ATTACHMENT TO LICENSE AMENDMENT NO. 140

FACILITY OPERATING LICENSE NO. NPF-69

DOCKET NO. 50-410

Replace the following pages of the License with the attached revise pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
4	4
5	5
10	10
11	11
12	12
13	13
	14
	15
	16
	17

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

Remove	Insert
1.1-5	1.1-5
2.0-1	2.0-1
3.1.7-3	3.1.7-3
3.2.1-1	3.2.1-1
3.2.2-1	3.2.2-1
3.2.3-1	3.2.3-1
3.3.1.1-2	3.3.1.1-2
3.3.1.1-4	3.3.1.1-4
3.3.1.1-6	3.3.1.1-6
3.3.1.1-8	3.3.1.1-8
3.3.1.1-9	3.3.1.1-9
3.3.1.1-10	3.3.1.1-10
3.3.2.2-1	3.3.2.2-1
3.3.2.2-2	3.3.2.2-2
3.3.4.1-1	3.3.4.1-1
3.3.4.1-2	3.3.4.1-2
3.3.4.1-3	3.3.4.1-3
3.3.6.1-6	3.3.6.1-6
3.4.3-2	3.4.3-2
3.7.5-1	3.7.5-1
4.0-2	4.0-2
4.0-3	4.0-3
4.0-4	4.0-4

(1) Maximum Power Level

Nine Mile Point Nuclear Station, LLC is authorized to operate the facility at reactor core power levels not in excess of 3988 megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, both of which are attached hereto, as revised through Amendment No. 140 are hereby incorporated into this license. Nine Mile Point Nuclear Station, LLC shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) Fuel Storage and Handling (Section 9.1, SSER 4)*

a. Fuel assemblies, when stored in their shipping containers, shall be stacked no more than three containers high.

b. When not in the reactor vessel, no more than three fuel assemblies shall be allowed outside of their shipping containers or storage racks in the New Fuel Vault or Spent Fuel Storage Facility.

c. The above three fuel assemblies shall maintain a minimum edge-to-edge spacing of twelve (12) inches from the shipping container array and approved storage rack locations.

d. The New Fuel Storage Vault shall have no more than ten fresh fuel assemblies uncovered at any one time.

(4) Turbine System Maintenance Program (Section 3.5.1.3.10, SER)

The operating licensee shall submit for NRC approval by October 31, 1989, a turbine system maintenance program based on the manufacturer's calculations of missile generation probabilities. (Submitted by NMPC letter dated October 30, 1989 from C.D. Terry and approved by NRC letter dated March 15, 1990 from Robert Martin to Mr. Lawrence Burkhardt, III).

* The parenthetical notation following the title of many license conditions denotes the section of the Safety Evaluation Report (SER) and/or its supplements wherein the license condition is discussed.

(5) Inservice Inspection (Sections 5.2.4.3 and 6.6.3, SSER 5)

The operating licensee shall submit an inservice inspection program in accordance with 10 CFR 50.55a(g)(4) for staff review by July 31, 1987.

(6) Initial Startup Test Program (Section 14, SER, SSERs 4 and 5)

Any changes to the Initial Test Program described in Section 14 of the Final Safety Analysis Report made in accordance with the provisions of 10 CFR 50.59 shall be reported in accordance with 50.59(b) within one month of such change.

(7) Operation with Reduced Feedwater Temperature (Section 15.1, SSER 4)

Nine Mile Point Nuclear Station, LLC shall not operate the facility with reduced feedwater temperature for the purpose of extending the normal fuel cycle. The facility shall not be operated with a feedwater heating capacity less than that required to produce a feedwater temperature of 420.5 °F at rated steady-state conditions unless analyses supporting such operations are submitted by Nine Mile Point Nuclear Station, LLC and approved by the staff.

(8) Safety Parameter Display System (SPDS) (Section 18.2, SSERs 3 and 5)

Prior to startup following the first refueling outage, the operating licensee shall have operational an SPDS that includes the revisions described in their letter of November 19, 1985. Before declaring the SPDS operational, the operating licensee shall complete testing adequate to ensure that no safety concerns exist regarding the operation of the Nine Mile Point Nuclear Station, Unit No. 2 SPDS.

(9) Detailed Control Room Design Review (Section 18.1, SSERs 5 and 6)

(a) Deleted per Amendment No. 24 (12-18-90)

(b) Prior to startup following the first refueling outage, the operating licensee shall provide the results of the reevaluation of normally lit and nuisance alarms for NRC review in accordance with its August 21, 1986 letter.

