAGARA MOHAWK POWER CORPORATION

DOCKET NO. 50-333

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 3
License No. DPR-59

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Power Authority of the State of New York and Niagara Mohawk Power Corporation, (the licensees) dated March 13, 1975, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations; and

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

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SURNAME ➤						
DATE ➤						

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility License No. DPR-59 is hereby amended to read as follows:

   "(2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. ."

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Reactor Licensing

Attachment:
Change No. 3
  Technical Specifications

Date of Issuance:
October 24, 1975

OFFICE➤						
SURNAME➤						
DATE➤						

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-333

POWER AUTHORITY OF THE STATE OF NEW YORK

AND

NIAGARA MOHAWK POWER CORPORATION

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Notice is hereby given that the U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No.3 to Facility Operating License No. DPR-59 issued to the Power Authority of the State of New York and Niagara Mohawk Power Corporation, located in Scriba, Oswego County, New York. The amendment is effective as of its date of issuance.

This amendment authorizes ~~27~~ 25 miscellaneous Appendix A and two Appendix B Technical Specification changes. The Appendix A changes relate to the correction of typographical errors, clarification of the specifications, and make the specifications consistent with the Final Safety Analysis Report (FSAR). The Appendix B changes establish consistency in the Site Radiological Environmental Monitoring Program for the two plants on the site (Facility License No. DPR-59 and License No. DPR-63, and achieve consistency with Appendix A in the administrative controls of the Technical Specifications.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations

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DATE➤						

in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment is not required since the amendment does not involve a significant hazards consideration.

For further details with respect to this action, see (1) the application for amendment dated March 13, 1975, (2) Amendment No. 3 to License No. DPR-59, with Change No. 3 and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW., Washington, D.C. and at the Oswego City Library, 120 East Second Street, Oswego, New York.

A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this **24th, October 1975**

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid , Chief
Operating Reactors Branch #4
Division of Reactor Licensing

DOCKET

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

Docket No. 50-333

Power Authority of the State of
New York
ATTN: Mr. Goerge T. Berry
General Manager and
Chief Engineer
10 Columbus Circle
New York, New York 10019

OCT 24 1975

Gentlemen:

The Commission has issued the enclosed Amendment No. 3 to Facility License No. DPR-59, for the James A. FitzPatrick Nuclear Power Plant. This amendment includes Change No. 3 to the Technical Specifications, and is in response to your request of March 13, 1975, and staff discussions.

This amendment authorizes 25 miscellaneous Appendix A and two Appendix B Technical Specifications changes. The Appendix A changes relate to the correction of typographical errors, clarification of the specifications, and make the specifications consistent with the Final Safety Analysis Report (FSAR). The Appendix B changes establish consistency in the Site Radiological Environmental Monitoring Program for the two plants on the site (Facility License No. DPR-59 and License No. DPR-63), and achieve consistency with Appendix A in the administrative controls of the Technical Specifications.

We have evaluated the potential for environmental impact of operating the plant with the proposed changes to Appendix B. The changes apply only to administrative details and to modifications in the Site Radiological Environmental Monitoring Program. The proposed changes will not cause a change in effluent types or amounts, do not result in a change in authorized power level, and will not result in any significant environmental impact. Having made this determination, the Commission has further concluded that pursuant to Section 10 CFR § 51.5(d)(4), an environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Power Authority of the State of New York

As described in our enclosed Safety Evaluation two of the proposed Technical Specification changes were withdrawn and one was modified pursuant to discussions with members of your staff. We will conclude that these modifications to your request are acceptable to you unless you inform us in writing to the contrary.

Copies of the related Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Reactor Licensing

Enclosures:
1. Amendment No. 3
2. Safety Evaluation
3. Federal Register Notice

cc: See next page

cc w/enclosures:
Scott B. Lilly, General Counsel
Power Authority of the
State of New York
10 Columbus Circle
New York, New York 10019

Arvin E. Upton, Esquire
LeBoeuf, Lamb, Leiby and MacRae
1757 N Street, NW.
Washington, D. C. 20036

Lauman Martin, Esquire
Senior Vice President
and General Counsel
Niagara Mohawk Corporation
300 Erie Boulevard West
Syracuse, New York 13202

Mr. Z. Chilazi
Power Authority of the
State of New York
10 Columbus Circle
New York, New York 10019

J. Bruce MacDonald, Deputy
Commissioner and Counsel
New York State Department of
Commerce and Counsel to the
Atomic Energy Council
99 Washington Avenue
Albany, New York 12210

Ecology Action
c/o Richard Goldsmith
Syracuse University
College of Law
E. I. White Hall Campus
Syracuse, New York 13210

Ms. Suzanne Weber
R.D. #3, West Lake Road
Oswego, New York 13126

Oswego City Library
120 East Second Street
Oswego, New York 13126

Mr. Robert P. Jones, Supervisor
Town of Scriba
R. D. #4
Oswego, New York 13126

Mr. Alvin L. Karkau
Chairman, County Legislature
County Office Building
46 East Bridge Street
Oswego, New York 13126

cc w/enclosures & incoming:
Dr. William E. Seymour
Staff Coordinator
New York State Atomic Energy
Council
New York State Department of
Commerce
112 State Street
Albany, New York 12207

Mr. Paul Arbesman
Environmental Protection Agency
26 Federal Plaza
New York, New York 10007

Anthony Z. Roisman, Esquire
Berlin, Roisman & Kessler
1712 N Street, NW
Washington, D.C. 20036

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

POWER AUTHORITY OF THE STATE OF NEW YORK

AND

NIAGARA MOHAWK POWER CORPORATION

DOCKET NO. 50-333

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 3
License No. DPR-59

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Power Authority of the State of New York and Niagara Mohawk Power Corporation, (the licensees) dated March 13, 1975, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations; and

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C.(2) of Facility License No. DPR-59 is hereby amended to read as follows:

   "(2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 3."

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid, Chief
Operating Reactors Branch #4
Division of Reactor Licensing

Attachment:
Change No. 3
Technical Specifications

Date of Issuance:
October 24, 1975

ATTACHMENT TO LICENSE AMENDMENT NO. 3

CHANGE NO. 3 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-59

DOCKET NO. 50-333

Revise Appendix A as follows:

Remove Pages	Insert Pages
1 & 2	1 & 2
29 & 30	29 & 30
63 thru 68	63 thru 68
79 thru 82	79 thru 82
85 & 86	85 & 86
91 & 92	91 & 92
115 & 116	115 & 116
123 & 124	123 & 124
165 & 166	165 & 166
179 & 180	179 & 180
199 & 200	199 & 200
247 & 248	247 & 248
271 thru 276	271 thru 276
283 & 284	283 & 284

Revise Appendix B as follows:

Remove Pages	Insert Pages
34 & 35	34 & 35
44 & 45	44 & 45

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 3 TO LICENSE NO. DPR-59

(CHANGE NO. 3 TO TECHNICAL SPECIFICATIONS)

POWER AUTHORITY OF THE STATE OF NEW YORK

AND

NIAGARA MOHAWK POWER CORPORATION

JAMES A. FITZPATRICK NUCLEAR POWER PLANT

DOCKET NO. 50-333

Introduction

By letter dated March 13, 1975, the Power Authority of the State of New York requested changes to the Technical Specifications appended to Facility Operating License DPR-59 for the James A. FitzPatrick Nuclear Power Plant. The letter requests 27 miscellaneous Appendix A Technical Specification changes and two Appendix B Technical Specification changes. Only those changes related to Appendix A Technical Specifications are discussed in this Safety Evaluation.

Discussion

The 27 proposed changes relate to : (1) correction of four typographical errors; (2) clarification of 19 TS; (3) establishment of consistency between the TS and the FSAR in one request; and (4) establishment of consistency between the FitzPatrick TS and the NRC Standard TS for the three final requests. These 27 change requests are described and evaluated in detail under the four groupings given above in the following section.

Evaluation

(1) The four following proposed TS changes relate to the correction of typographical errors.

TS 1.0D, p. 2, now reads ". . . and reactor pressure > 1005 psig" but should read ". . . and reactor pressure < 1005 psig."

TS 3.7A1, p. 165, now reads ". . . 3.5.F.3" but should read ". . . 3.5.F.2."

TS 4.7A5g, p. 179, now reads ". . . 3.7.A3a" but should read ". . . 3.7A5f."

Table 6.5-1, p. 272, under "Proposed Changes to Plant System" references the "Site Review Committee" but should reference the "Site Operations Review Committee." In addition the last two lines under Safety Review and Audit Board should have an "and" after the word "Spec."

The above group of four changes to the Technical Specifications serve to correct various errors, do not involve any significant hazards considerations, and are acceptable.

(2) The following group of 19 proposed Technical Specification changes relate to the clarification of the TS or restablishment of consistency between the TS and the FSAR.

Table 4.2-2, Item 3, p. 80. The licensee desires to delete the requirement for a simulated "auto actuation test" for the containment cooling subsystem logic system. As there is no automatic mode of operation for the containment cooling subsystem logic system, it is an acceptable change to delete the requirement. The inclusion of this requirement in the present TS was an error.

Table 4..2-2, Item 8, p. 80. This Item calls for a logic system functional test for the "Area Cooling for Safeguard System." Since the "Area Cooling for Safeguard System" has no logic system associated with it, the change is acceptable.

Table 4.2-3, p. 81. The licensee desires to reference the "Instrument Check" title heading of the Table to the definition of Instrument Check in TS 1.0.F.4. - Definitions; this change is acceptable as it clarifies the meaning of the tabular heading, "Instrument Check." In addition, the licensee desires to delete the requirement for "Instrument Check" of the Source Range Monitors and Intermediate Range Monitors channels related to line item 8 and 9, the "detector not in startup position." The deletion of the instrument check is acceptable as there are no instruments to check related to line items 8 and 9.

Notes, p. 85. Due to the tie-in of the reference of the Table 4.2-3 heading on p. 81 of the TS to the definition of instrument check in TS 1.0.F.4 an additional footnote, namely "(9) - See Technical Specification 1.0.F.4. - Definitions for meaning of term Instrument Check" is requested by the licensee. This change is acceptable.

