MAR 3 1971

Docket No. 50-220

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Niagara Mohawk Power Corporation
ATTN: Mr. Thomas J. Brosnan
300 Erie Boulevard West
Syracuse, New York 13202

Gentlemen:

A copy of the Notice of Proposed Issuance of Amendment to Provisional Operating License, which is being filed with the Office of the Federal Register for publication, is enclosed for your information. The proposed amendment (copy enclosed) would authorize operation of your Nine Mile Point Nuclear Station at power levels up to 1850 megawatts (thermal) and would incorporate changes to the Technical Specifications to provide for such operation. A copy of our related Safety Evaluation is also enclosed for your information.

It is our understanding that after the amendment and the changes to the Technical Specifications are issued, you plan to transmit to us for our use and distribution a reissued set of Technical Specifications which will incorporate all the changes to date to the Nine Mile Point Technical Specifications.

Sincerely,

Peter A. Morris, Director
Division of Reactor Licensing

Enclosures:
1. Federal Register Notice
2. Proposed Amendment to License and Changes to Technical Specifications
3. Safety Evaluation

cc w/enclosures:
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rec'd by OGC 3/3/71

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DATE ▶	2/25/71	2/25/71	3/1/71	3/3/71	3/3/71	3/3/71

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

March 3, 1971

Docket No. 50-220

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ATTN: Mr. Thomas J. Brosnan
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Syracuse, New York 13202

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Division of Reactor Licensing

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1. Federal Register Notice
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cc w/enclosures:
Arvin E. Upton, Esquire

UNITED STATES ATOMIC ENERGY COMMISSION

DOCKET NO. 50-220

NIAGARA MOHAWK POWER CORPORATION

NOTICE OF PROPOSED ISSUANCE OF AMENDMENT TO PROVISIONAL OPERATING LICENSE

The Atomic Energy Commission ("the Commission") is considering the issuance of an amendment to Provisional Operating License No. DPR-17 which presently authorizes the Niagara Mohawk Power Corporation to possess, use and operate the Nine Mile Point Nuclear Power Station located on the southeast corner of Lake Ontario in Oswego County, New York, at steady-state power levels up to a maximum of 1538 megawatts (thermal). The amendment would authorize Niagara Mohawk to operate the Nine Mile Point Nuclear Power Station at steady-state power levels up to a maximum of 1850 megawatts (thermal) in accordance with Niagara's application dated April 20, 1970, and amendments thereto.

The Commission has found that the application for the amendment complies with the requirements of the Atomic Energy Act of 1954, as amended, and the Commission's regulations published in 10 CFR Chapter I. The license amendment will be issued after the Commission makes the findings relating to its review of the application, which are set forth in the proposed amendment, and concludes that the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Within thirty days from the date of publication of the notice in the FEDERAL REGISTER, the applicant may file a request for a hearing and any person whose interest may be affected by this proceeding may file a petition for leave to intervene. Requests for a hearing and petitions to intervene shall be filed in accordance with the Commission's "Rules of Practice" in 10 CFR Part 2. If a request for a hearing or a petition for leave to intervene is filed within the time prescribed in this notice, the Commission will issue a notice of hearing or an appropriate order.

For further details with respect to this amendment, see (1) the application for license amendment dated April 20, 1970, Amendments 1 through 5 thereto, and letter dated November 23, 1970; (2) the Report of the Advisory Committee on Reactor Safeguards dated February 6, 1971; (3) the proposed amendment to the provisional operating license; (4) the proposed changes to the Technical Specifications which are incorporated in the proposed license amendment; and (5) a related Safety Evaluation prepared by the Division of Reactor Licensing, all of which are available for public inspection at the Commission's Public Document Room at 1717 H Street, N. W., Washington, D. C. A copy of each of items (3) through (5) above may be obtained upon request sent to the U. S. Atomic Energy Commission, Washington, D. C. 20545, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this 3rd day of March 1971.

FOR THE ATOMIC ENERGY COMMISSION

Peter A. Morris

Peter A. Morris, Director
Division of Reactor Licensing

UNITED STATES ATOMIC ENERGY COMMISSION

NIAGARA MOHAWK POWER CORPORATION

DOCKET NO. 50-220

PROPOSED AMENDMENT TO PROVISIONAL OPERATING LICENSE

License No. DPR-17
Amendment No. 2

The Atomic Energy Commission ("the Commission") has found that:

A. The application for amendment dated April 20, 1970, as supplemented by Amendments 1 through 5 thereto and letter dated November 23, 1970, complies with the requirements of the Atomic Energy Act of 1954, as amended ("the Act"), and the Commission's regulations set forth in 10 CFR Chapter I;

B. There is reasonable assurance (i) that the facility can be operated at power levels up to 1850 megawatts (thermal) in accordance with the license, as amended, without endangering the health and safety of the public, and (ii) that such operation will be conducted in compliance with the regulations of the Commission; and

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

Accordingly, Provisional Operating License No. DPR-17 issued to Niagara Mohawk Power Corporation for operation of the Nine Mile Point Nuclear Power Station is hereby further amended to restate subparagraphs 3.A., 3.B., and 3.C. in their entirety to read as follows:

3.A. Maximum Power Level

Niagara Mohawk is authorized to operate the facility at steady-state power levels up to a maximum of 1850 megawatts (thermal).

3.B. Technical Specifications

The Technical Specifications contained in Appendix A to the license, as modified by Changes Nos. 1 through 3 and Change No. 4 appended hereto as Attachment A, are hereby incorporated in this license. Niagara Mohawk shall operate the facility in accordance with these Technical Specifications. No changes shall be made in the Technical Specifications unless authorized by the Commission as provided in Section 50.59 of 10 CFR Part 50.

3.C. Reports

Niagara Mohawk shall make certain reports in accordance with the requirements of the Technical Specifications.

This amendment is effective as of the date of issuance.

FOR THE ATOMIC ENERGY COMMISSION

Peter A. Morris, Director
Division of Reactor Licensing

Attachment A - Change No. 4 to
the Technical Specifications

Date of Issuance:

ATTACHMENT A TO AMENDMENT NO. 2

NIAGARA MOHAWK POWER CORPORATION

LICENSE NO. DPR-17

CHANGE NO. 4 TO TECHNICAL SPECIFICATIONS

Make the following changes within the specifications and sections on the indicated pages:

SECTION 1 - DEFINITIONS

Change Definition 1.13 on pages 4 and 5 in its entirety to read as follows:

"An abnormal occurrence is defined as:

a. Violation of Limiting Safety System Settings.

b. Violation of Limiting Conditions for Operation.

c. Engineered safety system component malfunction or other component or system malfunctions which could, or threaten to, render the system incapable of performing its intended safety function.

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

e. Uncontrolled or unanticipated changes in reactivity.

f. Observed inadequacies in the implementation of either administrative or procedural controls, such that the inadequacy could have caused the existence or development of an unsafe condition in connection with the operation of the reactor."

Change Definition 1.15 on page 5 to read:

"Rated flux is the neutron flux that corresponds to a steady-state power level of 1850 thermal megawatts. The use of the term 100 percent also refers to the 1850 thermal megawatt power level."

SECTION 2 - SAFETY LIMITS AND LIMITING SAFETY SYSTEM SETTINGS

Specification 2.1.1.b - page 6

Change "307 MWt" to "333 MWt" at the end of the statement.

Specification 2.1.1.c - page 6

Change "3.0 seconds" to "1.7 seconds" within the first sentence.

Specification 2.1.1 Bases - page 7

First paragraph, line 6:

Change "APED-3892" to "APED-5286" and change the corresponding reference at the bottom of the page to "J. M. Healzer, J. R. Hench, E. Janssen, S. Levy 'Design Basis for Critical Heat Flux Conditions in Boiling Water Reactors', APED-5286, September 1966".

Third paragraph, line 3 and next to last line:

Change "1000 psig" to "1030 psig".

Fourth paragraph, line 3 and 12:

Change "3.13" to "3.06".

Fourth paragraph, line 8:

Change the sentence, "It is possible ... not be permitted" to read:

"It is possible that during temporary control rod manipulation activities or at the end of core life when it might be desirable from a scheduling standpoint to delay a refueling outage, a peaking factor greater than 3.06 could result."

<u>Specification 2.1.1 Bases - page 8</u>

Fourth paragraph, line 13:

Change sentence to read:

"This is equivalent to a core power of 333 thermal megawatts or 18 percent of the full design rating (1850 thermal megawatts)."

<u>Specification 2.1.1 Bases - page 9</u>

First paragraph:

In line 3, change "3.25 seconds" to "1.85 seconds".

Delete the entire sentence that begins at line 6.

In line 13, change "3.0 seconds" to "1.7 seconds".

Specification Figure 2.1.1 - page 10

Replace "Figure 2.1.1" with the attached "Figure 2.1.1 Revised".

Specification 2.1.2 Bases - page 11

Change first paragraph of Bases in its entirety to read:

"The LS^3 values were established on the basis of analysis starting at an operating power-flow characteristic curve as shown in revised Figure 2.1.1, ref. 9. Nominal LS^3 values were used in the malfunction analysis. (6, 7, 9) As discussed in the First Supplement*, (p. III-29) instrumentation errors are accounted for in the CHF correlation and in the assessment of effects on steady-state MCHFR calculations. Deviations such as inherent instrument error, operator setting error, and drift of the set point are included in the conservatism of the calculations. For the transient analyses, conservatism in the effects on MCHFR are provided by the conservatisms incorporated in the controlling factors used in the analysis, such as void reactivity coefficient, control rod scram worth, scram delay time, power shapes, etc. Most transients analyzed have at least two independent scram functions available to terminate the specific transient. However, where multiple scram protection does not exist, additional backup is provided by procedures and interlocks or an inherent safe response."

Add reference 9 to the bottom of page 11 to read:

"(9) Technical Supplement to Petition to Increase Power Level, dated April 1970."

Change the first paragraph of Bases a. as follows:

In line 2, replace "Volume I (Figure IV-13)*" with "Figure 2.1.1 Revised (ref. 9)".

In line 7, insert the word "Revised" after "Figure 2.1.2".

In line 12, replace "(Appendix E*)" with "(refs. 6, 7, and 9)".

In line 15, Change "1050" to "1080".

In line 16, after the word "psig" add the following to the end of the sentence:

", the turbine stop valve closure scram set at $\leq$ 10 percent from full open, and the generator load rejection scram set on loss of oil pressure to the acceleration relay when power is greater than 45%."

Specification 2.1.2 Bases - page 12

Add the following paragraph immediately prior to Bases b. as a last paragraph of Bases a:

"The thermal hydraulic safety limit of Figure 2.1.1 Revised was based on a total peaking factor of 3.06, and an adjustment is required in the unusual event of higher peaking factors. Likewise, power indication at which scram occurs must be adjusted to assure MCHFRs above 1.0 for expected transients in this derated condition. This is provided by application of the formula on Figure 2.1.2 Revised."

In the first paragraph of Bases b., change the sentence beginning on line 10 to read:

"This provides 50 percent margin between the maximum power and the safety limit at 18 percent of rated power."

Specification 2.1.2.c - page 13

Change to read:

"The reactor high-pressure scram trip setting shall be $\leq$ 1080 psig."

Specification 2.1.2 Bases c - page 13

Change first paragraph in its entirety to read:

"As demonstrated in Appendix E-1* and the Technical Supplement to Petition to Increase Power Level, the reactor high pressure scram is a backup to the neutron flux scram, turbine stop valve closure scram, generator load rejection scram, and main steam isolation-valve closure scram, for various reactor isolation incidents. However, rapid isolation at lower power levels generally results in high pressure scram preceding other scrams because the transients are slower and those trips associated with the turbine-generator are bypassed."

In line 1 and 3 of second paragraph, change "1050 psig" to "1080 psig".

Specification 2.1.2 Bases d - page 13

In line 5 of first paragraph, change "Appendix E-I.3.9*" to "Technical Supplement to Petition to Increase Power Level, dated April 1970".