(c) Prior to startup following the first refueling outage, the operating licensee shall complete permanent zone banding of meters in accordance with its August 4, 1986 letter.

(20) Potential Adverse Flow Effects

These license conditions provide for monitoring, evaluating, and taking prompt action in response to potential adverse flow effects as a result of power uprate operation on plant structures, systems, and components (including verifying the continued structural integrity of the steam dryer) for power ascension from CLTP (3467 MWt) to 120 percent OLTP (or 115 percent of CLTP) (3988 MWt) condition.

(a) The following requirements are placed on operation of the facility above the thermal power level of 3467 MWt for the power ascension from CLTP (3467 MWt):

1. NMPNS shall monitor the main steam line (MSL) strain gages during power ascension above 3467 MWt for increasing pressure fluctuations in the steam lines. While first increasing power above 3467 MWt, NMPNS shall collect data from the MSL strain gages at nominal 1 percent thermal power increments and evaluate steam dryer performance based on this data.

2. NMPNS shall hold the facility at 105 percent and 110 percent of 3467 MWt to collect data from the MSL strain gages required by Condition 1.a., conduct plant inspections and walkdowns, and evaluate steam dryer performance based on these data; shall provide the evaluation to the NRC staff by facsimile or electronic transmission to the NRC project manager upon completion of the evaluation; and shall not increase power above each hold point until 96 hours after the NRC project manager confirms receipt of the transmission.

3. During power ascension at each 2.5 percent power level above CLTP, the licensee shall perform stress analysis for the top 100 stress locations of the steam dryer using the applicable ACM 4.1 load definition and determine the minimum alternating stress ratio. The licensee shall confirm that this ratio is equal to or greater than the ratio based on the velocity-square relationship; otherwise, the licensee shall return the facility to a lower power level where the minimum alternating stress ratio satisfies the velocity-square relationship, and shall not further increase the power without approval from the NRC. A summary of the results shall be provided for NRC review at each 5 percent data review plateau. After completion of the full EPU test plateau (approximately 120 percent OLTP or 115 percent CLTP), the licensee shall provide the NRC a full startup test report and final stress analysis report within 90 days.

4. If any frequency peak from the MSL strain gage data exceeds the Level 1 limit curves, NMPNS shall return the facility to a power level at which the limit curve is not exceeded. NMPNS shall resolve the discrepancy, evaluate and document the continued structural integrity of the steam dryer, and provide that documentation by facsimile or electronic transmission to the NRC project manager prior to further increases in

reactor power, except when stress analysis is re-performed and new limit curves are developed. In that case, NMPNS shall not further increase power above each hold point until 96 hours after the NRC project manager confirms receipt of the transmission.

5. In addition to evaluating the MSL strain gage data, NMPNS shall monitor reactor pressure vessel water level instrumentation, and MSL piping accelerometers on an hourly basis during power ascension above 3467 MWt. If resonance frequencies are identified as increasing above nominal levels in proportion to strain gage instrumentation data, NMPNS shall stop power ascension, evaluate and document the continued structural integrity of the steam dryer, and provide that documentation to NRC staff by facsimile or electronic transmission to the NRC project manager prior to further increases in reactor power.

(b) NMPNS shall implement the following actions for the power ascension from CLTP (3467 MWt) to 120 percent OLTP (3988 MWt) condition.

1. In the event that acoustic signals (in MSL strain gage signals) are identified that challenge the limit curves during power ascension above 3467 MWt, NMPNS shall evaluate dryer loads, and stresses, including the effect of ±10 percent frequency shift, and re-establish the limit curves, and shall perform a frequency-specific assessment of ACM uncertainty at the acoustic signal frequency including application of 65 percent bias error and 10 percent uncertainty to all the SRV acoustic resonances. In the event that stress analyses are re-performed based on new strain gage data to address paragraph 1 above, the revised load definition, stress analysis, and limit curves shall include:

   (a) Application of 65 percent bias error and 10 percent uncertainty to all the SRV acoustic resonances.
   (b) Use of bump-up factors associated with all the SRV acoustic resonances and determined from the scale model test results.
   (c) Evaluation of the effect of ±10 percent frequency shifts in increments of 2.5 percent.

2. NMPNS shall incorporate in NMP2 steam dryer the design modifications identified in Section 2.2.6.1.2 of this SE before increasing the power above CLTP.