Table 4.2-4, p. 82. The licensee desires to add the same reference to the "Instrument Check" title heading of the Table using the same basis as given for Table 4.2-3 described above; thus the change is acceptable.

Table 4.2-4, Logic System Functional Test, Line Item 4, p. 82. The licensee desires to delete the entire line item from the table wherein the frequency of the functional test of the logic system for the main control room ventilation monitor is specified. Since the monitor provides only an alarm and has no "logic system," the change is acceptable.

Table 6.5-1 pages 272 through 274. The licensee desired to make this table consistent within FSAR question response by making the General Manager and Chief Engineer of PASNY rather than the Niagara Mohawk Power Corporation Vice President of Engineering responsible for participation in the Safety Review and Audit Board. We advised the licensee that the requested change could not be made since it would be in conflict with the record in this docket which contains no evidence on which we could base a finding that PASNY is technically qualified to operate the facility. The licensee after discussion with us decided to withdraw this proposed change and resubmit it in the future as a separate licensing action.

Notes to Table 3.2-1, p. 65. The licensee desires to change note 8 in this Table to reflect the fact, as verified by the NRC staff, that the key lock manual bypass feature is no longer a part of the main steam isolation valve control circuitry. The circuit change was effected prior to issuance of the Operating License and was approved as part of the FSAR review. This change is acceptable as it removes an inconsistency between the current technical specifications and the FSAR.

TS 4.1B, p. 30. The licensee desires to insert the phrase, ". . . while in the Run Mode . . .," to line 2 of this TS between the words, "operation," and "the." The evaluation of this change will be included with the evaluation of the following three change requests due to the identical subject matter of all four requests.

TS 3.5.H, p. 123. The licensee desired to add the phrase, "During steady state power operation" to the beginning of the first sentence of this TS, but withdrew the request on September 10, 1975.

TS 4.5.H, p. 123. The licensee desires to insert the words, ". . . while in the Run Mode . . ." to line 2 of this TS between the words "operation" and "the."

TS 4.5.1, page 124. The licensee desires to insert the words, ". . . while in the Run Mode . . ." to line 1 of this TS between the words "operation" and "the."

The three above changes wherein the phrase, ". . . while in the Run Mode . . ." is added to the TS, are required since measurements of heat flux, peaking factor and linear heat generation rate can only be performed while the reactor is in the Run Mode because the average power range monitors, instrumentation required for these measurements, are functional only when the reactor is in the Run Mode. In addition, these measurements are required only when the reactor is in the Run Mode. The changes are acceptable because they clarify the intent of the TS and in addition they make the subject TS consistent with Section 7.7 of the FSAR.

Table 6.5-1, p. 273. The licensee desires to add the phrase, " . . . relative to nuclear safety." after the Site Review and Audit Board's specified responsibility in directing and evaluating the performance of the entire facility staff as stated on page 273 of the TS. The change is acceptable as it clarifies the Board's role in terms of nuclear safety, and as not having responsibilities in other aspects of plant operation.

TS 4.3B3a, p. 92. The licensee desires to add an out-of-sequence control rod withdrawal test prior to reaching the 20% power level during a reactor shutdown. This additional test on shutdown supplements the same test on power ascension. The change is acceptable as it further verifies the capability of the rod sequence control system to properly fulfill its function.

Table 3.7-1, p. 199. The licensee requested a change in closure time for one of the valves. After investigation it became clear that the closure time for another valve in Table 3.7-1 in the same system was in error also. In accordance with discussion with the licensee it was agreed to change the closure time for both of these valves (24 inch and 18 inch) to 120 seconds and 90 seconds respectively in accordance with footnote (7) to the Table. This change is acceptable because the purpose of the test is to functionally demonstrate proper remote manual valve operator function. The newly stated closure times accomplish this purpose and correct the previous closure times which were in error.

Figure 6.1-2, p. 283. The licensee desires to change the titles of the Assistant to the Plant Superintendent for Operation (Nine Mile Point Nuclear Station) to Operations Supervisor and the Assistant to the Plant Superintendent for Operation (FitzPatrick Nuclear Power Plant) to Operations Supervisor. These title changes are acceptable as they reflect the titles used in the present organization.

TS 6.3A2, p. 248. This change is related to the change above and makes the TS section that references Fig. 6.1-2 give the same position titles as Fig. 6.1-2. Only the position titles are revised, not the duties of the incumbents, thus the changes are acceptable.

TS 4.5.B.1, p. 116.- Two Change Requests. The licensee desires to delete, ". . . at a design head of 250 ft." from lines 5 and 6; and to replace the last phrase, ". . . at 168 ft. discharge head, is also a requirement." in the last sentence with, ". . . will be tested in accordance with Section 4.11D." These changes are requested in order to make TS 4.5.B.1 consistent with our operating license review as reflected in the FSAR in regard to the containment cooling subsystem surveillance requirements for the residual heat removal service water pumps and the emergency service water pump. FSAR Section 9.7.3.7 requires testing of the pumps only for availability, reliability and flow characteristics in order to perform their safety function, thus pressure testing is not required; therefore, the changes are acceptable.

The above group of 19 changes to the Technical Specifications do not involve or result from an unreviewed safety question nor revise any safety limit settings. We conclude that the group of 19 changes do not involve any significant hazards considerations.

(3) Table 3.3-2, p. 68. The licensee desires to revise the trip level settings for the RHR (LPCI) pump pressure from less than or equal to 50 psig to 50 psig ± 10 psig; and the core spray discharge pressure trip level setting from less than or equal to 100 psig to 100 psig ± 20 psig. We agree that the trip level settings in the technical specifications should have upper and lower bounds; however, we did not find that the licensee adequately supported the increased span his proposal represented. We discussed our differences and mutually agreed to use the bounds on the pressure trip level settings as originally established, by analysis, in the FSAR and as given in Table 7.4-2 of the FSAR. The agreed upon settings are: RHR (LPCI) pump discharge pressure 50 psig ± 9 psig Core Spray pump discharge pressure 100 psig ± 10 psig. We conclude the above changes are acceptable as they make the TS trip level settings consistent with the FSAR analysis which has previously been reviewed and accepted.

(4) The following three change requests relate to making these three items consistent with the NRC Standard Technical Specifications.

Table 6.5-1, p. 275. For both the Site Operations Review Committee and the Safety Review and Audit Board the licensee desires to shift the duty of appointing alternate Committee or Board members from the permanent members to the Committee or Board Chairmen. This change is acceptable as it agrees with the NRC position given in the Standard Technical Specifications.

Table 6.5-1, p. 276. The licensee desires to define a quorum for the Safety Review and Audit Board as consisting of the Chairman or alternate and a majority of the members of the Board or alternates. This definition is acceptable as it agrees with the current NRC position given in the Standard Technical Specifications.

Figure 6.5-1, p. 284. The licensee desires to have the Site Operations and Review Committee (SORC) report to the General Superintendent Nuclear Generation; previously the SORC had only an advisory organizational tie between the Safety Review and Audit Board and the PASNY General Manager and Chief Engineer. The advisory ties are retained. This change is acceptable as it reflects the current organizational role of the SORC and reflects the current NRC position as given in the Standard Technical Specifications.

We conclude that the above group of three changes in the Administrative Controls section of the Technical Specifications are acceptable as they do not involve any significant hazards considerations.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the change does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the change does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date:
October 24, 1975

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-333

POWER AUTHORITY OF THE STATE OF NEW YORK

AND

NIAGARA MOHAWK POWER CORPORATION

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Notice is hereby given that the U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 3 to Facility Operating License No. DPR-59 issued to the Power Authority of the State of New York and Niagara Mohawk Power Corporation, located in Scriba, Oswego County, New York. The amendment is effective as of its date of issuance.

This amendment authorizes 25 miscellaneous Appendix A and two Appendix B Technical Specification changes. The Appendix A changes relate to the correction of typographical errors, clarification of the specifications, and make the specifications consistent with the Final Safety Analysis Report (FSAR). The Appendix B changes establish consistency in the Site Radiological Environmental Monitoring Program for the two plants on the site (Facility License No. DPR-59 and License No. DPR-63), and achieve consistency with Appendix A in the administrative controls of the Technical Specifications.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment is not required since the amendment does not involve a significant hazards consideration.

For further details with respect to this action, see (1) the application for amendment dated March 13, 1975, (2) Amendment No. 3 to License No. DPR-59, with Change No. 3 and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, NW., Washington, D.C. and at the Oswego City Library, 120 East Second Street, Oswego, New York.

A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this 24th, October 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert W. Reid

Robert W. Reid , Chief
Operating Reactors Branch #4
Division of Reactor Licensing

TECHNICAL SPECIFICATIONS

1.0 DEFINITIONS

The succeeding frequently used terms are explicitly defined so that a uniform interpretation of the specifications may be achieved.

A. Abnormal Occurrence - An abnormal occurrence is the occurrence of any of the following conditions:

1. A limiting safety system setting less conservative than the limiting setting established in Section 2 of these Technical Specifications.

2. Violation of a limiting condition for operation as established in Section 3 of these Technical Specifications.

3. An uncontrolled or unplanned release of radioactive material from any plant system designed to act as a boundary for such material in an amount of significance with respect to limits prescribed in technical specifications.

4. Failure of one or more components of an engineered safety feature or a plant protection system which causes or threatens to cause the feature or system to be incapable of performing its intended safety function.

5. Abnormal degradation of one of the several boundaries designed to contain the radioactive materials resulting from the fission process.

6. Uncontrolled or unanticipated changes in reactivity greater than 1%ΔK.

7. Observed inadequancies in the implementation of administrative or procedural controls such that the inadequacy causes or threatens to cause the existence or development of an unsafe condition in connection with the operation of the Plant.

8. Conditions arising from natural or man-made events that affect or threaten to affect the safe operation of the Plant.

B. Core Alteration - The act of moving any component in the region above the core support plate, below the upper grid and within the shroud. Normal control rod movement with the control rod drive hydraulic system is not defined as a core alteration. Normal movement of in-core instrumentation is not defined as a core alteration.