Specification 2.1.2 Bases f - page 14

In line 3, change " $\leq$ 110 percent" to " $\leq$ 106 percent".

Specification 2.1.2 Bases g - h - page 14

Change the last sentence of the first paragraph at line 20 to read:

"With the scrams set at $\leq$ 10 percent valve closure, there is no increase in neutron flux and peak pressure in the vessel dome is limited to 1141 psig (Technical Supplement to Petition to Increase Power Level, dated April 1970)."

Specification 2.1.2 - page 15

Add Limiting Safety System Setting Sections i and j and the corresponding Bases i and j as follows:

Section i. "The generator load rejection scram shall be initiated by the signal for turbine control valve fast closure due to a loss of oil pressure to the acceleration relay any time the turbine first stage steam pressure is above a value corresponding to 833 thermal megawatts. i.e., 45 percent of 1850 thermal megawatts."

Section j. "The turbine stop valve closure scram setting shall be initiated $\leq$ 10 percent of valve closure (stem position) from full open whenever the turbine first stage steam pressure is above a value corresponding to 833 thermal megawatts, i.e., 45 percent of 1850 thermal megawatts."

Bases i. "The generator load rejection scram is provided to anticipate the rapid increase in pressure and neutron flux resulting from fast closure of the turbine control valves due to the worst case transient of a load rejection and subsequent failure of the bypass. In fact, analysis (Technical Supplement to Petition to Increase Power Level) shows that heat flux does not increase from its initial value at all because of the fast action of the load rejection scram; thus, there is no MCHFR decrease."

Bases j. "The turbine stop valve closure scram is provided for the same reasons as discussed in i above. With a scram setting of $\leq$ 10 percent valve closure, the resultant transients are nearly the same as for those described in i above; and, thus, adequate margin exists."

<u>Specification Figure 2.1.1 - page 16</u>

Replace "Figure 2.1.2" with attached "Figure 2.1.2 Revised".

<u>Specification 2.2.1 Bases - page 17</u>

In third paragraph, last line, change "1000 psig" to "1030 psig".

<u>Specification 2.2.2 - page 18</u>

Section 2.2.2.a - Change the number of safety valves at 1218 psig set point to four and the total number of valves to 16 as follows:

Set Point (psig)	Number of Safety Valves
1218	4
1227	3
1236	3
1245	3
1254	3
	16

Section 2.2.2.b - Change the trip setting from " ≤1050 psig" to " ≤1080 psig".

Bases a - Change first paragraph in its entirety to read:

"The range of set points for safety valve actuation is selected in accordance with code requirements. A safety valve capability study presented in the Technical Supplement to Petition to Increase Power Level using the stated LS^3 valves has demonstrated the maximum pressures occurring at the bottom of the reactor vessel and the bottom of the recirculation piping are 1303 psig and 1315 psig, respectively, some 72 psig below the 1375 psig safety limit. This analysis has assumed the highly improbable event of reactor isolation occurring without scram, in spite of separate and redundant scram signals such that the power output reached 167 percent of rated (1850 thermal megawatts)."

Bases a, second paragraph - In line 11, change "15" to "16".

Bases b, both paragraphs - Change to read as follows:

"The reactor high pressure scram setting is relied upon to terminate rapid pressure transients if other scrams, which would normally occur first, fail to function. As demonstrated in

Appendix E-1 of the FSAR and the Technical Supplement to Petition to Increase Power Level, Page II-12, the reactor high pressure scram is a backup to the neutron flux scram, generator load rejection scram, and main steam isolation-valve closure scram for various reactor isolation incidents. However, rapid isolation at lower power levels generally results in high pressure scram preceding other scrams because the transients are slower and those trips associated with the turbine-generator are bypassed."

"The operator will not set the trip setting at 1080 psig or lower. However, the actual set point can be as much as 15.8 psi above the 1080 psig indicated set point due to the deviations discussed above."

<u>SECTION 3 - LIMITING CONDITIONS FOR OPERATION</u>; and

<u>SECTION 4 - SURVEILLANCE REQUIREMENTS</u>

<u>Specification 4.1.1.c - page 24</u>

Replace the specification c under Surveillance Requirement in its entirety with the following:

"c. Scram Insertion Times

(1) After each major refueling outage and prior to power operation with reactor pressure above 800 psig, all operable control rods shall be scram time tested from the fully withdrawn position.

(2) Following each reactor scram from rated pressure, the mean 90% insertion time shall be determined for eight selected rods. If the mean 90% insertion time of the selected control rod drives does not fall within the range of 2.4 to 3.1 seconds or the measured scram time of any one drive for 90% insertion does not fall within the range of 1.9 to 3.6 seconds, an evaluation shall be made to provide reasonable assurance that proper control rod drive performance is maintained.

(3) Following any outage not initiated by a reactor scram, eight rods shall be scram tested with reactor pressure above 800 psig. The same criteria of 4.1.1.c.(2) shall apply."

Specification 3.1.1.c/4.1.1.c Bases c - page 24

Change the last sentence (beginning line 11) of the first paragraph to read:

"Analyses presented in Appendix E-I*, the Second Supplement, and the Technical Supplement to Petition to Increase Power Level were based on these times, and demonstrate the adequacy of the scram times chosen."

Specification 4.1.1.d - page 24

Add Surveillance Requirement d as follows:

"d. Control Rod Accumulators

Once a shift check the status of the accumulator pressure and level alarms in the control room."

Specification 3.1.1.f., 4.1.1.f. and Bases f - page 25

Replace Specification 3.1.1.f. under Limiting Condition for Operation in its entirety with the following:

"f. Reactivity Anomalies

The difference between an observed and predicted control rod inventory shall not exceed the equivalent of one percent in reactivity. If this limit is exceeded, the reactor shall be brought to the cold, shutdown condition by normal orderly shutdown procedure. Operation shall not be permitted until the cause has been evaluated and appropriate corrective action has been completed. The AEC shall be notified within 24 hours of this situation in accordance with Specification 6.8.d."

Add Specification 4.1.1.f. under Surveillance Requirement to read as follows:

"f. Reactivity Anomalies

The observed control rod inventory shall be compared with a normalized computed prediction of the control rod inventory during startup, following refueling or major core alteration. These comparisons will be used as base data for reactivity monitoring during subsequent power operation throughout the fuel cycle. At specific power operating conditions, the actual control rod configuration will be compared with the expected configuration based upon appropriately corrected past data. This comparison will be made every equivalent full power month."

Under Bases, item f, following the third sentence of the paragraph (line 9 between words "... state." and "During ...") insert the following sentence:

"Equilibrium xenon, samarium and power distribution are considered in establishing the steady-state base condition to minimize any source of error."

Specification 3.1.2, 4.1.2 and Bases - page 26

Replace Specification 3.1.2.b under Limiting Condition for Operation in its entirety with the following:

"b. If a redundant component becomes inoperable, Specification 3.1.2.a. shall be considered fulfilled, provided that the component is returned to an operable condition within 7 days and the additional surveillance required is performed."

Change Specification 3.1.2.c. in line 3 from "Figure 3.1.2.a" to "Figure 3.1.2.a Revised".

Add a second and third paragraph to Specification 4.1.2.a.(1) under Surveillance Requirement as follows:

"Remove the squibs from the valves and verify that no deterioration has occurred by actual field firing of the removed squibs. In addition, field fire one squib from the batch of replacements.

Disassemble and inspect the squib-operated valves to verify that valve deterioration has not occurred."

Replace paragraphs two, three and four (on pages 26 and 27) under Bases with the following paragraphs:

"The liquid poison system is designed to provide the capability to bring the reactor from full design rating (1850 thermal megawatts) to a cold, xenon free shutdown condition assuming none of the control rods can be inserted. To meet this objective, the system is designed to inject a quantity of boron which produces a concentration of at least 600 ppm of boron in the reactor core in less than 120 minutes. This concentration will bring the reactor from full design rating (1850 thermal megawatts) to greater than 3 percent delta k subcritical (0.97 k_{eff}) considering the combined effects of the control rods, coolant voids, temperature change, fuel doppler, xenon, and samarium.

"In order to provide good mixing, the injection time has to be greater than 60 minutes. The maximum injection time of 120 minutes is necessary to override the rate of reactivity insertion due to cooldown of the reactor, including the xenon decay, by a considerable margin.

"The liquid poison storage tank volume-concentration requirements of Figure 3.1.2.a Revised assure that the above requirements for boron solution insertion are met with one 30 gpm liquid poison pump. The point (2000 gallons, 20.4%) results in the required amount of solution being inserted into the reactor in not less than 60 minutes, and therefore, defines the maximum concentration-minimum value requirement. The point (3800 gallons, 10.7%) results in the required amount of solution being injected into the reactor in not more than 120 minutes and therefore defines the minimum concentration requirement. The boundary line joining these points is the locus of points from which the required amount of boron, with 25 percent margin to allow for any unexpected non-uniform mixing, will be inserted into the reactor in the allowable times. The maximum volume, 4080 gallons, is established by the tank capacity. The tank volume requirements include consideration for 197 gallons of solution which is contained below the point where the pump takes suction from the tank and therefore cannot be inserted into the reactor."

Specification Figure 3.1.2.a - page 28

Replace "Figure 3.1.2.a" with the attached "Figure 3.1.2.a Revised".

Specification 3.1.3, 4.1.3 and Bases - page 30

Replace Specification 3.1.3.b. under Limiting Condition for Operation in its entirety with the following:

"b. If one emergency cooling system becomes inoperable, Specification 3.1.3.a. shall be considered fulfilled, provided that the inoperable system is returned to an operable condition within 7 days and the additional surveillance required is performed."

In line 3 of third paragraph under Bases, change "1060 psig" to "1090 psig".

Specification 3.1.4, 4.1.4 and Bases - page 32

Replace Specification 3.1.4.b and c under Limiting Condition for Operation in their entirety with the following:

"b. If a redundant component of a core spray system becomes inoperable, that system shall be considered operable provided that the component is returned to an operable condition within 15 days and the additional surveillance required is performed."

"c. If a redundant component in each of the core spray systems becomes inoperable, both systems shall be considered operable provided that the component is returned to an operable condition within 7 days and the additional surveillance required is performed."

Change the last sentence (beginning on line 4) of first paragraph under Bases to read as follows:

"For the worst line break, a loss-of-coolant accident, a core spray of at least 3400 gpm is required within 35 seconds to provide fuel stability sufficiently to assure effective core cooling."

Change item 3. under Bases (second paragraph on page 33) to read as follows:

"3. The core spray delivery rate of 3400 gpm shall be available at the core spray nozzles inside the reactor vessel within 35 seconds."

Specification 3.1.5/4.1.5 Bases - page 34

Replace reference in second paragraph (line 10) "(Appendix E-1.3.6*)" with "(Section II.xv, Technical Supplement to Petition to Increase Power Level, dated April 1970)".

Specification 3.1.6 - page 36

Replace Specification 3.1.6.b. under Limiting Condition for Operation in its entirety with the following:

"b. If a redundant component becomes inoperable, the control rod drive pump coolant injection system shall be considered operable provided that the component is returned to an operable condition within 7 days and the additional surveillance required is performed."

Specification 3.2.4, 4.2.4 and Bases - page 46

Change Specification 4.2.4.(1) under Surveillance Requirement to read as follows:

"(1) Samples shall be taken at least every 96 hours and analyzed for gross gamma activity."

Replace first paragraph under Bases in its entirety with the following:

"The primary coolant radioactivity concentration limit of 25 uCi total iodine per gram of water was calculated based on a steamline break accident which is isolated in 10.5 seconds. For this accident analysis, all the iodine in the mass of coolant released in this time period is assumed to be released to the atmosphere at the top of the turbine building (30 meters). By limiting the thyroid dose at the site boundary to a maximum of 30 Rem, the iodine concentration in the primary coolant is back-calculated assuming fumigation meteorology, Pasquill Type F at 1 m/sec. The iodine concentration in the primary coolant resulting from this analysis is 25 uCi/gm."