3. After reaching EPU conditions, NMPNS shall obtain measurements from the MSL strain gages and establish the steam dryer flow-induced vibration load fatigue margin for the facility, update the dryer stress report, and re-establish the limit curves with the updated ACM load definition, which will be provided to the NRC staff.

4. NMPNS shall revise plant procedures to reflect long-term monitoring of plant parameters potentially indicative of steam dryer failure; to reflect consistency of the facility's steam dryer inspection program with BWRVIP-139; and to identify the NRC project manager for the facility as the point of contact for providing power ascension testing information during power ascension.

5. NMPNS shall submit the final EPU steam dryer load definition for the facility to the NRC upon completion of the power ascension test program.

6. NMPNS shall submit the flow-induced vibration related portions of the EPU startup test procedure to the NRC, including methodology for updating the limit curve, prior to initial power ascension above 3467 MWt.

(c) NMPNS shall prepare the EPU startup test procedure to include:

1. The stress limit curves to be applied for evaluating steam dryer performance;

2. Specific hold points and their durations during EPU power ascension;

3. Activities to be accomplished during the hold points;

4. Plant parameters to be monitored;

5. Inspections and walkdowns to be conducted for steam, feedwater, and condensate systems and components during the hold points;

6. Methods to be used to trend plant parameters;

7. Acceptance criteria for monitoring and trending plant parameters, and conducting the walkdowns and inspections;

8. Actions to be taken if acceptance criteria are not satisfied; and

9. Verification of the completion of commitments and planned actions specified in its application and all supplements to the application in support of the EPU license amendment request pertaining to the steam dryer prior to power increase above 3467 MWt.

NMPNS shall provide the related EPU startup test procedure sections to the NRC by facsimile or electronic transmission to the NRC project manager prior to increasing power above 3467 MWt.

(d) The following key attributes of the program for verifying the continued structural integrity of the steam dryer shall not be made less restrictive without prior NRC approval:

1. During initial power ascension testing above 3467 MWt, each test plateau increment shall be approximately 5 percent of 3467 MWt.

2. Level 1 performance criteria; and

3. The methodology for establishing the limit curves used for the Level 1 and Level 2 performance

(e) The results of the power ascension testing to verify the continued structural integrity of the steam dryer and the final steam dryer load definition shall be submitted to the NRC staff in a report within 60 days following the completion of all 120 percent OLTP (EPU) power ascension testing.

(f) During the first two scheduled refueling outages after reaching 120 percent OLTP conditions, a visual inspection shall be conducted of all accessible, susceptible locations of the steam dryer in accordance with BWRVIP-139 inspection guidelines. In addition, a visual inspection of all accessible welds that were analyzed using embedded models shall be conducted. In addition, a visual inspection of the existing indications in the upper support ring, the drain channel to skirt weld, the tie bar-to-hood weld heat affected zone, and vertical support plates shall be conducted.

(g) The results of the visual inspections of the steam dryer shall be reported to the NRC staff within 90 days following startup from the respective refueling outage.

(h) At the end of the second refueling outage, following the implementation of the EPU, the licensee shall submit a long-term steam dryer inspection plan based on industry operating experience along with the baseline inspection results for NRC review and approval.

The license conditions in 2.C.(20) above shall expire (1) upon satisfaction of the requirements in paragraphs (f) and (g), provided that a visual inspection of the steam dryer does not reveal any new unacceptable flaw(s) or unacceptable flaw growth that is due to fatigue, and (2) upon satisfaction of the requirements specified in paragraph (h).

(21) Fatigue Monitoring Program

If stress based fatigue monitoring is used, it shall include all six stress terms in accordance with NB-3200. The condition for this requirement will be carried over and be applicable for operation under EPU conditions and in the plant life extension to 60 years.

D. The facility requires exemptions from certain requirements of 10 CFR Part 50 and 10 CFR Part 70.

i) An exemption from the critically alarm requirements of 10 CFR Part 70.24 was granted in the Special Nuclear Materials License No. SNM-1895 dated November 27, 1985. This exemption is described in Section 9.1 of Supplement 4 to the SER. This previously granted exemption is continued in this operating license.