C. Cold Condition - Reactor coolant temperature $\leq 212^\circ F$.

D. Hot Standby Condition - Hot Standby condition means operation with coolant temperature $>212^\circ$ F, the Mode Switch in Startup/Hot Standby and reactor pressure <1,005 psig. 3

E. Immediate - Immediate means that the required action will be initiated as soon as practicable considering the safe operation of the unit and the importance of the required action.

F. Instrumentation

1. Functional Test - A functional test is the manual operation or initiation of a system, subsystem, or component to verify that it functions within design tolerances (e.g., the manual start of a core spray pump to verify that it runs and that it pumps the required volume of water).

2. Instrument Channel Calibration- An instrument channel calibration means the adjustment of an instrument signal output so that it corresponds, within acceptable range, and accuracy, to a known value(s) of the parameter which the instrument monitors. Calibration shall encompass the entire instrument channel including actuation, alarm or trip.

3. Instrument Channel - An instrument channel means an arrangement of a sensor and auxiliary equipment required to generate and transmit to a trip system a single trip signal related to the plant parameter monitored by that instrument channel.

4. Instrument Check - An instrument check is qualitative determination of acceptable operability by observation of instrument behavior during operation. This determination shall include, where possible, comparison of the instrument with other independent instruments measuring the same variable.

5. Instrument Channel Functional Test - An instrument channel functional test means the injection of a simulated signal into the instrument primary sensor where possible to verify the proper instrument channel response, alarm and/or initiating action.

6. Logic System Function Test - A logic system functional test means a test of relays and contacts of a logic circuit from sensor to activated device to ensure components are operable per design intent. Where practicable, action will go to completion; i.e., pumps

1.2 and 2.2 BASES

The reactor coolant pressure boundary integrity is an important barrier in the prevention of uncontrolled release of fission products. It is essential that the integrity of this boundary be protected by establishing a pressure limit to be observed for all operating conditions and whenever there is irradiated fuel in the reactor vessel.

The pressure safety limit of 1,325 psig as measured by the vessel steam space pressure indicator is equivalent to 1,375 psig at the lowest elevation of the Reactor Coolant System. The 1,375 psig value is derived from the design pressures of the reactor pressure vessel and reactor coolant system piping. The respective design pressures are 1250 psig at 575° F for the reactor vessel, 1148 psig at 568°F for the recirculation suction piping and 1274 psig at 575° F for the discharge piping. The pressure safety limit was chosen as the lower of the pressure transients permitted by the applicable design codes: 1965 ASME Boiler and Pressure Vessel Code, Section III for the pressure vessel and 1969 ANSI B31.1 Code for the reactor coolant system piping. The ASME Boiler and Pressure Vessel Code permits pressure transients up to 10 percent over design pressure (110% X 1,250 = 1,375 psig), and the ANSI Code permits pressure transients up to 20 percent over the design pressure (120% X 1,150 = 1,380 psig). The safety limit pressure of 1,375 psig is referenced to the lowest elevation of the Reactor Coolant System.

The analyses in FSAR Section 14.5.1 state that the turbine trip from high power without bypass is the most severe abnormal operational transient resulting directly in a reactor coolant system pressure increase. The reactor vessel pressure code limit of 1,375 psig, given in FSAR Section 4.2, is well above the peak pressure produced by the worst overpressure transient above. Thus, the pressure safety limit is well above the peak pressure that can result from reasonably expected (1,375 psig) overpressure transients. Figure 4.4-4 of the FSAR presents the curve produced by this analysis. Reactor pressure is continuously indicated in the control room during operation.

A safety limit is applied to the Residual Heat Removal System (RHRS) when it is operating in the shutdown cooling mode. When operating in the shutdown cooling mode, the RHRS is included in the reactor coolant system.

3.1 LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the instrumentation and associated devices which initiate the reactor scram.

Objective:

To assure the operability of the Reactor Protection System.

Specification:

A. The setpoints, minimum number of trip systems, minimum number of instrument channels that must be operable for each position of the reactor mode switch shall be as shown on Table 3.1-1. The design system response time from the opening of the sensor contact to and including the opening of the trip actuator contacts shall not exceed 100 msec.

4.1 SURVEILLANCE REQUIREMENTS

4.1 REACTOR PROTECTION SYSTEM

Applicability:

Applies to the surveillance of the instrumentation and associated devices which initiate reactor scram.

Objective:

To specify the type and frequency of surveillance to be applied to the protection instrumentation.

Specification:

A. Instrumentation systems shall be functionally tested and calibrated as indicated in Tables 4.1-1 and 4.1-2 respectively.

B. Daily, during reactor power operation, while in the RUN MODE the peak heat flux and peaking factor shall be checked and the SCRAM and APRM Rod Block settings given by equations in Specifications 2.1.A.1 and 2.1.B shall be calculated if the peaking factor exceeds 2.6. 3

C. During reactor power operation with TPF≥2.43, MCHFR shall be calculated at least daily and following any change in power level or channel. Bypassing both channels for simultaneous testing should be avoided.

The most likely case would be to stipulate that one channel be bypassed, tested, and restored, and then immediately following, the second channel be bypassed, tested, and restored. This is shown by Curve No. 4. Note that there is no true minimum. The curve does have a definite knee and very little reduction in system unavailability is achieved by testing at a shorter interval than computed by the equation for a single channel.

The best test procedure of all those examined is to perfectly stagger the tests. That is, if the test interval is four months, test one or the other channel every two months. This is shown in Curve No. 5. The difference between Cases 4 and 5 is negligible. There may be other arguments, however, that more strongly support the perfectly staggered tests, including reductions in human error.

The conclusions to be drawn are these:

1. A 1 out of n system may be treated the same as a single channel in terms of choosing a test interval; and

2. More than one channel should not be bypassed for testing at any one time.

The radiation monitors in the refueling area ventilation duct which initiate building isolation and standby gas treatment operation are arranged in a 1 out of 2 logic system. The bases given above for the rod blocks apply here also and were used to arrive at the functional testing frequency. The air ejector offgas monitors are connected in a 2 out of 2 logic arrangement. Based on experience with instruments of similar design, a testing interval of once every three months has been found adequate.

The automatic pressure relief instrumentation can be considered to be a 1 out of 2 logic system and the discussion above applies also.

The bases for the radiation monitors are contained in their pertinent sections of the Environmental Technical Specifications.

TABLE 3.2-1

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

Minimum Number of Operable Instrument Channels per Trip System (1)	Instrument	Trip Level Setting	Total Number of Instrument Channels Provided By Design for Both Channels	Action (2)
2 (6)	Reactor Low Water Level	≥ 12.5 Indicated Level (3)	4 Inst. Channels	A
1	Reactor High Pressure (Shutdown Cooling Isolation)	≤ 75 psig	2 Inst. Channels	D
2	Reactor Low-Low Water Level	≥-38 in. indicated level (4)	4 Inst. Channels	A
2 (6)	High Drywell Pressure	≤ 2 psig	4 Inst. Channels	A
2	High Radiation Main Steam Line Tunnel	≤ 3 X Normal Rated Full Power Background	4 Inst. Channels	B
2	Low Pressure Main Steam Line	≥ 850 psig (7)	4 Inst. Channels	B
2	High Flow Main Steam Line	≤ 140% of Rated Steam Flow	4 Inst. Channels	B
2	Main Steam Line Tunnel Exhaust Duct High Temperature	≤ 40° F above max ambient	4 Inst. Channels	B
2	Main Steam Line Leak Detection High Temperature	≤ 40° F above max ambient	4 Inst. Channels	B
3	Reactor Cleanup System Equipment Area High Temperature	≤ 40° F above max ambient	6 Inst. Channels	C
1	Reactor Cleanup System High Temperature	≤ 140° F	1 Inst. Channel	C
2	Low Condenser Vacuum closes MSIV's 3	≥ 8" Hg. vac. (8)	4 Inst. Channels	B 3

TABLE 3.2-1 (Cont'd)

INSTRUMENTATION THAT INITIATES PRIMARY CONTAINMENT ISOLATION

NOTES FOR TABLE 3.2-1

1. Whenever Primary Containment integrity is required by Section 3.7, there shall be two operable or tripped trip systems for each function.

2. From and after the time it is found that the first column cannot be met for one of the trip systems, that trip system shall be tripped or the appropriate action listed below shall be taken.

   A. Initiate an orderly shutdown and have the reactor in cold shutdown condition in 24 hours.

   B. Initiate an orderly load reduction and have main steam lines isolated within eight hours.

   C. Isolate Reactor Water Cleanup System.

   D. Isolate shutdown cooling.

3. Instrument set point corresponds to 177 in. above top of active fuel.

4. Instrument set point corresponds to 126.5 in. above top of active fuel.

5. Two required for each steam line.

6. These signals also start SBGTS and initiate secondary containment isolation.

7. Only required in run mode (interlocked with Mode Switch).

8. Bypassed when reactor pressure is less than 600 psig and turbine stop valves are closed. 3

TABLE 3.2-2

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Total Number of Instrument Channels Provided by Design for Both Channels	Remarks
2	Reactor Low-Low Water Level	≥-38 in. indicated level	4 HPCI & RCIC Inst. Channels	Initiates HPCI, RCIC & SGTS.
2	Reactor Low-Low-Low Water Level	≥-146.5 in. indicated level (4)	4 Core Spray & RHR Instrument Channels 4 ADS Instrument Channels	1. In conjunction with Low Reactor Pressure initiates Core Spray and LPCI. 2. In conjunction with confirmatory low level High Drywell Pressure, 120 second time delay and LPCI or Core Spray pump interlock initiates Auto Blowdown (ADS). 3. Initiates starting of Diesel Generator.
2	Reactor High Water Level	≤ + 58 in. indicated level	2 Inst. Channels	Trips HPCI and RCIC turbines.
1	Reactor Low Level (inside shroud)	≥ +352 in. above vessel zero	2 Inst. Channels	Prevents inadvertent operation of containment spray during accident condition.
2	Containment High Pressure	1 < P < 2 psig	4 Inst. Channels	Prevents inadvertent operation of containment spray during accident condition.
1	Confirmatory Low Level	≥ 12.5 in. indicated level	2 Inst. Channels	ADS Permissive.