Specification Table 4.2.6 - pages 52, 53, 54

Delete the words "Main Steam" at item B.1 on page 52 under column heading Component.

Add new item B.4 on page 52 as follows:

	Component	Sample	Extent	Inspection Process See Note 1	Inspection Frequency See Note 2
"4."	"Main Steam Line"	"All accessible circumferential welds greater than 4" diameter"	"Entire circumference"	"UT"	"g"

Delete the words "Main Steam" at item E.1 on page 53 under column heading Component.

Add new item E.4 on page 53 as follows:

	Component	Sample	Extent	Inspection Process See Note 1	Inspection Frequency See Note 2
"4."	"Main Steam Line"	"All accessible circumferential welds greater than 4" diameter"	"Entire circumference"	"UT"	"h"

Under Note 2:, on page 54, add items (g) and (h) as follows:

"g. Inspect on a frequency that will cover each accessible weld at least every seven years of operation."

"h. Inspect on a frequency that will cover each accessible weld at least every eight years of operation."

Specification 3.2.8, 4.2.8 and Bases - page 57

Change line 4 of Specification 3.2.8.a. under Limiting Condition for Operation to read: "... all sixteen of the safety ...".

Change line 2 of Specification 4.2.8.a. under Surveillance Requirement to read: "... of the sixteen safety valves shall ...".

Change line 4 of first paragraph under Bases to read "... of all 16 safety valves will limit reactor ...".

Specification 3.2.9/4.2.9 Bases - page 58

Change the last line of the second paragraph under Bases from "Appendix E (I.3.11)" to "Technical Supplement to Petition to Increase Power Level, Section II.xv".

Specification 3.3.3, 4.3.3 and Bases - pages 65, 66, 67

On page 65 in line 3 of third paragraph under Bases, change "2.0%/day" to "1.9%/day".

On page 66 under Surveillance Requirement, make the following changes:

Item b.(1), line 2, change "1.6" to "1.5".

Item b.(2), equation on line 3, change "1.6" to "1.5".

On page 66 under Bases, make the changes as follows:

Second paragraph, lines 1 and 2, change from "... as specified in 4.3.3.a is 1.6%/day ..." to "... as specified in 4.3.3.b is 1.5%/day ...".

Second paragraph, line 4, change "2.0%/day" to "1.9%/day".

Third paragraph, line 2, change "3.2%/day" to "3.0%/day".

Third paragraph, line 4, change "1.6%/day" to "1.5%/day".

On page 67 under Surveillance Requirement, add Specification 4.3.3.h. to read as follows:

"h. Inspection

The accessible interior surfaces of the drywell shall be visually inspected each operating cycle for evidence of deterioration."

Specification 3.3.4, 4.3.4 and Bases - page 68

Add Specification 4.3.4.c. under Surveillance Requirement to read as follows:

"c. At least once per operating cycle, each instrument-line flow check valve will be tested for operability."

Add a fifth paragraph under the Bases to read as follows:

"In addition to routine surveillance as outlined in First Addendum to Technical Supplement to Petition to Increase Power Level, each instrument-line flow check valve will be tested for operability. All instruments on a given line will be isolated at each instrument. The line will be purged by isolating the flow check valve, opening the bypass valves, and opening the drain valve to the equipment drain tank. When purging is sufficient to clear the line of non-condensibles and crud the flow-check valve will be cut into service and the bypass valve closed. The main valve will again be opened and the flow check valve allowed to close. The flow check valve will be reset by closing the drain valve and opening the bypass valve depressurizing part of the system. Instruments will be cut into service after closing of the bypass valve. Repressurizing of the individual instruments assures that flow check valves have reset to the open position."

Specification 3.3.6 - page 71

Replace Specification 3.3.6.b. under Limiting Condition for Operation in its entirety with the following:

"b. If a redundant component in each set or either set of valves becomes inoperable, Specification 3.3.6.a. shall be considered fulfilled, provided that the component or set of valves is returned to an operable condition within 15 days and that the additional surveillance required is performed."

Specification 3.3.7 - page 73

Replace Specifications 3.3.7.b. and c. under Limiting Condition for Operation in its entirety with the following:

"b. If a redundant component of a containment spray system becomes inoperable, Specification 3.3.7.a. shall be considered fulfilled, provided that the component is returned to an operable condition within 15 days and that the additional surveillance required is performed."

"c. If a redundant component in each of the containment spray systems or their associated raw water systems become inoperable, both systems shall be considered operable provided that the component is returned to an operable condition within 7 days and that the additional surveillance required is performed."

Specification 3.4.4 and 4.4.4 - page 80

Replace Specification 3.4.4.b. under Limiting Condition for Operation in its entirety with the following:

"b. If one branch of the emergency ventilation system becomes inoperable, Specification 3.4.4.a. shall be considered fulfilled, provided that the branch is returned to an operable condition within 7 days and that the additional surveillance required is performed."

Change Specification 4.4.4.b.(1) and (2) under Surveillance Requirement to read as follows:

"(1) The removal efficiency of the particulate filters is not less than 99% for 0.3 micron mean particulate matter based on a hot dioctylphthalate (DOP) test."

"(2) The removal efficiency of the charcoal filters is not less than 99% based on a freon test."

Specification 3.6.1 - page 87

Change Section 3.6.1, Applicability, Objective and Specification 3.6.1.a.(1) and (2), inclusive, under Limiting Condition for Operation in its entirety to read as follows:

"3.6.1 Station Process Effluents

Applicability

Applies to the radioactive effluents from the station.

Objective

To assure that radioactive material is not released to the environment in an uncontrolled manner and to assure that any material released is kept as low as practical and, in any event, is within the limits of 10 CFR 20.

Specification

a. Stack Release

(1) The maximum release rates of gross activity, except iodines and particulates with half

lives longer than eight days, shall be limited in accordance with the following equation:

$$Q \leq \frac{0.57}{\bar{E}} \text{ (Ci/sec)}$$

where Q is the stack release rate (Ci/sec) of gross activity and $\bar{E}$ is the average gamma energy per disintegration (MeV/dis).

(2) The maximum release rate of iodines and particulates with half lives longer than eight days shall be limited in accordance with the following equation:

$$Q \leq 1.5 \times 10^4 \; MPC_i \text{ (Ci/sec)}$$

where Q is the stack release rate (Ci/sec) of iodines and particulates with half lives longer than eight days and MPC_i ($\mu Ci/cm^3$) is the maximum permissible concentration in air as defined in Column 1, Table II, of Appendix B and Note 1 thereto of 10 CFR 20."

<u>Specification 4.6.1 - page 87</u>

Change Specification 4.6.1.a.(1) and (2) under Surveillance Requirement in its entirety to read as follows:

"a. Stack Release

(1) Station records of gross stack release rate of gaseous activity shall be maintained on an hourly basis by evaluation of the recorded data from the stack gas monitor to assure that the specified rates are not exceeded and to yield information concerning general integrity of the fuel cladding. Records of isotopic analyses shall also be maintained. Within one month after initial commercial service of the unit and within one month following refuelings, an isotopic analysis will be made of the gaseous activity release rate. From this sample, a ratio of long-lived and short-lived activity will be established. Samples of off-gas will be taken at least every 96 hours and gross ratio of long-lived and short-lived activity determined. When these samples indicate a change in the ratio of greater than 20% from the ratio established by the previous isotopic analysis, a new isotopic analysis shall be performed. A new isotopic analysis of off-gas will be performed at least quarterly. Gaseous release of tritium shall be calculated on a monthly basis from measured data."

(2) Station records of stack release of iodines and particulates with half lives greater than eight days shall be maintained on the basis of all filter cartridges counted. These cartridges shall be analyzed weekly when the iodine or particulate release rate is less than 10 percent of the maximum release rate given in Specification 3.6.1.a.(2), otherwise the cartridges shall be removed for analysis daily. When the gross release rate exceeds 1 percent of the maximum release rate given in Specification 3.6.1.a.(1) and the average daily gross activity release rate increases by 50 percent over the previous day, the cartridges shall be analyzed to determine the release rate increase for iodines and particulates."

Add Specification 4.6.1.a.(5) and (6) under Surveillance Requirement to read as follows:

"(5) At least once during each shift, a sensor check of the off-gas and stack gas monitors shall be made."

"(6) At least once during each operating cycle (prior to startup), verify automatic securing and isolation of the mechanical vacuum pump."

Specification 3.6.1/4.6.1 Bases - pages 87,88

Replace all paragraphs under item (1) of "Stack Release" Bases with the following:

"a. Stack Release

(1) Detailed studies were conducted to establish a calculated rate for stack emission to the uncontrolled environment in accordance with the limits of 10 CFR 20 and are described in Appendix D of the FSAR. These calculations consider site meteorology, buoyancy characteristics, statistical tolerance for the environmental monitoring program, and isotopic content of the effluent as given in Table A-12 of the FSAR. Independent dose calculations for several locations offsite have been made by the AEC staff. The method utilized onsite meteorological data developed by the licensee and utilized diffusion assumptions appropriate to the site.

The method utilized by the AEC staff is described in Section 7-5.2.5 of "Meteorology and Atomic Energy - 1968", equation 7.63 being used. The results of these calculations were equivalent to those generated by the licensee provided the average gamma energy per disintegration for the assumed noble gas mixture with a

30-minute holdup is 0.7 MeV per disintegration. Based on these calculations, a maximum release rate limit of gross activity, except for iodines and particulates with half lives longer than eight days, in the amount of $0.57/\bar{E}$ curies per second will not result in offsite annual doses in excess of the limits specified in 10 CFR 20. The $\bar{E}$ determination need consider only the average gamma energy per disintegration since the controlling whole body dose is due to the cloud passage over the receptor and not cloud submersion in which the beta dose could be additive.

Field sampling and dose measurements in accordance with the environmental monitoring program will begin when the gross release rate of gaseous effluent approaches approximately 0.1 Ci/sec. The graded nature of the program and the location of sampling stations are described in Appendix D-4.1 of the FSAR. The sampling frequencies will be monthly during the appropriate seasons for each type of sample, provided that the effluent release rate is less than 1/3 of the maximum release rate limit. For higher release rates, the sampling frequency will be increased to weekly."

"(2) Detailed calculations of ground level air concentrations of iodines and particulates with half lives longer than eight days at several offsite locations have been made as described in Appendix D of the FSAR. These calculations consider site meteorology and buoyancy characteristics of the effluent. Based on these calculations, the release rate limit for these isotopes in the equation in Section 3.6.1.a.(2) is obtained. Use of this equation assures that releases will not result in offsite doses in excess of those specified in 10 CFR 20.

The assumptions used by the AEC staff for these calculations were: (1) onsite meteorological data for the most critical 22½ degree sector, (2) no building wake credit used, and (3) to consider possible reconcentration effects a reduction factor of 700 was applied to allow for the milk production and consumption mode of uptake. The reduction factor of 700 has been incorporated into the equation in Section 3.6.1.a.(2)."

On page 88 under Bases change the first paragraph designated as number "(2)" to number "(3)" and correspondingly "(3)" to "(4)".

Specification 3.6.1 - page 88

Replace Specification 3.6.1.b. under Limiting Condition for Operation in its entirety to read as follows:

"b. Liquid Effluent

(1) The concentration of gross beta activity (above background) in the condenser cooling water discharge canal shall not exceed the limits stated below unless the discharge is controlled on a radionuclide basis in accordance with Appendix B, Table II, Column 2, of 10 CFR 20 and note 1 thereto:

Maximum Concentration (excluding tritium) -

1×10^{-7} µCi/ml

(2) Maximum tritium concentration

3×10^{-3} µc/ml

(3) The radiation monitor on the discharge line from the waste disposal tanks to the discharge tunnel shall be operative or if not, two independent samples of each tank shall be taken and two Station personnel shall independently check valving prior to discharge of liquid effluents."

Specification 4.6.1 - Page 88

Replace Specification 4.6.1.b. under Surveillance Requirement in its entirety to read as follows:

"b. Liquid Effluent

"(1) Station records shall be maintained of the radioactive concentration and volume before dilution of each batch of liquid effluent released and of the average dilution flow and length of time over which each discharge occurred.