ii) Exemptions to certain requirements of Appendix J to 10 CFR Part 50 are described in Supplements 3, 4, and 5 to the SER. These include (a) (this item left intentionally blank); (b) an exemption from the requirement of Option B of Appendix J, exempting main steam isolation valve measured leakage from the combined leakage rate limit of 0.6 La. (Section 6.2.6 of SSER 5)*; (c) an exemption from Option B of Appendix J, exempting the hydraulic control system for the reactor recirculation flow control valves from Type A and Type C leak testing (Section 6.2.6 of SSER 3); (d) an exemption from Option B of Appendix J, exempting Type C testing on traversing incore probe system shear valves. (Section 6.2.6 SSER 4)

iii) An exemption to Appendix A to 10 CFR Part 50 exempting the Control Rod Drive (CRD) hydraulic lines to the reactor recirculation pump seal purge equipment from General Design Criterion (GDC) 55. The CRD hydraulic lines to the reactor recirculation pump seal purge equipment use two simple check valves for the isolation outside containment (one side). (Section 6.2.4, SSER 3)

iv) A schedular exemption to GDC 2, Appendix A to 10 CFR Part 50, until the first refueling outage, to demonstrate the adequacy of the downcomer design under the plant faulted condition. This exemption permits additional analysis and/or modifications, as necessary, to be completed by the end of the first refueling outage. (Section 6.2.1.7.4, SSER 3)

v) A schedular exemption to GDC 50, Appendix A to 10 CFR Part 50 to allow the operating licensee until start-up following the "mini-outage," which is to occur within 12 months of commencing power operation (entering Operational Condition 1), to install redundant fuses in circuits that use transformers for redundant penetration protection in accordance with their letter of August 29, 1986 (NMP2L 0860). (Section 8.4.2, SSER 5)

* The parenthetical notation following the discussion of each exemption denotes the section of the Safety Evaluation Report (SER) and/or its supplements wherein the safety evaluation of the exemption is discussed.

vi) A schedular exemption to 10 CFR 50.55a(h) for the Neutron Monitoring System until completion of the first refueling outage to allow the operating licensee to provide qualified isolation devices for Class 1E/non-1E interfaces described in their letters of June 23, 1987 (NMP2L 1057) and June 25, 1987 (NMP2L 1058). (Section 7.2.2.10, SSER 6).

For the schedular exemptions in iv), v), and vi), above, the operating licensee, in accordance with its letter of October 31, 1986, shall certify that all systems, components, and modifications have been completed to meet the requirements of the regulations for which the exemptions have been granted and shall provide a summary description of actions taken to ensure that the regulations have been met. This certification and summary shall be provided 10 days prior to the expiration of each exemption period as described above.

The exemptions set forth in this Section 2.D are authorized by law, will not present an undue risk to public health and safety, and are consistent with the common defense and security. These exemptions are hereby granted. The special circumstances regarding each exemption are identified in the referenced section of the Safety Evaluation Report and the supplements thereto. The exemptions in ii) through vi) are granted pursuant to 10 CFR 50.12.

With these exemptions, the facility will operate to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

E. Nine Mile Point Nuclear Station, LLC shall fully implement and maintain in effect all provisions of the Commission-approved physical security, training and qualification, and safeguards contingency plans, including amendments made pursuant to provisions of the Miscellaneous Amendments and Search Requirements revisions to 10 CFR 73.55 (51 FR 27817 and 27822) and to the authority of 10 CFR 50.90 and 10 CFR 50.54(p). The combined set of plans, which contain Safeguards Information protected under 10 CFR 73.21 is entitled "Nine Mile Point Nuclear Station, LLC Physical Security, Safeguards Contingency, and Security Training and Qualification Plan, Revision 1," and was submitted by letter dated April 26, 2006. Changes made in accordance with 10 CFR 73.55 shall be implemented in accordance with the schedule set forth therein.

F. Nine Mile Point Nuclear Station, LLC shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility through Amendment No. 27 and as described in submittals dated March 25, May 7 and 9, June 10 and 25, July 11 and 16, August 19 and 22, September 5, 12, and 23, October 10, 21, and 22, and December 9, 1986, and April 10 and May 20, 1987, and as approved in the SER dated February 1985 (and Supplements 1 through 6) subject to the following provision:

Nine Mile Point Nuclear Station, LLC may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

G. The licensees shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

H. This license is effective as of the date of issuance and shall expire at midnight on October 31, 2046.

I. The UFSAR supplement, as revised, submitted pursuant to 10 CFR 54.21(d), shall be included in the next scheduled update to the USAR required by 10 CFR 50.71(e)(4) following the issuance of this renewed operating license. Until that update is complete, NMP LLC may make changes to the programs and activities described in the supplement without prior Commission approval, provided that NMP LLC evaluates such changes pursuant to the criteria set forth in 10 CFR 50.59 and otherwise complies with the requirements in that section.