TABLE 3.2-2 (Cont'd)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Total Number of Instrument Channels Provided by Design for Both Channels	Remarks
2	High Drywell Pressure	≤ 2 psig	4 HPCI Inst. Channels 4 RHR & Core Spray Inst. Channels	1. Initiates Core Spray LPCI; HPCI & SGTS. 2. Initiates starting of Diesel Generators
2	Reactor Low Pressure	≥ 450 psig	4 Inst. Channels	Permissive for opening Core Spray and LPCI Admission valves. Coincident with high drywell pressure, starts LPCI and Core Spray pumps.
1	Reactor Low Pressure	50 ≤ P ≤ 75 psig	2 Inst. Channels	In conjunction with PCIS signal permits closure of RHR (LPCI) injection valves.
2	Reactor Low Pressure	≥ 900 psig	4 Inst. Channels	Prevents actuation of LPCI break detection circuit.
2	High Drywell Pressure	≤ 2 psig	4 Inst. Channels	1. In conjunction with Low-Low Reactor Water Level, 120 second time delay and LPCI or Core Spray pump running, initiates Auto Blowdown (ADS).
1	Core Spray Pump Start Timer	11 ± 0.6 sec	2 Timers 2 Timers	Initates starting of core spray and RHR pumps.
1	RHR Pump Start Timer Pump #1 Pump #2	 1.0 + 0.5, -0 sec 6.0 ± 0.5 sec	 2 Timers 2 Timers	

TABLE 3.2-2 (Cont'd)

INSTRUMENTATION THAT INITIATES OR CONTROLS THE CORE AND CONTAINMENT COOLING SYSTEMS

Minimum No. of Operable Instrument Channels Per Trip System (1)	Trip Function	Trip Level Setting	Total Number of Instrument Channels Provided by Design for Both Channels	Remarks	
1	Auto Blowdown Timer	120 sec ± 5 sec	2 Timers	In conjunction with Low Reactor Water Level High Drywell Pressure and LPCI or Core Spray Pump running interlock, initiates Auto Blowdown.	
2	RHR (LPCI) Pump Discharge Pressure Interlock	50 psig $\pm$ 9 psig	4 Channels	Defers ADS actuation pending confirmation of low pressure core cooling system operation.	3
2	Core Spray Pump Discharge Pressure Interlock	100 psig $\pm$ 10 psig	4 Channels	LPCI or Core Spray Pump running interlock.	3
1	RHR (LPCI) Trip System bus power monitor	≥12.5 volts d-c	2 Inst. Channels	Monitors availability of power to logic systems.	
1	Core Spray Trip System bus power monitor	≥12.5 volts d-c	2 Inst. Channels	Monitors availability of power to logic systems.	
1	ADS Trip System bus power monitor	≥12.5 volts d-c	2 Inst. Channels	Monitors availability of power to logic systems.	
1	HPCI Trip System bus power monitor	≥12.5 volts d-c	2 Inst. Channels	Monitors availability of power to logic systems.	
1	RCIC Trip System bus power monitor	≥12.5 volts d-c	2 Inst. Channels	Monitors availability of power to logic systems.	

TABLE 4.2-2

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CORE AND CONTAINMENT COOLING SYSTEMS

Instrument Channel	Instrument Functional Test	Calibration Frequency	Instrument Check
1) Reactor Water Level	(1)	Once/3 months	Once/day
2) Drywell Pressure	(1)	Once/3 months	None
3) Reactor Pressure	(1)	Once/3 months	None
4) Auto Sequencing Timers	NA	Once/operating cycle	None
5) ADS - LPCI or CS Pump Disch. Pressure Interlock	(1)	Once/3 months	None
6) Trip System Bus Power Monitors	(1)	Once/operating cycle	None
7) Recirculation System d/p	(1)	Once/3 months	Once/day
8) Core Spray Sparger d/p	(1)	Once/6 months	Once/day
9) Steam Line High Flow (HPCI & RCIC)	(1)	Once/3 months	None
10) Steam Line High Temp. (HPCI & RCIC)	(1)	Once/operating cycle	Once/day
11) Safeguards Area High Temp.	(1)	Once/operating cycle	None
12) HPCI and RCIC Steam Line Low Pressure	(1)	Once/3 months	None
13) HPCI Suction Source Levels	(1)	Once/3 months	None
14) 4KV Emergency Power System Voltage Relays	Once/operating cycle	Once/5 years	None
15) HPCI and RCIC Exhaust Pressure High	(1)	Once/3 months	None
16) HPCI and RCIC Low Pump Suction Pressure	(1)	Once/3 months	None

NOTE: See listing of notes following Table 4.2-6 for the notes referred to herein.

TABLE 4.2-2 (CONT'D)

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CORE AND CONTAINMENT COOLING SYSTEMS

Logic System Functional Test		Frequency	
1) Core Spray Subsystem	(4)(6)	Once/6 months	
2) Low Pressure Coolant Injection Subsystem	(4)(6)	Once/6 months	3
3) Containment Cooling Subsystem	(6)	Once/6 months	3
4) HPCI Subsystem	(4)(6)	Once/6 months	
5) HPCI Subsystem Auto Isolation	(4)(6)	Once/6 months	
6) ADS Subsystem	(4)(6)	Once/6 months	
7) RCIC Subsystem Auto Isolation	(4)(6)	Once/6 months	3
8) ADS Relief Valve Bellow Pressure Switch	(4)(6)	Once/operating cycle	

NOTE: See listing of notes following Table 4.2-6 for the notes referred to herein.

TABLE 4.2-3

MINIMUM TEST AND CALIBRATION FREQUENCY FOR CONTROL ROD BLOCKS ACTUATION

Instrument Channel	Instrument Functional Test	Calibration	Instrument Check (9)
1) APRM - Downscale	(1) (3)	Once/3 months	Once/day
2) APRM - Upscale	(1) (3)	Once/3 months	Once/day
3) IRM - Upscale	(2) (3)	(2)	(2)
4) IRM - Downscale	(2) (3)	(2)	(2)
5) RBM - Upscale	(1) (3)	Once/6 months	Once/day
6) RBM - Downscale	(1) (3)	Once/6 months	Once/day
7) SRM - Upscale	(2) (3)	(2)	(2)
8) SRM - Detector Not in Startup Position	(2) (3)	(2)	
9) IRM - Detector Not in Startup Position	(2) (3)	(2)	

3

3

Logic System Functional Test (4) (6)	Frequency
1) System Logic Check	Once/6 months

NOTE: See listing of notes following Table 4.2-6 for the notes referred to herein.

TABLE 4.2-4

MINIMUM TEST AND CALIBRATION FREQUENCY FOR RADIATION MONITORING SYSTEMS

	Instrument Channels	Instrument Functional Test	Calibration	Instrument Check (2) (9)	
1)	Refuel Area Exhaust Monitors	(1)	Once/3 months	Once/day	3
2)	Reactor Building Area Exhaust Monitors	(1)	Once/3 months	Once/day	
	Turbine Building Exhaust Monitors	(1)	Once/6 months	Once/day	
	Radwaste Building Exhaust Monitors	(1)	Once/6 months	Once/day	
3)	Off-Gas Radiation Monitors	(1)	Once/3 months	Once/day	
4)	Main Control Room Ventilation Monitor	(1)	Once/3 months	Once/day	
5)	Mechanical Vacuum Pump Isolation	SEE TABLES 4.1-2 & 4.1-2			
6)	Liquid Radwaste Discharge Monitor	(1)	Once/3 months	Once/day When discharging	

	Logic System Functional Test (4) (6)	Frequency	
1)	Reactor Building Isolation	Once/6 months	
2)	Standby Gas Treatment System Actuation	Once/6 months	
3)	Steam Jet Air Ejector Off-Gas Line Isolation	Once/6 months	
4)	Mechanical Vacuum Pump Isolation	Once/Operating Cycle	3
5)	Liquid Radwaste Discharge Isolation	Once/6 months	

NOTE: See listing of notes following Table 4.2-6 for the notes referred to herein.

NOTES FOR TABLES 4.2-1 THROUGH 4.2-6

1. Initially once every month until acceptable failure rate data are available; thereafter, a request may be made to the AEC to change the test frequency. The compilation of instrument failure rate data may include data obtained from other boiling water reactors for which the same design instrument operates in an environment similar to that of JAFNPP.

2. Functional tests, calibrations and instrument checks are not required when these instruments are not required to be operable or are tripped. Functional tests shall be performed before each startup with a required frequency not to exceed once per week. Calibrations shall be performed prior to each startup or prior to preplanned shutdowns with a required frequency not to exceed once per week. Instrument checks shall be performed at least once per day during these periods when the instruments are required to be operable.

3. This instrumentation is excepted from the functional test definition. The functional test will consist of injecting a simulated electrical signal into the measurement channel.

   These instrument channels will be calibrated using simulated electrical signals once every three months.

4. Simulated automatic actuation shall be performed once each operating cycle. Where possible, all logic system functional tests will be performed using the test jacks.

5. Reactor low water level, high drywell pressure and high radiation main steam line tunnel are not included on Table 4.2-1 since they are tested on Table 4.1-2.

6. The logic system functional tests shall include a calibration of time delay relays and timers necessary for proper functioning of the trip systems.

7. At least one (1) Main Stack Dilution Fan is required to be in operation in order to isokineticaly sample the Main Stack.

8. Uses same instrumentation as Main Steam Line High Radiation. See Table 4.1-2.

9. See Technical Specification 1.0.F.4, Definitions, for meaning of term "Instrument Check". 3

TABLE 4.2-7

MINIMUM TEST AND CALIBRATION FREQUENCY FOR RECIRCULATION PUMP TRIP

Instrument Channel	Instrument Functional Check	Calibration Frequency
1) Reactor High Pressure	Once/operating cycle	Once/operating cycle
2) Reactor Low Water Level	Once/operating cycle	Once/operating cycle

Logic System Functional Test	Frequency
1) Recirculation Pump Trip	Once/refueling cycle

B. Control Rods

1. Each control rods shall be coupled to its drive or completely inserted and the control rod directional control valves disarmed electrically. This requirement does not apply in the refuel condition when the reactor is vented. Two control rod drives may be removed as long as Specification 3.3.A.1 is met.