"(2) Each batch of radioactive liquid effluent shall be sampled and analyzed prior to release.

"(3) The liquid effluent radiation monitor shall be calibrated quarterly, shall have an instrument channel test monthly, and a sensor check daily.

"(4) Isotopic analysis of a representative batch of liquid waste shall be performed at least once per quarter. Each batch of liquid waste shall be counted for gross beta activity and when released on a radionuclide basis, the analysis shall also include a gross gamma count and gamma scan. If gamma energy peaks other than those determined by the previous isotopic analysis are found, a new isotopic analysis shall be performed and recorded. An isotopic analysis shall also be performed should there be significant changes in the gamma to beta ratio of the batch.

"(5) Grab samples shall be collected monthly from the discharge canal and analyzed for gross beta activity.

"(6) A sample of a representative waste batch shall be analyzed for tritium at least once per quarter.

"(7) The performance and results of independent samples and valve checks shall be logged."

Specification 3.6.1/4.6.1 Bases - Pages 88, 89

Replace all paragraphs under items (1) and (2) of "Liquid Effluent" Bases with the following:

"b. Liquid Effluent

"(1) Radioactive effluents released from the Station to unrestricted areas on the basis of gross beta analysis are based on the assumption that iodine 129 and radium are not present. Accordingly, Appendix B, Table II, Column 2 of 10 CFR 20 will permit a concentration up to 1 x 10^{-7} uCi/ml in the cooling water discharge canal.

If radioactive effluents are released to unrestricted areas on a radionuclide basis, the MPC shall be determined and controlled in the cooling water discharge canal in accordance with Appendix B, Table II, Column 2 of 10 CFR 20 and note 1 thereto.

The release of effluents on a radionuclide basis shall be based on an isotopic analysis of a typical waste batch. This analysis shall be performed at least quarterly and shall include specific radiochemical separations for ^{90}Sr and ^{131}I. Along with an isotopic analysis, a gross gamma and a gross beta counting efficiency shall be determined for the particular isotopic mixture and a gamma/beta ratio established.

A required dilution factor for the isotopic mixture shall be determined using the following formula:

$$\text{Required D.F.} = \frac{C_1}{MPC_1} + \frac{C_2}{MPC_2} + \ldots + \frac{C_n}{MPC_n}$$

Where: C_1 = concentration of Isotope 1, etc.

MPC_1 = MPC from Appendix B, Table II, Column 2, 10 CFR 20

This dilution factor can be expressed as a MPC for the isotopic mixture thus:

$$\text{Mixture MPC} = \frac{\text{gross concentration}}{\text{Required D.F.}}$$

This mixture MPC shall be used to determine the appropriate discharge rates for waste batches but can only be used for the particular mixture as determined above.

In order to verify that the mixture has not significantly changed, each batch shall be counted for gross beta, gross gamma and shall have a gamma scan performed. Significant changes ($\pm$ 50%) in the gamma/beta ratio or the appearance of new energy peaks in the gamma scan shall require a new isotopic analysis to be performed.

The minimum frequency of the isotopic analysis will be varied depending on the average discharge canal concentration. A minimum of one isotopic analysis per quarter will be performed as long as the average concentration of the discharge canal is less than 1% of the previously calculated Mixture MPC. An average concentration of between 1% and 10% of Mixture MPC shall require an isotopic analysis at least monthly. For average concentrations greater than 10% of the Mixture MPC, each batch shall be isotopically analyzed. The average concentration shall be calculated daily and shall be a running average annual concentration.

An environmental monitoring program in the lake will be conducted as outlined in Appendix D-4.2 of the FSAR. Samples required under this graded program will be taken twice a year unless the average discharge canal concentration exceeds 1×10^{-7} in which case weekly analysis of the lake water will be performed. Semi-annual samples required under the graded program are to be taken at times when the biologic cycle indicates that the concentrations of radionuclides should be the highest.

"(2) Procedures require sampling of each waste batch prior to release to the discharge canal. This procedure is backed up by the radiation monitors in the line from the waste disposal tanks to the discharge canal. The hi hi alarm point shall be set on these monitors such that they will warn of a higher than appropriate MPC in the discharge canal. In the event of the hi-hi alarm, the discharge shall cease until the cause is corrected. In the event the effluent monitor is out of service, two independent samples of each waste batch shall be taken and two Station personnel will independently check valving prior to discharge of liquid waste batches."

Specification 3.6.1d and 4.6.1.d - Page 89

Add new Specification 3.6.1.d. under "Limiting Condition for Operation" to read as follows:

"d. General

It is expected that releases of radioactive material in effluents will be kept at small fractions of the limits specified in Section 20.106 of 10 CFR Part 20. At the same time, the licensee is permitted the flexibility of operation, compatible with considerations of health and safety, to assure that the public is provided a dependable source of power even under unusual operating conditions, which may temporarily result in releases higher than such small fractions, but still within the limits specified in Section 20.106 of 10 CFR Part 20. It is expected that in using this operational flexibility under unusual operating conditions the licensee will exert his best efforts to keep levels of radioactive material in effluents as low as practical."

Add new Specification 4.6.1.d. under "Surveillance Requirement" to read as follows:

"d. General

Operating procedures shall be developed and used, and equipment which has been installed to maintain control of radioactive materials in gaseous and liquid effluents produced during normal reactor operations, including expected operational occurrences, shall be maintained and used to keep levels of radioactive material in effluents released to unrestricted areas as low as practical. The environmental monitoring program specified in Table 4.6.1 shall be conducted."

"A report shall be submitted to the Commission at the end of each six-month period of operation specifying total quantities of radioactive material released to unrestricted areas in liquid and gaseous effluents during the previous six months and such other information on releases as may be required to estimate exposures to the public resulting from effluent releases. If quantities of radioactive material released during the reporting period are unusual for normal reactor operations, including expected operation occurrences, the report shall cover this specifically. On the basis of such reports and any additional information the Commission may obtain from the licensee or others, the Commission may from time to time require the licensee to take such action as the Commission deems appropriate."

Add new Table 4.6.1 as follows:

TABLE 4.6.1

SAMPLE COLLECTION AND ANALYSIS

NINE MILE NUCLEAR POWER STATION - ENVIRONMENTAL MONITORING PROGRAM

A. Lake Program (Described in Appendix D-4 of the FSAR)

	Type of Sample	Type of Analysis	Collection Frequency	Number of Locations
1.	Fish	GB and Sr-90	Spring and Fall	Two
2.	Clams	GB, GSA, Sr-90	Spring and Fall	Two
3.	Gammarus (Fresh Water Shrimp)	GB, GSA, Sr-90	Spring and Fall	Two
4.	Lake Water	GB, GSA	Weekly	Downstream of Effluent Discharge

Coding: GB - gross beta
GSA - gamma spectral analysis

Notes on Graded Program:

A. No environmental lake program for effluent discharged at less than 1×10^{-8} uCi/ml average concentration.

B. Standard environmental lake program as shown for items 1 thru 3 for effluent discharged between 1×10^{-8} to 1×10^{-7} uCi/ml average concentration.

C. Standard environmental lake program as shown for items 1 thru 4 for effluent discharged above 1×10^{-7} uCi/ml but less than MPC in accordance with Appendix B, Table II, Column 2, of 10 CFR 20 and note 1 thereto.

D. An appropriate number of samples shall be taken at each location.

TABLE 4.6.1 (continued)

B. Land Program [Eleven sampling stations (5 onsite and 6 offsite) are employed as described in Appendix D-4 of FSAR.]

	Type of Sample	Type of Analysis	Collection Frequency	Number of Stations	Location
1.	Air Particulates	GSA (monthly) GB - all (24 hrs decay)	Weekly	Eleven	5 onsite 6 offsite
2.	Precipitation	GB & GSA	Monthly	Eleven	5 onsite 6 offsite
3.	Film Badges	Gross Gamma	Monthly	Eleven	5 onsite 6 offsite
4.	Radiation Monitors	Gross Gamma	Continuous	Six	5 onsite 1 offsite
5.	Farm Milk	Gross Beta, SR-90, I-131	Monthly	Adjacent Dairy Herds	Plant vicinity
6.	Airborne Halogens	GSA	Weekly	Eleven	5 onsite 6 offsite

Coding: GSA - Gamma spectral analysis
GB - gross beta
GB & GSA - gross beta and gamma spectral analysis

Notes on Graded Program:

A. No environmental land program for stack releases less than approximately 3 percent of maximum release rate.

B. Standard environmental land program as shown for items 1 thru 5 for stack releases between approximately 3 to 10 percent of maximum release rate.

C. Standard environmental land program as shown for items 1 thru 6 plus weekly for farm milk samples for stack releases between 10 to 30 percent of maximum release rate.

D. Environmental land program upgraded to twice weekly onsite for item 1, weekly onsite for item 2, bi-monthly onsite for item 3 and weekly for item 5 for stack releases greater than approximately 30 percent of maximum release rate.

E. After substantiating data is analyzed for any of the release rate levels, the environmental land program is degraded by one level, i.e., B. to A., C. to B. and D. to C.

Specification 3.6.2, 4.6.2 and Bases - pages 90, 92

Add Specification 4.6.2.c. under Surveillance Requirement to read as follows:

"c. At least daily during reactor power operation, the reactor neutron flux peaking factor shall be estimated and the flow-referenced APRM scram and rod block signals shall be adjusted, if necessary, as specified in Figure 2.1.2 Revised."

On page 92, add the paragraph which follows as the fifth paragraph of the Bases:

"The set points on the generator load rejection and turbine stop valve closure scram trips are set to anticipate and minimize the consequences of turbine trip with failure of the turbine bypass system as described in the bases for Specification 2.1.2. Since the severity of the transients is dependent on the reactor operating power level, bypassing of the scrams below the specified power level is permissible."

Change the last sentence under the Bases beginning on page 92 and ending in the first line of page 93 to read as follows:

"These errors are compensated for in the transient analyses by conservatism in the controlling parameter assumptions as discussed in the bases for Specification 2.1.2."

Tables 3.6.2a and 4.6.2a - Page 95

"3.6.2a - Limiting Condition for Operation, Parameter (2) High Reactor Pressure - Replace the set point of ' ≤1050 psig' with ' ≤ 1080 psig'."

Tables 3.6.2a and 4.6.2a - Page 96

"3.6.2a - Limiting Condition for Operation and 4.6.2a - Surveillance Requirement - Add Parameters 11 and 12 and notes (i) and (j) to Tables 3.6.2a and 4.6.2a", as shown below:

SEE PAGE 40a ATTACHED

Tables 3.6.2C and 4.6.2.C - Page 99

"3.6.2.C - Limiting Condition for Operation, Parameter (1) High-High Reactor Pressure - Replace the set point of ' ≤1060 psig' with ' ≤ 1090 psig'."

Tables 3.6.2.h and 4.6.2.h - Page 106

INSTRUMENTATION THAT INITIATES SCRAM

Parameter	3.6.2a - Limiting Condition for Operations							4.6.2a - Surveillance Requirement		
	Minimum No. of Tripped or Operable Trip Systems	Minimum No. of Operable Instrument Channels per Operable Trip System	Set Point	Reactor Mode Switch Position In Which Function Must be Operable: Shutdown	Refuel	Startup	Run	Sensor Check	Instrument Channel Test	Instrument Channel Calibration
(11) Turbine Stop Valve Closure	2	4	≤10% valve closure				(i)	None	Once per 3 months	None
(12) Generator Load Rejection	2	2	(j)				(i)	None	Once per month	Once per 3 months

Additional Notes for Tables 3.6.2a and 4.6.2a

(i) May be bypassed when reactor power level is below 45%.

(j) Trip upon loss of oil pressure to the acceleration relay.

3.6.2.h - Limiting Condition for Operation, Parameter (1) a. Upscale-Replace the set point of "$\leq$ 2.13 Ci/sec" with " $\leq$0.57 Ci/sec".