J. The UFSAR supplement, as revised, describes certain future activities to be completed prior to the period of extended operation. NMP LLC shall complete these activities in accordance with Appendix A of NUREG-1900, "Safety Evaluation Report Related to the License Renewal of Nine Mile Point Nuclear Station, Units 1 and 2", dated September 2006, and shall notify the NRC in writing when implementation of these activities is complete and can be verified by NRC inspection.

K. For the renewed license term, all capsules in the reactor vessel that are removed and tested must meet the test procedures and reporting requirements of the most recent NRC-approved version of the Boiling Water Reactor Vessels and Internals Project (BWRVIP) Integrated Surveillance Program (ISP) appropriate for the configuration of the specimens in the capsule. All capsules placed in storage must be maintained for future insertion. Any changes to storage requirements must be approved by the NRC, as required by 10 CFR Part 50, Appendix H.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed by

J. E. Dyer, Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Appendix A – Technical Specifications (NUREG-1253)
2. Appendix B – Environmental Protection Plan

Date of Issuance: October 31, 2006

1.1 Definitions (continued)

MODE	A MODE shall correspond to any one inclusive combination of mode switch position, average reactor coolant temperature, and reactor vessel head closure bolt tensioning specified in Table 1.1-1 with fuel in the reactor vessel.
OPERABLE – OPERABILITY	A system, subsystem, division, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, division, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s).
PHYSICS TESTS	PHYSICS TESTS shall be those tests performed to measure the fundamental nuclear characteristics of the reactor core and related instrumentation. These tests are: a. Described in Chapter 14, Initial Test Program of the FSAR; b. Authorized under the provisions of 10 CFR 50.59; or c. Otherwise approved by the Nuclear Regulatory Commission.
RATED THERMAL POWER (RTP)	RTP shall be a total reactor core heat transfer rate to the reactor coolant of 3988 MWt.
REACTOR PROTECTION SYSTEM (RPS) RESPONSE TIME	The RPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its RPS trip setpoint at the channel sensor until de-energization of the scram pilot valve solenoids. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.

(continued)

2.0 SAFETY LIMITS (SLs)

2.1 SLs

2.1.1 Reactor Core SLs

2.1.1.1 With the reactor steam dome pressure < 785 psig or core flow < 10% rated core flow:

THERMAL POWER shall be ≤ 23% RTP.

2.1.1.2 With the reactor steam dome pressure ≥ 785 psig and core flow ≥ 10% rated core flow:

MCPR shall be ≥ 1.07 for two recirculation loop operation or ≥ 1.09 for single recirculation loop operation.

2.1.1.3 Reactor vessel water level shall be greater than the top of active irradiated fuel.

2.1.2 Reactor Coolant System Pressure SL

Reactor steam dome pressure shall be ≤ 1325 psig.

2.2 SL Violations

With any SL violation, the following actions shall be completed within 2 hours:

2.2.1 Restore compliance with all SLs; and

2.2.2 Insert all insertable control rods.

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.1.7.7	Verify each pump develops a flow rate ≥ 41.2 gpm at a discharge pressure ≥ 1327 psig.	In accordance with the Inservice Testing Program
SR 3.1.7.8	Verify flow through one SLC subsystem from pump into reactor pressure vessel.	24 months on a STAGGERED TEST BASIS
SR 3.1.7.9	Verify all heat traced piping between storage tank and pump suction valve is unblocked.	24 months <u>AND</u> Once within 24 hours after piping temperature is restored to ≥ 70°F
SR 3.1.7.10	Verify sodium pentaborate enrichment is ≥ 25 atom percent B-10.	Prior to addition to SLC tank

3.2 POWER DISTRIBUTION LIMITS

3.2.1 AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR)

LCO 3.2.1 All APLHGRs shall be less than or equal to the limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 23% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any APLHGR not within limits.	A.1 Restore APLHGR(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 23% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
SR 3.2.1.1 Verify all APLHGRs are less than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ 23% RTP <u>AND</u> 24 hours thereafter

3.2 POWER DISTRIBUTION LIMITS

3.2.2 MINIMUM CRITICAL POWER RATIO (MCPR)

LCO 3.2.2 All MCPRs shall be greater than or equal to the MCPR operating limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 23% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any MCPR not within limits.	A.1 Restore MCPR(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 23% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.2.1	Verify all MCPRs are greater than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ 23% RTP <u>AND</u> 24 hours thereafter

(continued)

3.2 POWER DISTRIBUTION LIMITS

3.2.3 LINEAR HEAT GENERATION RATE (LHGR)

LCO 3.2.3 All LHGRs shall be less than or equal to the limits specified in the COLR.