2. The control rod drive housing support system shall be in place during reactor power operation or

B. Control Rods

1. The coupling integrity shall be verified for each withdrawn control rod as follows:

   a. When a rod is withdrawn the first time after each refueling outage or after maintenance, observe discernible response of the nuclear instrumentation. However, for initial rods when response is not discernible, subsequent exercising of these rods after the reactor is above 20 percent power shall be performed to verify instrumentation response.

   b. When the rod is fully withdrawn the first time after each refueling outage or after maintenance, observe that the drive does not go to the overtravel position.

   c. During each refueling outage and after each control rod maintenance, observe that the drive does not go to the overtravel position.

2. The control rod drive housing support system shall be inspected after reassembly and the results of the inspection recorded.

when the reactor coolant system is pressurized above atmospheric pressure with fuel in the reactor vessel, unless all control rods are fully inserted and Specification 3.3.A.1 is met.

3. a. Whenever the reactor is in the startup or run modes below 20 percent rated power the Rod Sequence Control System shall be operable.

   b. Whenever the reactor is in the startup or run modes below 20 percent rated power the Rod Worth Minimizer shall be operable or a second licensed operator

withdrawal, fully withdrawn, and all other control rods capable of insertion fully inserted if Specification 4.3.A.1

3. Prior to the start of control rod withdrawal at startup, and prior to attaining 20 percent rated power during rod insertion at shutdown, the capability of the Rod Sequence Control System and the Rod Worth Minimizer t› properly fulfill their functions shall be verified by the following checks:

   a. The capability of the RSCS to properly fulfill its function shall be verified by the following tests:

   Sequence portion - Select a sequence and attempt to withdraw a rod in the remaining sequences. Move one rod in a sequence and select the remaining sequences and attempt to move a rod in each. Repeat for all sequences. Prior to reaching 20% power shutdown: Sequence portion - Select a sequence and attempt to select a rod in the remaining sequences.

   Group notch portion - For each of the six comparator circuits go through test initiate; comparator inhibit; verify; reset. On seventh attempt test is allowed to continue until completion is indicated by illumination of test complete light.

   b. The capability of the Rod Worth Minimizer (RWM) shall be verified by the following checks:

b. From the time that the LPCI mode is made or found to be inoperable for any reason, continued reactor operation is permissible during the succeeding 7 days unless the LPCI mode is made operable earlier provided that during these 7 days all active components of both Core Spray Systems, the containment spray sub-system (including two RHR pumps) and the emergency diesel generators shall be operable.

4. The reactor shall not be started up with the RHR System supplying cooling to the fuel pool.

5. If the requirements of 3.5.A cannot be met, the reactor shall be placed in the cold condition within 24 hr.

B. <u>Containment Cooling Subsystem Mode (of the RHR System)</u>

1. Both subsystems of the containment cooling mode, each including two RHR, one ESW pump and two RHRSW pumps shall be operable whenever there is irradiated fuel in the reactor

b. When it is determined that the LPCI mode is inoperable, both Core Spray Systems, the containment spray subsystem, and the emergency diesel generators shall be demonstrated to be operable immediately and daily thereafter.

B. <u>Containment Cooling Subsystem Mode (of the RHR System)</u>

1. Subsystems of the containment cooling mode are tested in conjunction with the tests performed on the LPCI System and given in 4.5.A.1.a, b, c, and d. Residual heat removal

vessel, prior to startup from a cold condition, and reactor coolant temperature ≥212°F, except as specified below:

2. Continued reactor operation is permissible for 30 days with one spray loop inoperable and with reactor water temperature greater than 212°F.

3. Should one RHR pump and/or one RHRSW pump of the components required in 3.5.B.1 above be made or found inoperable, continued reactor operation is permissible only during the succeeding 30 days provided that during such 30 days all remaining active components of the containment cooling mode are operable.

service water pumps, each loop consisting of two pumps operating in parallel, will be included in testing, supplying 8,000 gpm. The Emergency Service Water System, each loop of which consists of a single operating emergency service water pump of 3,700 gpm will be tested in accordance with Section 4.11D. 3

During each five-year period an air test shall be performed on the containment spray headers and nozzles.

2. When it is determined that one RHR pump and/or one RHRSW pump of the components required in 3.5.B.1 above are inoperable, the remaining redundant active components of the containment cooling mode subsystems shall be demonstrated to be operable immediately and daily thereafter.

3. When one containment cooling subsystem loop becomes inoperable, the operable loop including its associated diesel generator shall be demonstrated to be operable immediately and daily thereafter.

condition, that pump shall be considered inoperable for purposes satisfying Specifications 3.5.A, 3.5.C, and 3.5.E.

H. The average LHGR at any axial cross section of any fuel bundle in the core (Maximum Average Planar Linear Heat Generation Rate, MAPLHGR) shall not exceed the operating level (Peak MAPlHGR) shown in Figure 3.5-1 (9.5.Q curve from topical report) for fuel types I, II, III, etc.

2. Following any period where the LPCI subsystems or core spray subsystems have not been required to be operable, the discharge piping of the inoperable system shall be vented from the high point prior to the return of the system to service.

3. Whenever the HPCI, RCIC, or Core Spray System is lined up to take suction from the condensate storage tank, the discharge piping of the HPCI, RCIC, and Core Spray shall be vented from the high point of the system, and water flow observed on a monthly basis.

4. The pressure switches which monitor the Core Spray and LPCI discharge lines to ensure that they are full shall be functionally tested every month and calibrated every three months.

H. Daily during reactor power operation, while in the RUN MODE, the MAPLHGR shall be checked against Figure 3.5-1 and adjusted if required. 3

Each fuel type and exposure will be compared to the appropriate curve as shown in Figure 3.5-1.

3.5 (cont'd)

I. During steady state power operation, the LHGR of any rod in any fuel assembly at any axial location shall not exceed the maximum allowable LHGR as calculated by the following relations:

$$LHGR_m = LHGR_L \left[1.0 - \left(\frac{\Delta P}{P} \right) max \; \frac{L}{L_T} \right]$$

$LHGR_m$ = Maximum local linear heat generation rate, key

$LHGR_L$ = Local linear heat generation rate, license limit

L = Axial position from bottom of core

L_T = Total core length

$(\Delta P/P)max$ = Maximum value of power spiking penalty

4.5 (cont'd)

I. Daily during power operation, while in the RUN MODE, the maximum local LHGR shall be determined and adjusted if required. 3

3.7 LIMITING CONDITIONS FOR OPERATION

3.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the operating status of the primary and secondary containment systems.

Objective:

To assure the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment

1. The volume and temperature of the water in the pressure suppression chamber shall at all times, except as specified in Specification 3.5.F.2, be 3
maintained within the following limits:

    a. Maximum water temperature 95°F during normal power operation.

    b. Minimum water volume 105,600 ft^3 corresponding to a vent submergence level of 4 ft 2 in.

    c. Maximum water volume 110,100 ft^3 corresponding to a vent submergence level of 56 in.

4.7 SURVEILLANCE REQUIREMENTS

4.7 CONTAINMENT SYSTEMS

Applicability:

Applies to the primary and secondary containment integrity.

Objective:

To verify the integrity of the primary and secondary containment systems.

Specification:

A. Primary Containment

1. The pressure suppression chamber water level and temperature shall be checked once per day. The accessible interior surfaces of the drywell and above the water line of the pressure suppression chamber shall be inspected at each refueling outage for evidence of deterioration.

d. Maximum water temperature during RCIC, HPCI, or relief valve operation--130°F.

e. In order to continue reactor power operation following RCIC, HPCI, or relief valve operation, the suppression chamber temperature must be reduced to 95°F or less within 24 hr. following resumption of power operation.

2. Primary containment integrity shall be maintained at all times when the reactor is critical or when the reactor water temperature is above 212°F, and fuel is in the reactor vessel, except while performing low power physics tests at atmospheric pressure at power levels not to exceed 5 MWt.

2. The primary containment integrity shall be demonstrated as follows:

a. Type A Test (primary Containment Integrated Leakage Rate Test)

(1.) Containment inspection shall be performed as a prerequisite to the performance of Type A tests. During the period between the initiation of the containment inspection and the performance of the Type A test, no repairs or adjustments shall be made.

e. Leakage between the drywell and suppression chamber shall not exceed a rate of 71 scfm as monitored via the suppression chamber 10 min pressure transient of 0.25 in. water/min.

f. The self actuated vacuum breakers shall open when subjected to a force equivalent to 0.5 psid acting on the valve disc.

g. From and after the date that one of the pressure suppression chamberreactor building vacuum breakers is made or found to be inoperable for any reason, the vacuum breaker shall be locked closed and reactor operation is permissible only during the succeeding seven days unless such vacuum breaker is sooner made operable, provided that the repair procedure does not violate primary containment integrity.

e. Not applicable

f. Not applicable

g. During each refueling outage each vacuum breaker shall be tested to determine that the force required to open the vacuum breaker does not exceed the force specified in Specification 3.7.A5.f. 3
and each vacuum breaker shall be inspected and verified to meet design requirements.

6. Oxygen Concentration

a. After completion of the startup test program and demonstration of plant electrical output, the primary containment atmosphere shall be reduced to less than four percent oxygen with nitrogen gas during reactor power operation with reactor coolant pressure above 100 psig, except as specified in 3.7.A.6.b.

b. Within the 24 hr. period subsequent to placing the reactor in the run mode following a shutdown, the containment atmosphere oxygen concentration shall be reduced to less than 4 percent by weight and maintained in this condition. De-inerting may commence 24 hr. prior to a shutdown.

7. a. If the specifications of 3.7.A.1 through 3.7.A.6 cannot be met, an orderly shutdown shall be initiated, and the reactor shall be in a cold condition within 24 hr.

6. Oxygen Concentration

a. The primary containment oxygen concentration shall be measured and recorded at least twice weekly.