Specification 3.6.3 - pages 109, 110

Delete the words "or a written report shall be submitted to the Atomic Energy Commission" in each item listed below of Specification 3.6.3 under Limiting Condition for Operation as follows:

b. Lines 5 and 6.

c. Lines 5 and 6; again in lines 10 and 11.

d. Lines 2, 3 and 4.

h. Lines 2, 3 and 4.

On page 109 under Limiting Condition for Operation column, Specification 3.6.3.b in line 4, change the words from "within 24 hours" to "within 7 days".

SECTION 6 - ADMINISTRATIVE CONTROLS

Section 6.1 - Pages 118 through 124

Page 118 - Change the second paragraph from top of page to read as follows:

"In the absence of the Chairman, an acting Chairman who meets the education and experience qualifications will be designated."

Page 118 - Change the title "Executive Engineer" to "Chief Engineer" in line 1 of the third paragraph from the top of the page.

Page 118 - Under item (3) (b), Change "Chief System Project Engineer - Chairman" to "Staff Engineer or Manager - Chairman".

Page 118 - Delete the word "management" from the last line on the page.

Page 119 - Second line from the top of page, Change "Chief Nuclear Engineer" to "Staff Engineer - Nuclear".

Page 119 - Line 6 from top of page, Change "Chief Mechanical Engineer" to "Staff Engineer - Mechanical or Electrical".

Page 119 - Line 10 from top of page, Change "Group Head - Reactor Engineering" to "Staff Engineer - Environment".

Page 122 - Organization Chart top block, Change "Executive Engineer" to "Chief Engineer".

Page 122 - Organization chart block titled Safety Review & Audit Board, Change titles as follows:

From	To
Chief System Project Engr. Chm.	Staff Engineer or Manager - Chm.
Chief Nuclear Engr.	Staff Engineer Nuclear
Chief Mechanical Engr. Group Head - Reactor Engr.	Staff Engineer - Mech. or Electrical
Group Head - Environmental Engr.	Staff Engineer - Environment

Page 123 - Under column headed Safety Review & Audit Board, Change the title "Exec Engr." to "Chief Engr." as is appears in item 1, line 4; repeat item 1, line 4; item 2, line 6; and make the same change in the right-hand column heading.

Page 124 - Under column headed Safety Review & Audit Board, Change the title "Exec. Engr." to "Chief Engr." in line 3 of the last paragraph; and make the same change in the right-hand column heading.

Section 6.7 Reporting Requirements - pages 128-130

Replace Section 6.7 in its entirety with the following:

6.7 Reporting Requirements

In addition to reports required by applicable regulations, the following information shall also be provided:

6.7.1 Events requiring reports within 24 hours by telephone or telegraph to Region I Compliance Office followed by a written report within 10 days to the Director, Division of Reactor Licensing, USAEC, Washington, D. C. 20545; with a copy to Region I Compliance Office. The written report, and to the extent possible the preliminary telephone or telegraph report, shall describe, analyze and evaluate safety implications, and outline the corrective actions and measures taken or planned to prevent recurrence of a., b. and c., below:

a. Any significant variation of measured values of thermal, nuclear or hydraulic characteristics from a corresponding predicted value.

b. Any abnormal occurrences as specified in the Definitions Section of these specifications.

c. Incidents or conditions which resulted in a safety limit established in these Specifications being exceeded.

6.7.2 Events requiring reports within 30 days (in writing to the Director, Division of Reactor Licensing, USAEC, Washington, D. C. 20545; with a copy of Region I Compliance Office):

a. Any change in transient or accident analyses, as described in the Final Safety Analysis Report, which involves an unreviewed safety question as defined in Section 50.59(c) of 10 CFR 50.

b. Any changes in plant operating organization which involve positions for which minimum qualifications are specified in the Technical Specifications, or in personnel assigned to these positions.

6.7.3 Routine Operating Reports (in writing to the Director, Division of Reactor Licensing, Washington, D. C. 20545):

1. A routine operating report shall be prepared for each six-month period to January 1 and July 1 of each year. Such reports are to be submitted within 60 days after the end of each reporting period. The following information shall be provided (summarized on a monthly bases):

a. Nuclear

(1) Number of hours the plant was operated.

(2) Number of times the reactor was made critical.

(3) Gross thermal power generated.

(4) Operating histogram, showing the thermal power level of the reactor versus time for the report period.

(5) Equivalent Full Power Hours.

b. Electrical

(1) Gross power generated (in MWh).

(2) Net power generated (in MWh).

(3) Length of time generator was on line (in hours).

c. Shutdowns

(1) Number of scrams and shutdowns.

(2) Duration of down time (in hours).

(3) Reasons for outage.

d. Maintenance (on systems or components designed to prevent or mitigate the consequences of nuclear accidents)

(1) Nature of the maintenance; e.g., routine, emergency, preventive, or corrective.

(2) The effect, if any, on the safe operation of the reactor.

(3) The cause of any malfunction for which corrective maintenance was required.

(4) The effects of any such malfunctions.

(5) Corrective and preventive action taken to preclude recurrence of malfunctions.

(6) Time required for completion.

e. Radioactive Liquid Waste

(1) Total curie activity discharged.

(2) Total volume (in gallons before dilution) of liquid waste discharged.

(3) Total volume (in gallons) of dilution water used.

(4) Average concentration (in $\mu C/cc$) at point of discharge.

(5) Maximum concentration released for any day during the reporting period, including time and date.

(6) Percentage of Technical Specification limit released.

(7) Results of required isotopic analyses and estimated curies of each identified nuclide released.

(8) Total curie activity of tritium discharged.

f. Gaseous Waste

(1) Total curies activity discharged separated into noble gases, iodine, and particulates.

(2) Maximum activity released for any day during the reporting period, including time and date.

(3) Percentage of Technical Specification limit released and MPC value.

(4) Results of required isotopic analyses and estimated total curies of each identified nuclide released.

g. Solid Radioactive Waste

(1) Total volume (in cubic feet) of solid waste generated.

(2) Gross curie activity involved.

(3) Dates and disposition of the material if shipped off-site.

h. Evnironmental Monitoring

(1) For each medium sampled during the six-month period, the following information shall be provided:

a. Number of sampling locations.

b. Total number of samples.

c. Number of locations at which levels are found to be at least 10 percent above local backgrounds.

d. Highest, lowest and the annual average concentrations or levels of radiation for the sampling point with the highest average and description of the location of that point with respect to the site.

(2) If levels of radioactive material in environmental media indicate the likelihood of public intakes in excess of 3 percent of those that could result from continuous exposure to the concentration values listed in Appendix B, Table II, Part 20, estimates of the likely resultant exposure to individuals and to population groups and assumptions upon which estimates are based shall be provided. (These values are comparable to the top of Range I as defined in FRC Report No. 2.)

(3) If offsite environmental concentrations are observed which are greater than normal background fluctuations, correlation of these results with effluent releases shall be provided.

<u>Section 6.8 Special Reports - Page 130 - 132</u>

In the first line of Subsection 6.8.b, insert the words "as required in Section 6.7 above" between "report" and "shall".

Delete Subsection d. including items (1), (2), (3) and (4) on Page 131.

Reidentify Subsection "e" on Page 132 as Subsection "d".