APPLICABILITY: THERMAL POWER ≥ 23% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Any LHGR not within limits.	A.1 Restore LHGR(s) to within limits.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 23% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.2.3.1	Verify all LHGRs are less than or equal to the limits specified in the COLR.	Once within 12 hours after ≥ 23% RTP <u>AND</u> 24 hours thereafter

ACTIONS (continued)

CONDITION	REQUIRED ACTION	COMPLETION TIME
C. One or more Functions with RPS trip capability not maintained.	C.1 Restore RPS trip capability.	1 hour
D. Required Action and associated Completion Time of Condition A, B, or C not met.	D.1 Enter the Condition referenced in Table 3.3.1.1-1 for the channel.	Immediately
E. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	E.1 Reduce THERMAL POWER to < 26% RTP.	4 hours
F. As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	F.1 Initiate alternate method to detect and suppress thermal-hydraulic instability oscillations. <u>AND</u> F.2 Restore required channel to OPERABLE status.	12 hours 120 days
G. Required Action and associated Completion Time of Condition F not met. <u>OR</u> As required by Required Action D.1 and referenced in Table 3.3.1.1-1.	G.1 Be in MODE 2.	6 hours

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.3	------------------------------NOTE------------------------------ Not required to be performed until 12 hours after THERMAL POWER ≥ 23% RTP. --- Verify the absolute difference between the average power range monitor (APRM) channels and the calculated power ≤ 2% RTP while operating at ≥ 23% RTP.	7 days
SR 3.3.1.1.4	------------------------------NOTE------------------------------ For Functions 1.a and 1.b, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. --- Perform CHANNEL FUNCTIONAL TEST.	7 days
SR 3.3.1.1.5	Verify the source range monitor (SRM) and intermediate range monitor (IRM) channels overlap.	Prior to fully withdrawing SRMs
SR 3.3.1.1.6	------------------------------NOTE------------------------------ Only required to be met during entry into MODE 2 from MODE 1. --- Verify the IRM and APRM channels overlap.	7days

(continued)

SURVEILLANCE REQUIREMENTS (continued)

SURVEILLANCE		FREQUENCY
SR 3.3.1.1.13	------NOTES------ 1. Neutron detectors are excluded. 2. For Functions 1.a and 2.a, not required to be performed when entering MODE 2 from MODE 1 until 12 hours after entering MODE 2. 3. For Function 2.e, the CHANNEL CALIBRATION only requires a verification of ORRM-Upscale setpoints in the APRM by the review of the "Show Parameters" display. ------------ Perform CHANNEL CALIBRATION.	24 months
SR 3.3.1.1.14	Perform LOGIC SYSTEM FUNCTIONAL TEST.	24 months
SR 3.3.1.1.15	Verify Turbine Stop Valve – Closure, and Turbine Control Valve Fast Closure, Trip Oil Pressure – Low Functions are not bypassed when THERMAL POWER is ≥ 26% RTP.	24 months
SR 3.3.1.1.16	Verify APRM OPRM-Upscale Function is not bypassed when THERMAL POWER is ≥ 26% RTP and recirculation drive flow is < 60% of rated recirculation drive flow.	24 months

(continued)

Table 3.3.1.1-1 (page 1 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Intermediate Range Monitors					
a. Neutron Flux — Upscale	2	3	H	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.5 SR 3.3.1.1.6 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 122/125 divisions of full scale
	5(a)	3	I	SR 3.3.1.1.1 SR 3.3.1.1.4 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 122/125 divisions of full scale
b. Inop	2	3	H	SR 3.3.1.1.4 SR 3.3.1.1.14	NA
	5(a)	3	I	SR 3.3.1.1.4 SR 3.3.1.1.14	NA
2. Average Power Range Monitors					
a. Neutron Flux – Upscale, Setdown	2	3 per logic channel	H	SR 3.3.1.1.2 SR 3.3.1.1.6 SR 3.3.1.1.7 SR 3.3.1.1.10 SR 3.3.1.1.13	≤ 20% RTP
b. Flow Biased Simulated Thermal Power – Upscale	1	3 per logic channel	G	SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.7 SR 3.3.1.1.10 SR 3.3.1.1.13(c),(d)	≤ .55W + 60.5% RTP and ≤ 115.5% RTP(b)
c. Fixed Neutron Flux – Upscale	1	3 per logic channel	G	SR 3.3.1.1.2 SR 3.3.1.1.3 SR 3.3.1.1.7 SR 3.3.1.1.10 SR 3.3.1.1.13	≤ 120% RTP
d. Inop	1,2	3 per logic channel	H	SR 3.3.1.1.7 SR 3.3.1.1.10	NA
e. OPRM-Upscale	1	3 per logic channel	F	SR 3.3.1.1.2 SR 3.3.1.1.7 SR 3.3.1.1.10 SR 3.3.1.1.13 SR 3.3.1.1.16	As specified in the COLR
f. 2-Out-Of-4 Voter	1,2	2	H	SR 3.3.1.1.2 SR 3.3.1.1.10 SR 3.3.1.1.14 SR 3.3.1.1.17	NA