7. Not applicable.

TABLE 3.7-1 (Cont'd)

PROCESS PIPELINE PENETRATING PRIMARY CONTAINMENT
(Numbers in parentheses are keyed to numbers on following pages; signal codes are listed on following pages.)

Line Isolated	Drywell Penetration	Valve Type (6)	Power to Open (5) (6)	Group	Location Ref. to Drywell	Power to Close (5) (6)	Isolation Signal	Closing Time (7)	Normal Status	Remarks and Exceptions	
Mini-purge to recirc pump	X-31AC X-31BC	Check	Process	C	Outside	Process	Rev. flow	Not applicable	Open		
Mini-purge to recirc pump	X-31AC X-31BC	Check	Process	C	Inside	Process	Rev. flow	Not applicable	Open		
RHR Reactor shutdown cooling supply	X-12	MO Gate	Dc	A	Outside	Dc	A,U,F,RM	38 sec	Closed		
RHR Reactor shutdown cooling supply	X-12	MO Gate	Ac	A	Inside	Ac	A,U,F,RM	38 sec	Closed		
RHR to suppression spray header	X-211A,B	MO Globe	Ac	B	Outside	Ac	G,S,RM	10 sec	Closed	Throttling type valve Note (2)	
RHR - containment spray	X-39A,B	MO Gate	Ac	B	Outside	Ac	G,S,RM	10 sec	Closed	Note (2)	
RHR - containment spray	X-39A,B	MO Gate	Ac	B	Outside	Ac	G,S,RM	10 sec	Closed	Note (2)	
RHR - reactor head spray	X-17	MO Gate	Ac	A	Inside	Ac	A,U,F,RM	20 sec	Closed		
RHR - reactor head spray	X-17	MO Gate	Dc	A	Outside	Dc	A,U,F,RM	20 sec	Closed		
RHR to suppression pool	X-210A,B	MO Globe	Ac	B	Outside	Ac	G,RM	70 sec	Closed	Throttling type valve - Note (2)	
RHR - LPCI to reactor	X-13A,B	MO Gate	Ac	A	Outside	Ac	RM,H	120 sec	Closed		3
RHR - LPCI to reactor	X-13A,B	MO Globe	Ac	A	Outside	Ac	RM, H	90 sec	Open	Throttling type valve - Note (10)	3
RHR - LPCI to reactor	X-13A,B	AO Check		A	Inside	Process	Rev. flow	Not applicable	Closed	Testable check valve	

TABLE 3.7-1 (Cont'd)

PROCESS PIPELINE PENETRATING PRIMARY CONTAINMENT
(Numbers in parentheses are keyed to numbers on following pages; signal codes are listed on following pages.)

Line Isolated	Drywell Penetration	Valve Type (6)	Power to Open (5) (6)	Group	Location Ref. to Drywell	Power to Close (5) (6)	Isolation Signal	Closing Time (7)	Normal Status	Remarks and Exceptions
Air pump suction from suppression pool	X-225A,B	MO Gate	Ac	B	Outside	Ac	RM	Not applicable	Open	
Standby liquid control	X-42	Check	-	A	Outside	Process	Rev. flow	Not applicable	Closed	
Standby liquid control	X-42	Check	-	A	Inside	Process	Rev. flow	Not applicable	Closed	
Reactor water cleanup from reactor	X-14	MO Gate	Ac	A	Inside	Ac	A,J,RM	30 sec	Open	
Reactor water cleanup from reactor	X-14	MO Gate	Dc	A	Outside	Dc	A,V,J,RM	30 sec	Open	
Reactor water from reactor warm-up	X-14	MO Gate	Dc	A	Outside	Dc	A,V,Y,J,RM	10 sec	Closed	
Reactor water cleanup return	X-9A	Check	-	A	Outside	Process	Rev. flow	Not applicable	Open	
RCIC - turbine steam supply	X-10	MO Gate	Ac	A	Inside	Ac	K,RM	15 sec	Open)	Opens on Sig B; line break Sig K overrides to close valves
RCIC - turbine steam supply	X-10	MO Gate	Dc	A	Outside	Dc	K,RM	15 sec	Open)	
RCIC - turbine exhaust	X-212	Check	Fwd. flow	B	Outside	Process	Rev. flow	-	Closed	
RCIC - minimum pump flow	X-210A	MO Globe	Dc	B	Outside	Dc	K, RM	5 sec	Closed	
RCIC - Pump discharge	X-9A	AO Check	Fwd. flow	B	Outside	Process	Rev. flow (3)	Not applicable	Closed	
RCIC - pump discharge	X-9A	MO Gate	Dc	B	Outside	DC	RM	Not applicable	Closed	

6.0 ADMINISTRATIVE CONTROLS

Administrative controls are the means by which plant operations are subject to management control. Measures specified in this section provide for the assignment of responsibilities, plant organization, staffing qualifications and related requirements, review and audit mechanisms, procedural controls, and reporting requirements. Each of these measures are necessary to ensure safe and efficient facility operation and are applicable throughout plant life.

6.1 RESPONSIBILITY

A. The General Superintendent shall have direct responsibility for the safe operation of all generating units on the NMP-JAF site. During periods when the General Superintendent is unavailable, he may delegate his responsibilities to either the Station Superintendent Nine Mile Point Nuclear Station or the Plant Superintendent James A. FitzPatrick Nuclear Power Plant.

B. The portions of Niagara Mohawk management which relate to the operation of the plant are shown in Fig. 6.1-1.

6.2 PLANT STAFF ORGANIZATION

A. The plant staff organization shall be as shown in Fig. 6.1-2 and function as follows:

1. A senior licensed operator shall be onsite at all times when there is fuel in the reactor.

2. In addition to item 1 above, a licensed operator shall be in the control room at all times when there is fuel in the reactor.

3. In addition to items 1 and 2 above, a licensed operator shall be readily available onsite whenever the reactor is in other than cold condition.

4. Two licensed operators shall be in the control room during startups and scheduled shutdowns.

5. A senior licensed operator shall be responsible for all movement of new and irradiated fuel within the site boundary. A licensed operator will be required to manipulate the controls of all fuel moving equipment except the reactor building crane. All fuel movements by the reactor building crane except new fuel movements from receipt through dry storage shall be under the direct supervision of a licensed operator. All fuel moves within the core shall be directly monitored by a member of the reactor analyst group.

6. The shift crew shall, as a minimum, be composed of three persons, plus two additional persons whenever the unit is other than in the cold condition.

7. Operating personnel shall be qualified to implement radiation control procedures.

8. Due to illness or absenteeism up to two hours is allowed to restore shift crew to the normal complement.

6.3 PLANT STAFF QUALIFICATIONS

A. Minimum qualifications with regard to educational background and experience for plant staff positions shown in Fig. 6.1- 2 shall be as 3
follows:

1. General Superintendent Nuclear Generation/Plant Superintendent

The General Superintendent Nuclear Generation/Plant Superintendent shall have ten years of responsible power plant experience of which a minimum of three years shall be nuclear power plant experience. A maximum of four years of the remaining seven years of experience may be fulfilled by academic training on a one-for-one time basis. This academic training shall be in an engineering or scientific field generally associated with power production. The General Superintendent Nuclear Generation/Plant Superintendent shall have acquired the experience and training normally required for examination by the NRC for a Senior Operator License whether or not the examination is taken.

2. Operations Supervisor 3

The Operations Supervisor shall 3
have a minimum of eight years of responsible power plant experience of which a minimum of three years shall be nuclear power plant experience. A maximum of two years of the remaining five years of power plant experience may be fulfilled by satisfactory completion of academic or related technical training on a one-for-one time basis. He shall hold a Senior Operator License.

3. Results Supervisor

The Results Supervisor shall have a minimum of eight years in responsible positions, of which one year shall be nuclear power plant experience. A

1. Records and drawing changes reflecting plant design modifications made to systems and equipment described in the FSAR.

2. Records of new and spent fuel inventory, transfers of fuel, and assembly histories.

3. Records of plant radiation and contamination surveys.

4. Records of offsite environmental monitoring surveys.

5. Records of radiation exposure of all plant personnel, and others who enter radiation control areas.

6. Records of radioactivity in liquid and gaseous wastes released to the environment.

7. Records of transient or operational cycling for those plant components that have been designed to operate safely for a limited number of transients or operational cycles.

8. Records of individual plant staff members indicating qualifications, experience, training, and retraining, or until personnel are no longer employed in the plant.

9. Inservice inspections of the Reactor Coolant System.

10. Minutes of meetings of the Site Operations Review Committee and Safety Review and Audit Board.

TABLE 6.5-1

RESPONSIBILITIES AND AUTHORITY OF SAFETY REVIEW ORGANIZATION
JAMES A. FITZPATRICK NUCLEAR STATION

SITE OPERATIONS REVIEW COMMITTEE	SAFETY REVIEW AND AUDIT BOARD	GENERAL MANAGER/ CHIEF ENGINEER
Responsibilities:		
Review proposed normal, off-normal and emergency operating procedures; proposed changes thereto and any other proposed procedures or changes as determined by the General Superintendent Nuclear Generation to affect nuclear safety.	Review any abnormalities arising during plant operation. Advise General Manager and Chief Engineer, PASNY, V.P.-Eng. and V.P.-Electric Operations, NMPC.	Provide overall management guidance.
Review all proposed test and experiments.	Review proposed test and experiments whose performance may constitute an unreviewed safety question as defined in 10CFR50.59. Submit analysis to V.P. Electric Operations for formal submittal to NRC.	
Review all proposed changes to the Technical Specifications.	Review proposed Technical Specification changes or license amendments. Submit analysis to V.P.-Electric Oper. and V.P. Eng. NMPC. for formal submittal to NRC.	
Review all proposed changes or modifications to plant system.	Review proposed changes to equipment systems and procedures which may involve an unreviewed safety question as defined in 10 CFR 50.59 (c) or which are referred to the Board by the Site Operations Review Committee. a. Review nuclear safety matters deemed essential to the safe operation of the plant by Site Operations Review Committee. b. Review significant operating abnormalities or deviations from normal performance of plant equipment, as determined by the Site Operations Review Committee.	Approve all major changes. Formally submit to NRC all necessary reports and applications.