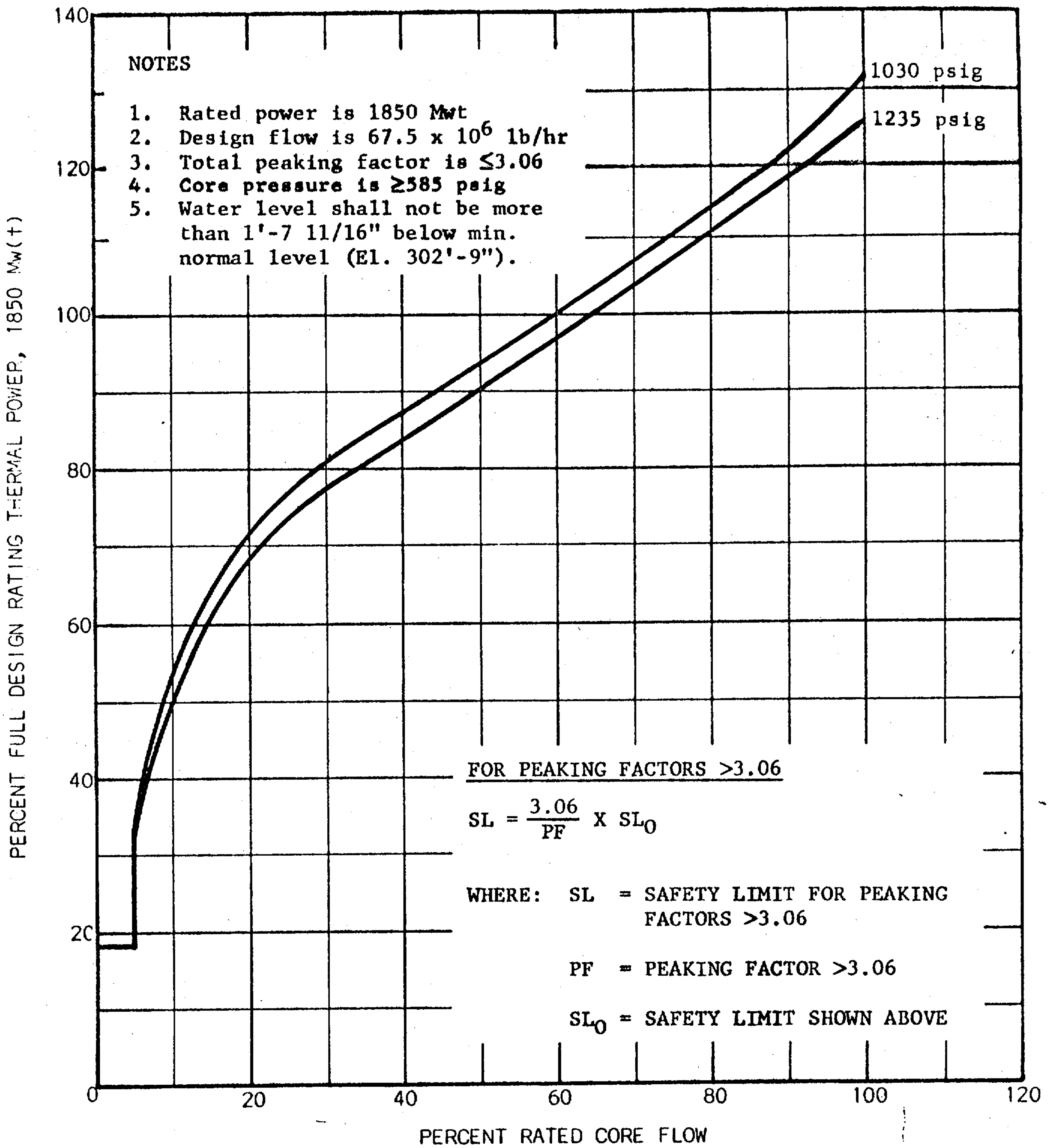
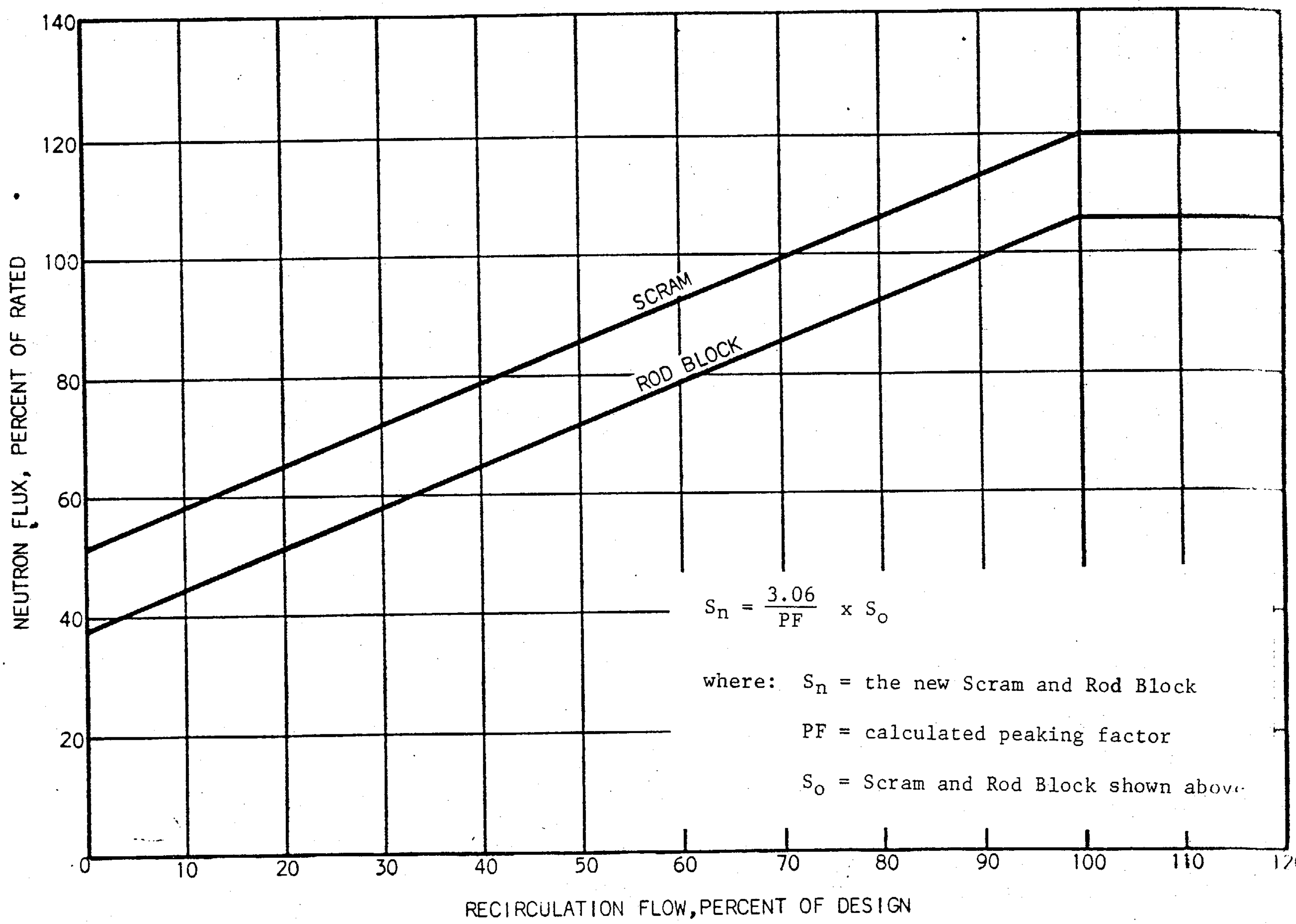
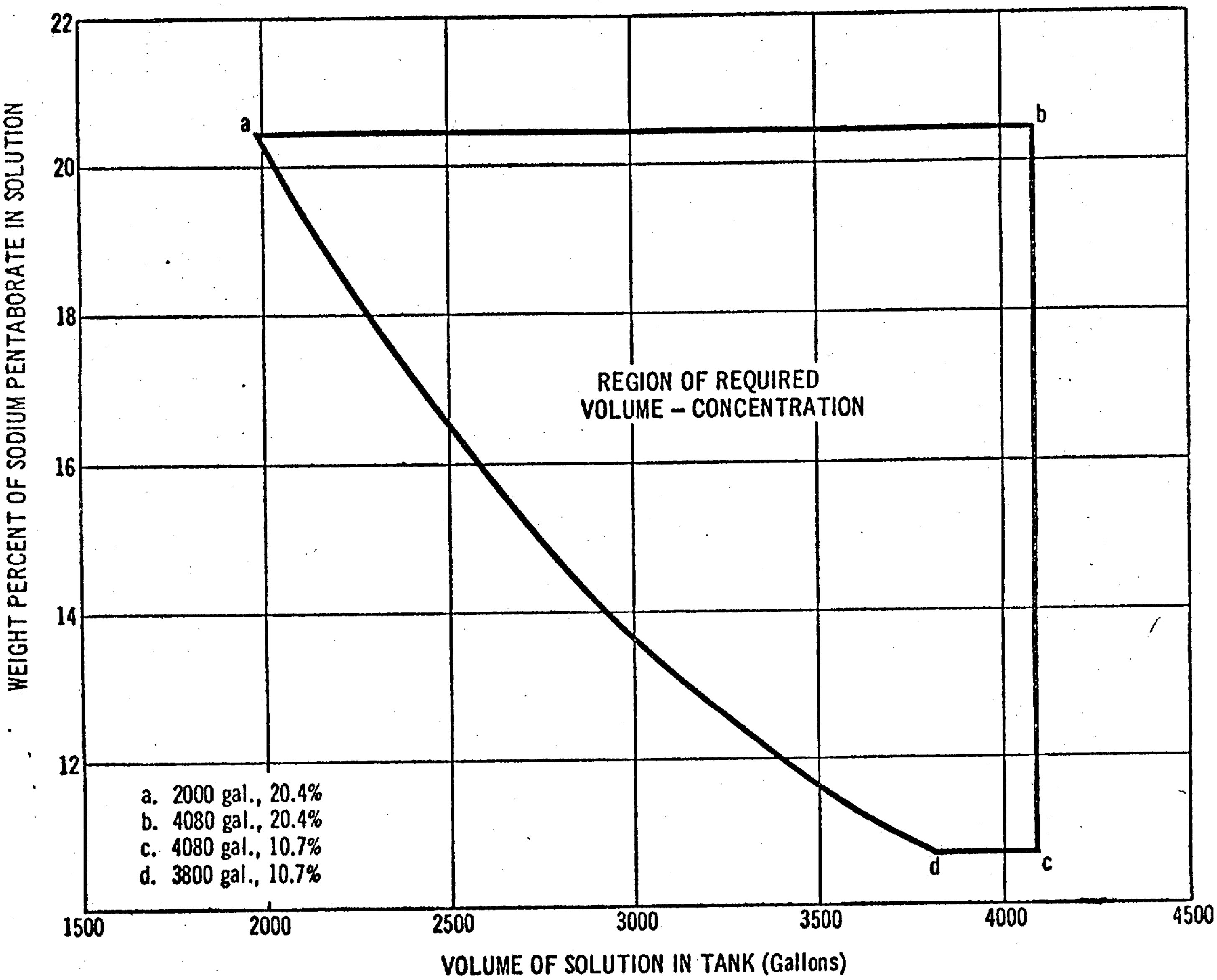

NOTES
1. Rated power is 1850 Mwt
2. Design flow is 67.5 x 10^6 lb/hr
3. Total peaking factor is ≤3.06
4. Core pressure is ≥585 psig
5. Water level shall not be more than 1'-7 11/16" below min. normal level (El. 302'-9").
1030 psig
1235 psig
FOR PEAKING FACTORS >3.06
SL = 3.06/PF X SL0
WHERE: SL = SAFETY LIMIT FOR PEAKING FACTORS >3.06
PF = PEAKING FACTOR >3.06
SL0 = SAFETY LIMIT SHOWN ABOVE
PERCENT FULL DESIGN RATING THERMAL POWER, 1850 Mw(t)
PERCENT RATED CORE FLOW
0
20
40
60
80
100
120
140

FIGURE 2.1.2 Revised

FLOW BIASED SCRAM AND APRM ROD BLOCK




22
20
18
16
14
12
a
b
c
d
REGION OF REQUIRED
VOLUME – CONCENTRATION
a. 2000 gal., 20.4%
b. 4080 gal., 20.4%
c. 4080 gal., 10.7%
d. 3800 gal., 10.7%
1500
2000
2500
3000
3500
4000
4500
WEIGHT PERCENT OF SODIUM PENTABORATE IN SOLUTION
VOLUME OF SOLUTION IN TANK (Gallons)

UNITED STATES ATOMIC ENERGY COMMISSION

SAFETY EVALUATION BY THE DIVISION OF REACTOR LICENSING

DOCKET NO. 50-220

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR STATION

POWER INCREASE

Date: March 3, 1971

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1.0 INTRODUCTION

On August 22, 1969, the Atomic Energy Commission issued Provisional Operating License No. DPR-17 to Niagara Mohawk Power Corporation authorizing operation of the Nine Mile Point (NMP) facility at steady-state power levels up to 1538 MWt. By application dated April 20, 1970, the Niagara Mohawk Power Corporation requested an amendment of its license to permit operation at steady-state power levels up to 1850 MWt.

Initial criticality of NMP was achieved on September 5, 1969, full licensed power level (1538 MWt) was reached on January 19, 1970, and the 100-hour full power demonstration run was completed on February 7, 1970.

The NMP facility was designed initially to operate at a power level of 1779 MWt, but with minor modifications the facility could be operated at a power level of 1850 MWt. The Niagara Mohawk Power Corporation chose to operate the plant at 1538 MWt for a period of time during which the performance of the plant would be evaluated.

Our review of the application for a provisional operating license was based on the 1538 MWt power level; however, this evaluation considered the capability of the plant engineered safety features and radiological consequences of accidents at the stretch rating of 1779 MWt.

We have evaluated the NMP facility for operation at power levels up to 1850 MWt with the present core loading. This evaluation is based on review of: "Technical Supplement to Petition to Increase Power Level - Nine Mile Point Nuclear Station" dated April 1970 and five amendments thereto, two dated October 1970, two dated December 1970, one dated January 1971, and a letter dated November 23, 1970, revising Amendment No. 2, submitted by Niagara Mohawk Power Corporation in support of its application to increase power; review of operations at power levels up to 1538 MWt; and review of the startup test program results. The application for the proposed increase in power to 1850 MWt included analysis of core thermal performance using the Hench-Levy heat transfer correlation rather than the Janssen-Levy correlation previously used, and included proposed minor modifications to the plant.

We have examined the reanalyses provided by the applicant of all anticipated operational transients affected by the power increase that might be expected to result from any single operator error or equipment malfunction. The results show that the design and performance objectives will be satisfied at the proposed power level of 1850 MWt. In addition, the design basis accidents have been reexamined for the higher power level. The radiological doses calculated to result from these accidents at the proposed power level of 1850 MWt will not be increased because the limits on the permitted primary coolant activity will remain unchanged and because the allowable containment leak rate will be reduced for operation at the higher power level.

The Advisory Committee on Reactor Safeguards (ACRS) completed its review of the application for operation at power levels up to 1850 MWt at its 130th meeting (February 4-6, 1971). A copy of the ACRS report is attached.

On the basis of our review, we have concluded that there is reasonable assurance that the health and safety of the public will not be endangered by the operation of Nine Mile Point Nuclear Station at steady-state power levels up to a maximum of 1850 MWt.

2.0 SITE AND STRUCTURES

2.1 General

The calculated consequences of four design basis accidents, as presented in the Final Safety Analysis Report (FSAR) and evaluated prior to issuance of the provisional operating license, were based on a power level of 1779 MWt. These accidents have been reanalyzed for operation at a power level of 1850 MWt. Because an appropriate reduction in the allowable containment leakage rate has been specified by the applicant, the potential calculated radiological doses are not changed from values previously determined to be acceptable. Therefore, we have concluded that the present site is suitable for operation of the NMP facility at power levels up to 1850 MWt.

2.2 Effluent Releases

The concentrations of radioactive effluents released from the NMP facility have been well below the limits set forth in the Technical Specifications. The liquid waste activity released represents less than 0.1% of the 10 CFR Part 20 concentration limits for individual isotopes. The total gaseous activity discharged represents less than 0.01% of the corresponding 10 CFR 20 concentration limits. The effluent radioactivity levels are not expected to increase significantly if the power level is increased to 1850 MWt.