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

(b) Allowable Value is .50(W - 5%) + 53.5% RTP when reset for single loop operation per LCO 3.4.1, "Recirculation Loops Operating."

(c) If the As-Found channel setpoint is outside its predefined As-Found tolerances, then the channel shall be evaluated to verify that it is functioning as required before returning the channel to service.

(d) The instrument channel setpoint shall be reset to a value within the As-Left tolerance around the nominal trip setpoint at the completion of the surveillance; otherwise, the channel shall be declared inoperable. Setpoints more conservative than the nominal trip setpoint are acceptable provided that the As-Found and As-Left tolerances apply to the actual setpoint implemented in the surveillance procedures to confirm channel performance. The nominal trip setpoint and the methodologies used to determine the As-Found and the As-Left tolerances are specified in the Bases associated with the specified function.

Table 3.3.1.1-1 (page 2 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. Reactor Vessel Steam Dome Pressure – High	1,2	2	H	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.17	≤ 1072 psig
4. Reactor Vessel Water Level – Low, Level 3	1,2	2	H	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.17	≥ 157.8 inches
5. Main Steam Isolation Valve – Closure	1	8	G	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.17	≤ 12% closed
6. Drywell Pressure – High	1,2	2	H	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 1.88 psig
7. Scram Discharge Volume Water Level – High					
a. Transmitter/Trip Unit	1,2	2	H	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.14	≤ 49.5 inches
	5(a)	2	I	SR 3.3.1.1.1 SR 3.3.1.1.8 SR 3.3.1.1.9 SR 3.3.1.1.11 SR 3.3.1.1.14	≤ 49.5 inches
b. Float Switch	1,2	2	H	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 49.5 inches
	5(a)	2	I	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14	≤ 49.5 inches
8. Turbine Stop Valve – Closure	≥ 26% RTP	4	E	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.15 SR 3.3.1.1.17	≤ 7% closed

(continued)

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

Table 3.3.1.1-1 (page 3 of 3)
Reactor Protection System Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION D.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
9. Turbine Control Valve Fast Closure, Trip Oil Pressure – Low	≥ 26% RTP	2	E	SR 3.3.1.1.8 SR 3.3.1.1.13 SR 3.3.1.1.14 SR 3.3.1.1.15 SR 3.3.1.1.17	≥ 465 psig
10. Reactor Mode Switch – Shutdown Position	1,2	2	H	SR 3.3.1.1.12 SR 3.3.1.1.14	NA
	5(a)	2	I	SR 3.3.1.1.12 SR 3.3.1.1.14	NA
11. Manual Scram	1,2	4	H	SR 3.3.1.1.4 SR 3.3.1.1.14	NA
	5(a)	4	I	SR 3.3.1.1.4 SR 3.3.1.1.14	NA

(a) With any control rod withdrawn from a core cell containing one or more fuel assemblies.

3.3 INSTRUMENTATION

3.3.2.2 Feedwater System and Main Turbine High Water Level Trip Instrumentation

LCO 3.3.2.2 Three channels of feedwater system and main turbine high water level trip instrumentation shall be OPERABLE.

APPLICABILITY: THERMAL POWER ≥ 23% RTP.

ACTIONS

--- NOTE ---
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One feedwater system and main turbine high water level trip channel inoperable.	A.1 Place channel in trip.	7 days
B. Two or more feedwater system and main turbine high water level trip channels inoperable.	B.1 Restore feedwater system and main turbine high water level trip capability.	2 hours

(continued)

ACTIONS (continued)

CONDITION	REQUIRED ACTION		COMPLETION TIME
C. Required Action and associated Completion Time not met.	C.1	--------------NOTE-------------- Only applicable if inoperable channel is the result of an inoperable feedwater pump breaker. -----------------------------------	
		Remove affected feedwater pump(s) from service.	4 hours
	OR		
	C.2	Reduce THERMAL POWER to < 23% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

--- NOTE ---

When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided feedwater system and main turbine high water level trip capability is maintained.