Review nuclear unit operations to detect any potential safety hazards.	The board shall direct and evaluate the results of periodic audits performed to ensure safe facility operation. These audits shall encompass: a. The conformance of facility operation to all provisions contained within the Tech. Spec. and applicable license requirements,	Review and approve results of periodic audits and consult with Vice President-Electric Operations.	3 3

TABLE 6.5-1 (CONT'D)

RESPONSIBILITIES AND AUTHORITY OF SAFETY REVIEW ORGANIZATION
JAMES A. FITZPATRICK NUCLEAR STATION

3

SITE OPERATIONS REVIEW COMMITTEE	SAFETY REVIEW AND AUDIT BOARD	GENERAL MANAGER/ CHIEF ENGINEER
	b. The performance of the entire facility staff relative to nuclear safety. c. The results of all actions taken to correct anomalies occurring in the facility, equipment, structures, systems or method of operation. d. The adequacy of the Quality Assurance Program to meet the criteria specified in 10CFR50, Appendix B. e. Any other area of facility operation considered appropriate by the Board or the Vice President-Electrical Operations, and the V.P.-Eng. NMPC	
Investigate all violations of applicable statutes, regulations, orders, license requirements, or internal procedures or instructions having safety significance on Plant operation. Prepare and forward a report covering their evaluation and recommendations to prevent re-occurrence, to General Supt. Nuclear Generation and the Chairman of Safety Review and Audit Board.	Review violations of applicable statutes, regulations, orders, license requirements, or internal procedures or instructions having safety significance on Plant operation. Submit safety analysis to V.P.-Electric Operations and V.P.-Eng. NMPC.	
Investigate all violations of the Tech. Specs. (including abnormal occurrences) and prepare and forward a report covering their evaluation and recommendations to prevent re-occurrence to the General Supt. Nuclear Generation and the Chairman of Safety Review and Audit Board.	Review violations of Tech. Specs. (including abnormal occurrences). Submit safety analysis to V.P.-Electric Operations and V.P.-Eng. for formal submittal to NRC.	To be advised for concurrence with formal submittal.
Perform special reviews and investigations and render reports thereon as requested by the Chairman of the Safety Review & Audit Board, and the General Superintendent Nuclear Generation.	Review reports and meeting minutes of Site Operations Review Committee.	

TABLE 6.5-1 (CONT'D)

RESPONSIBILITIES AND AUTHORITY OF SAFETY REVIEW ORGANIZATION
JAMES A. FITZPATRICK NUCLEAR STATION

	SITE OPERATIONS REVIEW COMMITTEE	SAFETY REVIEW AND AUDIT BOARD	GENERAL MANAGER/ CHIEF ENGINEER
	Cause periodic drills to be conducted on emergency procedures including evaluation (partial or complete) of the site and check adequacy of communications with offsite support groups.		
	Review the Plant Security Plan and implementing procedures and recommended changes to the Chairman of the Safety Review and Audit Board.	Review Facility Security Plan and implementing procedures.	
	Review the Emergency Plan and implementing procedures and recommend changes to the Chairman of the Safety Review and Audit Board.	Review Facility Emergency Plan and implementing procedures.	
		Review Environmental Monitoring Program.	
		Review reports submitted to the Atomic Energy Commission and Associated responses.	
Authority:	Advisory to the General Superintendent Nuclear Generation.	The Board shall report to and advise the Vice President-Electric Operations and Vice President-Engineering in writing on all matters related to nuclear safety.	
	Make tentative determinations as to whether or not proposals considered by the committee involve unreviewed safety questions. This determination subject to review by the Safety Review and Audit Board.		
Records:	Minutes shall be kept at the plant of all meetings of the SORC and copies shall be sent to the General Superintendent Nuclear Generation and to the Chairman of the Safety Review and Audit Board.	Written minutes of each meeting shall be prepared, formally approved, and promptly distributed to each Board member, the V.P.-Engineering, V.P.-Electric Operations, PASNY General Manager and Chief Engineer, and other members of management having responsibility in the areas reviewed. Permanent copies of these minutes shall	

TABLE 6.5-1 (CONT'D)

RESPONSIBILITIES AND AUTHORITY OF SAFETY REVIEW ORGANIZATION
JAMES A. FITZPATRICK NUCLEAR STATION

	SITE OPERATIONS REVIEW COMMITTEE	SAFETY REVIEW AND AUDIT BOARD	GENERAL MANAGER/ CHIEF ENGINEER	
		be retained as specified in Section 6.13. Written reports of each audit function performed including follow-up action and re-audits shall be prepared, approved and forwarded to the V.P. -Electric Operations and PASNY General Manager and Chief Engineer and to the other management members having responsibility in the areas audited. Copies of these reports shall be retained as specified in Section 6.13.		31
Procedures:	Written administrative procedures shall be prepared and maintained that describe: a. The method of submission and the content for presentations to the committee. b. Provisions for the use of subcommittees. c. The method for review and approval of written committee evaluations and recommendations. d. The distribution of minutes; and e. Such other matters as may be appropriate.	The board shall be constituted by a written charter stating: a. Subjects within purview of the board. b. Responsibility and authority. c. Mechanisms for convening meetings. d. Provisions for use of subgroups. e. Authority for access to unit records f. Reporting requirements.		
Membership:	See Figure 6.5-1	See Figure 6.5-1		
	Alternates: Alternate members shall be appointed by the Chairman on a temporary basis to provide expertise in their respective discipline; however, no more than two (2) alternate members shall serve on the committee at any one time.	Alternates: a. Alternate members shall be appointed by the Chairman on a temporary basis. Board records shall be maintained showing each current designation. b. No more than two alternates shall serve on board at any one time.		3
	Consultants: Additional personnel with expertise in specific areas may serve as consultants to the Site Operations Review Committee.	When the nature of a particular situation dictates, special consultants shall be utilized to provide expert advice to		

TABLE 6.5-1 (CONT'D)

RESPONSIBILITIES AND AUTHORITY OF SAFETY REVIEW ORGANIZATION
JAMES A. FITZPATRICK NUCLEAR STATION

	SITE OPERATIONS REVIEW COMMITTEE	SAFETY REVIEW AND AUDIT BOARD	GENERAL MANAGER/ CHIEF ENGINEER
		Board members upon request of any two permanent members.	
Meeting Frequency:	Monthly, and as required on call of the Chairman.	Meetings shall be convened no less frequently than quarterly, the first year of operation; semi-annually, thereafter.	As required.
Quorum:	Chairman and four members including designated alternates.	The Chairman and a majority of SR&AB Members or alternates shall constitute a quorum. No more than a minority of the quorum shall have direct line responsibility for plant operation, nor be alternates.	