2.3 Structures

At the time of our review of NMP for a provisional operating license, we informed Niagara Mohawk that it should install a strong-motion seismograph at the facility. Subsequently, we were informed by Niagara Mohawk that a strong-motion seismograph would be installed at NMP prior to the first major refueling outage. We consider that this matter is being resolved satisfactorily.

The biological shield consists of an approximately 24-foot diameter cylinder attached to the reactor vessel support pedestal and extending upward about 45 feet. The capability of this shield to withstand the pressure that could be developed as a result of failure of a nozzle safe end has been analyzed by the licensee and submitted in the Second Addendum dated October 1970. Based on our evaluation of this information, we consider the biological shield wall to have been adequately designed.

Additional studies by the applicant of the effects of dropping a fuel cask into the fuel storage pool have revealed the possibility of damage to the pool structures. Niagara Mohawk is investigating methods for preventing such pool damage and they have stated that the appropriate corrective measures will be taken when the investigation has been completed and the appropriate action determined. Methods being considered for protecting the fuel storage pool include: (1) crane interlocks and administrative controls to minimize the potential of dropping a cask in the pool and (2) installation of energy-absorbing material to cushion the impact of a falling cask.

3.0 REACTOR CORE CHARACTERISTICS

3.1 Core Performance

The core thermal and hydraulic performances were evaluated at various power levels during the startup testing program. Tests were performed at power levels of 387, 786, 1125, 1283, 1495 and 1538 MWt. Based on the data obtained at the 1538 MWt power level from the Local Power Range Monitor (LPRM) located nearest to the point of maximum heat flux, the licensee calculates a heat flux of 281,000 Btu/hr-ft^2 for the hottest rod. A Minimum Critical Heat Flux Ratio (MCHFR) of 2.58 at 120% power and a total peaking factor of 2.60 were calculated from these data. The corresponding design values given in the FSAR are 299,000 Btu/hr-ft^2, MCHFR $\geq$ 1.5 at 120% power, and total peaking factor of 3.08; thus, the results demonstrate that the reactor core operated within the thermal and hydraulic limits on which the Technical Specifications are based.

3.2 Thermal and Hydraulic Analysis for 1850 MWt

Operation of the NMP reactor at 1850 MWt with rated circulation flow results in thermal and hydraulic core parameters equivalent to those of Dresden Unit 2 (Docket No. 50-237). The increased reactor power rating is achieved primarily by using the more recent Hench-Levy heat transfer correlation (APED-5286, September 1966) in lieu of the Janssen-Levy correlation (APED-3892, April 1962). This change in heat transfer correlation results in a change of the thermal-hydraulic limits principally for recirculation flow rates in the range from 50% through 100% of rated; however, the margin in power between the limiting safety system settings and the safety limits within this flow range remain nearly the same for 1850 MWt as they were for 1538 MWt. For flow rates between 20% and 50% of rated, these margins are reduced slightly; however, the margins over this range are at least as large as those for flow rates between 50% and 100% of rated. We reviewed the use of the Hench-Levy heat transfer correlation in connection with our evaluation of the application for an operating license for Dresden Unit 2 and found it to be acceptable. Because of the design similarities between Dresden Unit 2 and the Nine Mile Point reactor, application of the Hench-Levy correlation in the analysis of Nine Mile Point is acceptable.

Table I compares the thermal and hydraulic data for NMP at 1538 and 1850 MWt with those of Dresden Unit 2 at 2527 MWt which has been evaluated and accepted. The comparison shows the core parameter values for NMP at 1850 MWt to be equivalent to those of Dresden Unit 2 and, therefore, the proposed operation of NMP represents no extension of previously approved BWR thermal and hydraulic operating limits.

TABLE I

COMPARISON OF NMP WITH DRESDEN-2
THERMAL AND HYDRAULIC DESIGN PARAMETERS

	Nine Mile Point		
Parameter	Current Rating	Design Rating	Dresden-2
Power level, MWt	1538	1850	2527
No. of fuel bundles	532	532	724
MCHFR	1.5 @ 120% power	1.9 @ 100% power	1.9 @ 100% power
Heat transfer correlation	Janssen-Levy	Hench-Levy	Hench-Levy
Average power density, kW/liter	34.1	41	41
Maximum linear heat generation rate, kW/ft	13.1	17.5	17.5
Maximum heat flux, 10^3 Btu/hr-ft^2	335	400	405
Maximum center fuel temperature, °F	3600	4250	4530
Peaking factors			
Local	1.30	1.30	1.30
Axial	1.57	1.57	1.57
Radial	1.51	1.50	1.50
Total	3.08	3.06	3.06
Primary system pressure, psig	1000	1030	1000

3.3 Reactivity Control

The equipment and systems for reactivity control of the NMP reactor, such as control rods, control rod drives and hydraulic system, and standby liquid control system, are not significantly affected by the proposed increase of power level. The shutdown margin was demonstrated to meet the requirements of the Technical Specifications during startup testing. In addition, measurements indicated that the maximum control rod notch worth is less than the permissible 0.1 percent delta k per notch.

4.0 PRIMARY COOLANT SYSTEM

Our review of the reactor primary coolant system included evaluation of the effect of the proposed power increase on the reactor vessel, on the adequacy of the number of safety valves and on the operating performance of the primary coolant system. In evaluating the operating performance of the primary coolant system, we considered recent experience with furnace-sensitized stainless steel components, the inservice inspection program, the structural integrity of the biological shield, and the current status of primary system leak detection capability.

4.1 Reactor Vessel

The reactor vessel is made of SA-302, Grade B, carbon steel and the calculated lifetime neutron fluence on the vessel wall is reported as 5×10^{-17} (energies greater than 1 meV). The proposed power increase from 1538 to 1850 MWt would change the estimated nil ductility transition temperatures from 40 to 42°F for the lower bound and from 150 to 160°F for the upper bound. We have concluded that this change is negligible considering the uncertainties inherent in these estimates.

4.2 Safety and Relief Valves

The transients associated with operation of the safety and relief valves have been reanalyzed by the licensee for a power level of 1850 MWt. The transient analysis of safety valve actuation shows that the addition of a sixteenth safety valve is necessary to meet the original design criteria at the proposed 1850 MWt power level. The applicant will install an additional safety valve. Since the basis for sizing the safety valves has not changed, we have concluded that the safety and relief valves provided for 1850 MWt operation are acceptable for their intended purpose.

4.3 Primary Coolant System Piping

On March 6, 1970, during a routine inspection of the drywell, primary coolant leakage was observed from the west core spray nozzle safe end. The ensuing investigation, which showed that the leak resulted from cracks in the safe end, included evaluations of materials, stress analyses and metallurgical examinations. We reviewed the results of Niagara Mohawk's investigation and they were discussed with the ACRS at its 122nd meeting (June 11-13, 1970) prior to resumption of operation of NMP. During its review, the ACRS made recommendations regarding primary system leak detection, discussed below, and biological shield integrity as discussed in Section 2.3.

Systems are presently installed to detect reactor coolant system leakage within the primary containment by measurement of the sump liquid accumulation rate and the dew point of the containment atmosphere. A third leak detection system will be installed. This system will recirculate a portion of the primary containment atmosphere through an external loop by a positive displacement type blower. Samples will be drawn continuously from this loop through an air radiation monitor having a belt-type filter and an alarm. The applicant has informed us that delivery of the equipment for this system is expected by early 1971 and that installation will be accomplished at the first convenient outage following delivery.

The current operational experience with the two functioning leak detection systems indicates that the sump accumulation rate method will detect leak rates as low as about 0.5 gpm and the dew point system may detect rates of 0.5 to 1.0 gpm.

5.0 EMERGENCY CORE COOLING

Our evaluation of the efficacy of the Emergency Core Cooling System (ECCS) for the NMP plant at 1850 MWt included consideration of the results of the Full Length Emergency Cooling Heat Transfer (FLECHT) Boiling Water Reactor (BWR) experiment program. The total power levels and some of the spray initiation temperatures obtained in the FLECHT tests are somewhat less than those expected in the Nine Mile Point core at the proposed power level, although the linear power density is comparable. Because of the differences between the conditions of the tests and of the NMP reactor at 1850 MWt, our evaluation of the NMP ECCS performance was based on the use of analytical models developed by correlation of the full range of experimental data available.

The licensee has submitted calculations of ECCS performance using an analytical model developed by the General Electric Company (GE). Our consultant, the Idaho Nuclear Corporation (INC), has developed an analytical model independently also based on the FLECHT experimental results. Because of the complexity of the phenomena involved in the modeling of the spray cooling phase of a loss-of-coolant accident, differences exist between the two models. We are continuing to evaluate the available analytical models as refinements are developed.

For loss-of-coolant accidents resulting from the postulated rupture of large pipes, the calculated peak fuel rod cladding temperatures are dependent upon the time required for achieving rated flow of the core spray system. In earlier calculations, a time of 60 seconds was assumed. The licensee proposes to revise time settings on the emergency power system so as to reduce this core spray initiation time to 35 seconds or less. For an assumed initiation time of 35 seconds, we have concluded that the peak clad temperature calculated by either the GE or the INC calculational model will not exceed 2300°F for accidents resulting from the break of a recirculation line.

In Addendum No. 5 to the application for power increase, the applicant has submitted calculations for the loss-of-coolant-accident resulting from postulated small breaks of coolant lines. The calculated peak clad temperature is also below 2300°F.

On the basis of our evaluations, we conclude that for power levels up to 1850 MWt and with a spray initiation time of 35 seconds or less, the ECCS for the NMP reactor will: (a) limit the peak clad temperature to less than 2300°F, which is well below the clad melting temperature, (b) limit the fuel clad-water reaction to less than one percent of the total clad mass, (c) terminate the temperature transient before the core geometry necessary for core cooling is lost and before the cladding is so embrittled as to fail upon quenching, and (d) reduce the core temperature and remove core decay heat for an extended period of time, for the entire spectrum of postulated break sizes including the double-ended break of a recirculation line.

6.0 INSTRUMENTATION AND CONTROL

6.1 Control Rod Drives

During the startup testing of all 129 control rod drives, it was determined that the rod withdrawal rates and the rod scram insertion times were within the Technical Specification limits. As the testing program progressed, the control rod scram insertion times were observed to increase, but they did not exceed the limits of the Technical Specifications. Because of this progressive increase in control rod scram insertion times the 400-mesh inner control rod drive screens were replaced with strainers having rectangular openings (10 mils by 50 mils).

During the shutdown for control rod drive screen replacement at NMP, deformations of the control rod drive index tubes were observed at the Oyster Creek reactor. To preclude a similar occurrence at NMP, the nitrogen precharge pressure on the accumulators was reduced from 800 psig to 575 psig (with concomitant reduction in operating accumulator pressure from 1400 psig to 1100 psig). On December 9, 1969, the NMP reactor was returned to operation, and measurements of the control rod scram insertion times showed that the time for 90% insertion of all rods averaged 2.75 seconds, well within the Technical Specification limit of 5 seconds. The licensee subsequently inaugurated a program of periodically measuring the scram times of eight selected control rods to ascertain if any deterioration of the system has occurred. This surveillance program has been incorporated as a requirement of the Technical Specifications. We have concluded that the control rod surveillance program is adequate to monitor control rod drive performance and that no new safety considerations, in this regard, are introduced by the proposed increase in power.

6.2 Addition of Turbine Trip and Load Rejection Scram

To reduce the severity of the transient that would result from a turbine trip in the event of failure of the bypass valves to open, two additional scram circuits will be installed. The scrams will be initiated by two sets of instruments: (1) a turbine trip scram taking a signal from the turbine stop valve closure, and (2) a load rejection scram taking a signal from the turbine control valve acceleration relay.

The startup testing program included turbine trip tests at 384, 768, 783 and 1538 MWt. The maximum heat flux associated with the turbine trip at 1538 MWt was 247,000 Btu/hr-ft^2, compared with the predicted 299,000 Btu/hr-ft^2; and the MCHFR was 1.96 @ 120% power compared with the predicted design MCHFR limit of 1.5. The comparison of the results of the turbine trip tests with predictions provide confidence in the applicant's analysis of the same transient at the higher power level of 1850 MWt and in the determination of the safety valve sizing as described in Section 4.2 of this report. To assure that the MCHFR limit will not be exceeded during a turbine trip transient at 1850 MWt, scrams will be initiated whenever the turbine stop valve reaches a position of greater than 10 percent closed and when acceleration relays indicate fast closure of the turbine valve. The applicant's analysis of the turbine trip transient shows that reactor scram would be initiated within 10 milliseconds and the resultant flux and pressure peaks in the reactor would be less severe than those associated with operation at 1538 MWt without these scrams.

We have determined that the turbine stop valve closure scram circuits satisfy the provisions of the Proposed IEEE Criteria for Nuclear Power Plant Protection Systems(IEEE-279). Further, this scram circuit and the turbine control valve fast closure scram circuit are independent of the existing reactor protection systems.

The load rejection scram provided by the turbine control valve fast closure acceleration relays is not immune to potential failures of single components; however, because this scram function merely anticipates the action of the redundant turbine trip circuits, we have concluded that its design is satisfactory for the purpose intended.

An automatic bypass circuit is to be added as part of the modification. This circuit will defeat the turbine trip and load rejection scrams at power levels below 45 percent of full power. Power is sensed as a function of turbine first-stage pressure by four pressure switches. We have determined that the pressure switches are redundant and that they are properly grouped to achieve adequate physical separation.

We have concluded that the proposed addition of scrams from turbine trip and load rejection is acceptable and that the instrumentation proposed to accomplish these actions is adequate.

7.0 ANALYSIS OF ACCIDENTS AND EXPECTED TRANSIENTS

7.1 Accident Analysis

The four major postulated accidents (design basis accidents: loss of primary system coolant, steam line break outside the drywell, drop of a fuel assembly during refueling, control rod drop) have been reanalyzed for the proposed power level of 1850 MWt. For the main steam line break accident, the offsite radiation doses depend upon the concentrations of radioactivity in the primary coolant system; thus, the increase in power level does not change the calculated radiological doses since the limit on primary coolant activity is not changed. For the other three design basis accidents, the calculated doses are directly proportional to reactor power level and the applicable containment leak rate. During our review of the provisional operating license application, we calculated that the maximum two-hour doses at the exclusion area boundary would result from the loss-of-coolant accident at 1779 MWt and were well below the guidelines of 10 CFR 100. The applicant has proposed reducing the technical specification limit for containment leakage rate such that estimated offsite doses will be no greater for operation at 1850 MWt than those originally calculated and accepted for 1779 MWt. The proposed containment leak rate reduction is from the present allowable 1.6%/day to 1.5%/day at 22 psig test pressure. The results of testing to date show a containment leak rate of about 0.8%/day which is well within the proposed technical specification limit. Our previous conclusion that the calculated radiological consequences of the design basis accidents were well within the 10 CFR 100 guidelines is unchanged.

7.2 Transient Analysis

The licensee has reanalyzed all anticipated operational transients affected by the power increase that might be expected for any single operator error or equipment malfunction. The results show that the design and performance objectives continue to be met for the proposed operation at 1850 MWt. The addition of scrams from turbine stop valve closure and from turbine control valve fast closure will reduce the pressure transient resulting from a turbine trip. We have concluded that the safety criteria regarding fuel damage limits applicable to current operation are satisfied and that these criteria will also be satisfied for operation at the proposed power level of 1850 MWt.

8.0 STARTUP AND POWER OPERATIONS

8.1 Startup and Power Testing at 1538 MWt

As indicated above, the results of the startup and power testing programs substantiated design predictions. The core thermal and hydraulic performance showed that the core operated within the specified thermal and hydraulic limits. The transients resulting from recirculation pump trips at one-half, three-fourths and full power were consistent with predictions. For example, the 5-pump trip test results at 1538 MWt showed a transient MCHFR of 2.3 at 2.75 seconds after the trip which is consistent with the prediction. For the turbine trips at similar powers, the data showed maximum heat flux and MCHFR values to be within applicable predictions and limits. Reactor system stability measurements were within applicable criteria. Control rod reactivity worth measurements and rod insertion scram times were satisfactory.

8.2 1850 MWt Power Test Program

The applicant proposes a power escalation test program similar to that performed during the initial approach to power. A set of base conditions will be measured at 1538 MWt before power escalation is initiated to serve as a basis for comparison with subsequent tests. These base conditions will include chemical and radioactivity levels at typical locations, radiation measurements, APRM calibrations, LPRM response characteristics, power distribution measurements and a core performance evaluation. During the power escalation program, these tests will be repeated at about 1700 MWt and again at the full-power level of 1850 MWt. Tests at 1850 MWt will include induced transients, such as a step change of the pressure regulator set point, opening of one turbine bypass valve, shutdown of recirculating pumps and turbine trip, to determine reactor response. We have concluded that the proposed tests will provide the information necessary to demonstrate the adequacy of the NMP facility to operate at the proposed power level of 1850 MWt.

9.0 TECHNICAL COMPETENCE

The operating organization, its qualifications and responsibilities, operating procedures, records, maintenance, and review and audit functions are not changed from those we found acceptable during the provisional operating license review. There has been only one change in the supervisory staff of the operating organization from that previously reviewed. The general technical performance of this staff has been shown to be satisfactory during the startup and power operations to date.

10.0 TECHNICAL SPECIFICATIONS

Several changes to the Technical Specifications will be necessary in connection with the proposed power increase. These changes involve references to power level or parameters associated with reactor pressure. In addition to the changes directly related to the power increase, the Technical Specifications have been updated in other areas by incorporation of the current requirements for effluent releases, testing of instrument line flow check valves, reactivity anomalies, environmental monitoring program and reporting requirements.

11.0 REPORT OF ADVISORY COMMITTEE ON REACTOR SAFEGUARDS (ACRS)

The ACRS has reviewed Niagara Mohawk's application for an increase in power level of the NMP reactor from 1538 MWt to 1850 MWt. The Committee completed its review during its 130th meeting held February 4-6, 1971. A copy of the ACRS letter, dated February 6, 1971, is attached.

The ACRS, in its letter, made comments regarding several matters which will be considered prior to and during operation of NMP at the increased power level. These matters have also been considered in our evaluation.

We will follow the licensee's implementation of the recommendations and response to the comments during operation of the facility.

The ACRS concluded in its letter that if due regard is given to its comments, the Nine Mile Point Nuclear Station can be operated at power levels up to 1850 MWt without undue hazard to the health and safety of the public.

12.0 CONCLUSION

Based upon our review of the application, and of relevant information regarding facility operation to date as discussed in this evaluation, we have concluded that there is reasonable assurance that the Nine Mile Point Nuclear Station can be operated at steady-state power levels up to a maximum of 1850 MWt without endangering the health and safety of the public.

Robert J. Schemel, for
Donald J. Skovholt
Assistant Director
for Reactor Operations
Division of Reactor Licensing

Attachment:
ACRS Report dtd. 2/6/71

ATTACHMENT TO ... RL SAFETY EVALUATION ON NMP PO... INCREASE

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

February 6, 1971

Honorable Glenn T. Seaborg
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON NINE MILE POINT NUCLEAR STATION

Dear Dr. Seaborg:

During its 128th meeting, December 10-12, 1970, and its 130th meeting, February 4-6, 1971, the Advisory Committee on Reactor Safeguards reviewed the application of the Niagara Mohawk Power Corporation for an increase in the licensed power level of the Nine Mile Point Nuclear Station from 1538 MW(t) to 1850 MW(t). The application was also considered at subcommittee meetings held in Washington, D. C. on December 9, 1970, and February 2, 1971. During its review, the Committee had the benefit of discussions with representatives of Niagara Mohawk Power Corporation, the General Electric Company, the AEC Regulatory Staff, and their consultants, and of the documents listed. The Committee previously reported to you on this project on June 16, 1970.

The proposed increase in power level is based in part on favorable preoperational test results and initial operating experience, and on use of an improved heat transfer correlation for evaluation of core thermal performance. Also, the normal reactor operating pressure will be increased from 1000 to 1030 psig, and a number of minor modifications to the plant will be made.

The applicant intends to install one additional safety valve (for a total of 16) on the reactor coolant system so as to meet at 1850 MW(t) the same design criterion for pressure relief as was met at the original power level.

Two new reactor scram trips will be added, one based on turbine stop valve closure and the other based on turbine control valve high rate of closure. Both trips will be operative at all power levels above 45 percent of full power, and are provided to assure that safety limits within the core are not exceeded during a transient resulting from turbine trip with assumed failure of the steam bypass valves to open.

Performance of the emergency core spray cooling system has been re-evaluated for 1850 MW(t) operation. The applicant proposes to revise time settings on the emergency power system so as to reduce core spray initiation time from 60 seconds to 35 seconds. With this change, and in light of results from the Commission's FLECHT Program, the core spray system appears acceptable for the proposed higher power operation. However, the Committee believes the applicant should continue to seek refinement in the models for evaluation of peak clad temperatures reached during postulated loss of coolant accidents. Also, confirmatory analyses currently underway by the Regulatory Staff should continue to be pursued.

Doses calculated for design basis accidents have also been reexamined for 1850 MW(t) operation. The applicant proposes to reduce the allowable containment leak rate from 1.6 to 1.5 percent per day (at 22 psig test pressure) and to maintain unchanged the existing primary coolant activity limits. With these provisions, the calculated doses based on the higher power level are no higher than those originally calculated for the stretch power rating of 1779 MW(t), and are within the 10 CFR 100 guidelines.

Further study by the applicant has indicated that adequate integrity of the spent fuel pool may not be assured in the postulated event of dropping of a fuel cask into the pool. Some possible corrective measures have been identified, and the applicant states that appropriate modifications to the plant will be made. The Regulatory Staff should follow this matter and assure implementation on an appropriate time scale.

The applicant has developed improved plans for in-service inspection of the main steam lines both inside and outside of containment. For piping beyond the second isolation valve, two welds in each pipe will be completely inspected by ultrasonic testing each year, with every such weld being so inspected at least once per eight years. This program will be initiated at the next plant outage.

Analyses by the applicant indicate that the biological shield surrounding the reactor can withstand satisfactorily the effects of failure of a reactor vessel safe end. The Regulatory Staff agrees with this conclusion.

The applicant has studied improved leak detection methods for use within the containment, and plans to supplement the existing systems. In addition to the sump accumulation rate and dew point measurement systems already in operation, he will install an atmospheric radioactivity monitoring system. This system will recirculate a portion of the containment atmosphere through an external loop and an air monitor. Installation is expected to be completed within a few months.

The Committee wishes to re-emphasize its belief that additional means for assuring continued reactor pressure vessel integrity, including possible improvement in access to the vessel surfaces for augmentation of in-service inspection, should be actively studied and implemented to the degree practical.

The applicant is actively studying means for control of buildup of hydrogen in the containment which might follow in the unlikely event of a loss of coolant accident. The Committee wishes to be kept informed of the resolution of this matter.

The applicant is continuing to study further means of preventing common failure modes from negating reactor scram action, and of design features to make tolerable the consequences of failure to scram during anticipated transients. The Committee wishes to be kept informed of the resolution of this matter.

The Advisory Committee on Reactor Safeguards believes that, if due regard is given to the items mentioned above and in its reports of April 17, 1969 and June 16, 1970, there is reasonable assurance that the Nine Mile Point Nuclear Station can be operated at power levels up to 1850 MW(t) without undue risk to the health and safety of the public.

Sincerely yours,

Spencer H. Bush

Spencer H. Bush
Chairman

References - Nine Mile Point Nuclear Station

1. Niagara Mohawk Power Corporation Petition Requesting Amendment of License dated April 20, 1970, with Technical Supplement to Increase Power Level.

2. First through Fifth Addenda to Technical Supplement to Increase Power Level.

3. Niagara Mohawk Power Corporation letter dated November 23, 1970, forwarding corrections to Second Addendum to Technical Supplement to Petition to Increase Power Level.