3

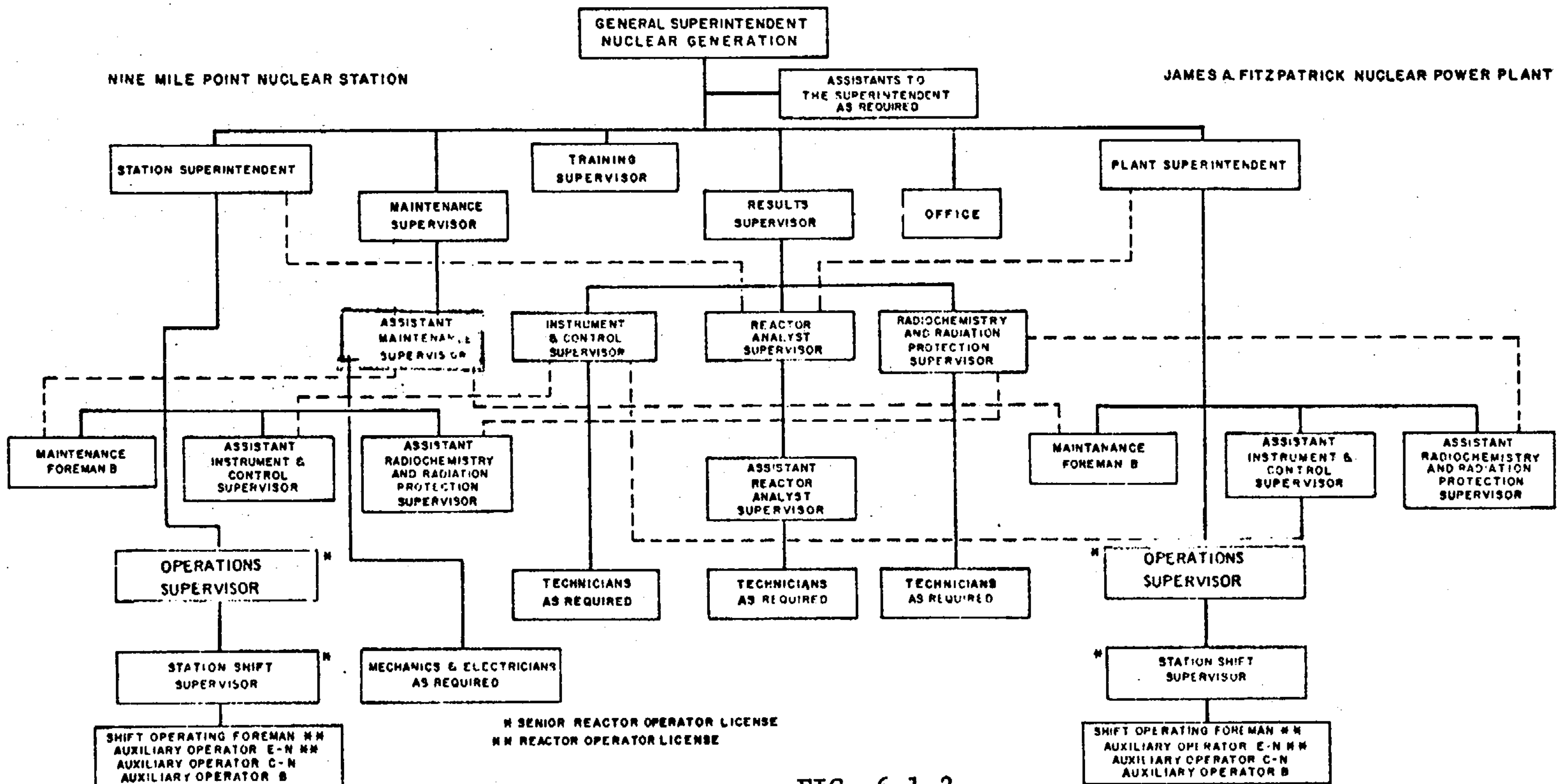

FIG. 6.1-2

JAF-NMP SITE ORGANIZATION

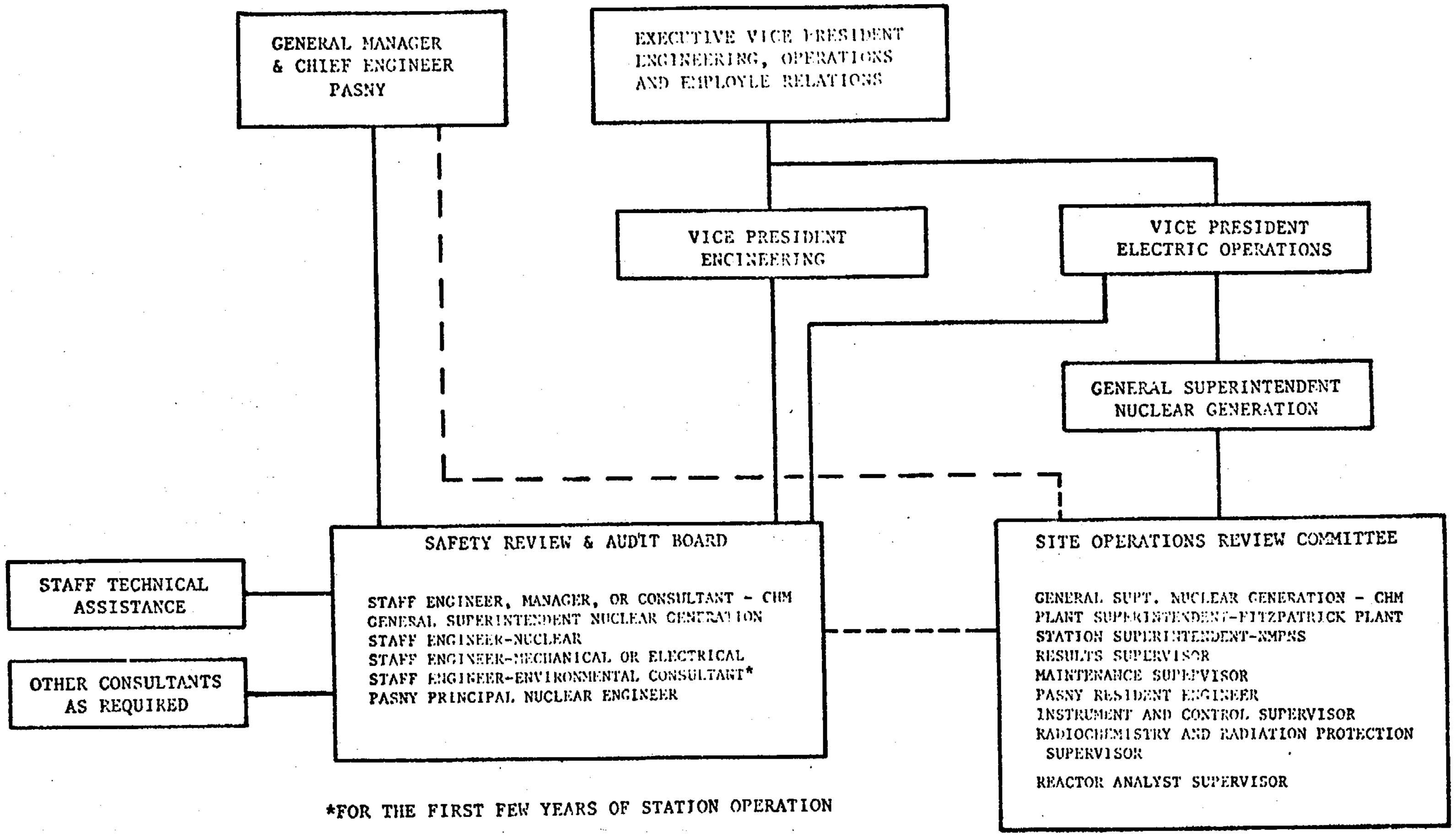

FIG. 6.5-1

NMP-JAF NUCLEAR SITE SAFETY REVIEW ORGANIZATION

TABLE 4.3.1

SAMPLE COLLECTION AND ANALYSIS

SITE RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM 3

A. LAKE PROGRAM(1)

MEDIA	ANALYSIS(5)	FREQUENCY(4)	LOCATIONS(2)	
1. Fish	GeLi, ^{89}Sr & ^{90}Sr	2/yr	2 onsite	1 offsite
2. Mollusks	GeLi, ^{89}Sr & ^{90}Sr	2/yr	2 onsite	1 offsite 3
3. Gammarus	GeLi, ^{89}Sr & ^{90}Sr	2/yr	2 onsite	1 offsite
4. Bottom Sediments	GeLi, ^{90}Sr	2/yr	2 onsite	1 offsite
5. Periphyton	GeLi	2/yr	2 onsite	1 offsite 3
6. Lake Water	GB, GSA or GeLi ^{3}H, ^{89}Sr, ^{90}Sr	M Comp. Qtr. Comp.	3(3)	

Notes:

(1) Program continued for at least three years after the startup of James A. FitzPatrick Nuclear Power Plant.
(2) Onsite locations samples collected in the vicinity of discharges, offsite samples collected at a distance of at least five miles from site. 3
(3) The three lake water samples to include Nine Mile Point Unit 1 intake water, James A. FitzPatrick intake water, and Oswego City water.
(4) Samples of items 1 through 5 collected in spring, summer and fall when available.
(5) GeLi analysis will have a MDL of 3 times σ of background based on a 400 minute count on a 55 cc GeLi system.

TABLE 4.3.1 (Cont'd)

SAMPLE COLLECTION AND ANALYSIS

SITE RADIOLOGICAL ENVIRONMENTAL MONITORING PROGRAM 3

B. LAND PROGRAM(1)

	MEDIA	ANALYSIS	FREQUENCY	NO. OF LOCATIONS	LOCATIONS		
1.	Air Particulates	GB GSA	W M Comp.(6)	At least 10	9 onsite	6 offsite	3
2.	Soil	GSA, ^{90}Sr	Every 3 years	15	9 onsite	6 offsite	3
3.	TLD	Gamma Dose	Qtr.	20	14 onsite	6 offsite	
4.	Radiation Monitors	Gamma Dose	C	10	9 onsite	1 offsite	
5.	Airborne - I^{131}	GSA	W	At least 10	9 onsite	6 offsite	
6.	Milk	I GSA, ^{90}Sr	M M Comp.	4(7)	(8)		
7.	Human Food Crops	GSA, ^{131}I	A	3	(8)		3
8.	Meat, Poultry, Eggs	GSA Edible Portions	SA	3	(8)		

Notes: (Continued)

(6) Onsite samples counted together, offsite counted together, any high count samples counted separately.
(7) Frequency applied only during grazing season. 3
(8) Samples to be collected from farms within a 10-mile radius having the highest potential concentrations of radionuclides.

Abbreviations:

M Comp. - Monthly composite of weekly or bi-weekly samples
GB - Gross beta analysis
GeLi - Gamma spectral analysis on a GeLi system (quantitative)
GSA - Gamma spectral analysis on a NaI system (quantitative)

A - Annually
W - Weekly
M - Monthly
C - Continuous

BW - Bi-weekly (alternate wks.
Qtr. - Quarterly
SA - Semiannually 3

TABLE 5.2.2-1

RESPONSIBILITIES AND AUTHORITY FOR ENVIRONMENTAL REVIEW ORGANIZATION

Site Operations Review Committee	Safety Review and Audit Board	General Manager/Chief Engineer
RESPONSIBILITIES		
Review results of environmental monitoring programs prior to submittal in each semiannual environmental monitoring report.		The function of the General Manager and Chief Engineer are identical for environmental matters with those described for safety in the plant operating technical specifications.
Review proposed changes to the environmental technical specifications and the evaluated impact of the change.		
Review proposed changes or modifications to the plant systems or equipment and the evaluated impact which would require a change in the procedures or which would affect the evaluation of the plant's environmental impact.	Review proposed environmental technical specification changes or license amendments. Submit analysis to V.P. - Electric Operations and General Manager/Chief Engineer for formal submittal to the N.R.C.	3
Review the environmental technical specification development with the safety technical specifications to avoid conflicts and for consistency.	Review violations of environmental technical specifications and submit an analysis to the V.P. Electric Operations.	3
Review all proposed procedures or changes which as determined by the plant Superintendent may affect the plant's environmental impact.		
Investigate all reported violations of environmental technical specifications. Where the investigation indicates, prepare and forward a report covering their		

TABLE 5.2.2-1 (Cont'd)

Site Operations Review Committee	Safety Review and Audit Board	General Manager/Chief Engineer
evaluation and recommendation to prevent recurrence, to the General Superintendent Nuclear Generation, and the Chairman of the Safety Review and Audit Board.		
AUTHORITY Advisory to the General Superintendent Nuclear Generation. Make tentative determinations as to whether or not proposals submitted to the committee involve a change in the plant's environmental impact. This determination subject to review by the Safety Review and Audit Board.	The Board shall report to and advise the V.P. Electric Operations and the General Manager/Chief Engineer in all matters relating to environmental impact.	3
RECORDS Separate minutes shall be kept of all meetings of the SORC when convened for review of environmental matters. Copies shall be sent to the General Superintendent Nuclear Generation and to the Chairman of the Safety Review and Audit Board.		
PROCEDURES, MEMBERSHIP, QUORUM The procedures, membership and quorum requirements shall be identical with those listed in Table 6.5-2 of Appendix A to the operating license except as noted in paragraph 5.2.2 of this environmental technical specifications.		