SURVEILLANCE		FREQUENCY
SR 3.3.2.2.1	Perform CHANNEL CHECK.	24 hours
SR 3.3.2.2.2	Perform CHANNEL FUNCTIONAL TEST.	92 days

(continued)

3.3 INSTRUMENTATION

3.3.4.1 End of Cycle Recirculation Pump Trip (EOC-RPT) Instrumentation

LCO 3.3.4.1

a. Two channels per trip system for each EOC-RPT instrumentation Function listed below shall be OPERABLE:

1. Turbine Stop Valve (TSV) – Closure; and
2. Turbine Control Valve (TCV) Fast Closure, Trip Oil Pressure – Low.

<u>OR</u>

b. LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for inoperable EOC-RPT as specified in the COLR are made applicable.

APPLICABILITY: THERMAL POWER $\geq$ 26% RTP with any recirculation pump in fast speed.

ACTIONS

--- NOTE ---
Separate Condition entry is allowed for each channel.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more required channels inoperable.	A.1 Restore channel to OPERABLE status. <u>OR</u>	72 hours (continued)

ACTIONS

CONDITION	REQUIRED ACTION		COMPLETION TIME
A. (continued)	A.2	------------NOTE------------ Not applicable if inoperable channel is the result of an inoperable breaker. -------------------------------- Place channel in trip.	 72 hours
B. One or more Functions with EOC-RPT trip capability not maintained. <u>AND</u> MCPR limit for inoperable EOC-RPT not made applicable.	B.1	Restore EOC-RPT trip capability.	2 hours
	<u>OR</u>		
	B.2	Apply the MCPR limit for inoperable EOC-RPT as specified in the COLR.	2 hours
C. Required Action and associated Completion Time not met.	C.1	Remove the associated recirculation pump fast speed breaker from service.	4 hours
	<u>OR</u>		
	C.2	Reduce THERMAL POWER to < 26% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

--- NOTE ---
When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours provided the associated Function maintains EOC-RPT trip capability.

SURVEILLANCE		FREQUENCY
SR 3.3.4.1.1	Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.4.1.2	Perform CHANNEL CALIBRATION. The Allowable Values shall be: a. TSV – Closure: ≤ 7% closed; and b. TCV Fast Closure, Trip Oil Pressure – Low: ≥ 465 psig.	24 months
SR 3.3.4.1.3	Perform LOGIC SYSTEM FUNCTIONAL TEST, including breaker actuation.	24 months
SR 3.3.4.1.4	Verify TSV – Closure and TCV Fast Closure, Trip Oil Pressure – Low Functions are not bypassed when THERMAL POWER is ≥ 26% RTP.	24 months
SR 3.3.4.1.5	------------------------------NOTE------------------------------ Breaker arc suppression time may be assumed from the most recent performance of SR 3.3.4.1.6. --- Verify the EOC-RPT SYSTEM RESPONSE TIME is within limits.	24 months on a STAGGERED TEST BASIS

(continued)

Table 3.3.6.1-1 (page 1 of 5)
Primary Containment Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level – Low Low Low, Level 1	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ 10.8 inches
b. Main Steam Line Pressure – Low	1	2	E	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	≥ 746 psig
c. Main Steam Line Flow – High	1,2,3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6 SR 3.3.6.1.7	≤ 184.4 psid
d. Condenser Vacuum – Low	1,2(a), 3(a)	2	D	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 7.6 inches Hg vacuum
e. Main Steam Line Tunnel Temperature – High	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 170.6°F
f. Main Steam Line Tunnel Differential Temperature – High	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 71.7°F
g. Main Steam Line Tunnel Lead Enclosure Temperature – High	1,2,3	2 per area	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 151.6°F(b)
h. Manual Initiation	1,2,3	4	G	SR 3.3.6.1.6	NA

(continued)

(a) With any turbine stop valve not closed.

(b) 151.6°F + (0.6)(T_{amb} - 90°F) and ≤ 175.6°F provided the absence of steam leaks in the main steam line tunnel lead enclosure area is verified by visual inspection prior to establishing the Allowable Value.

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.4.3.1	------------------------------NOTES--------------------------- 1. Not required to be performed until 4 hours after associated recirculation loop is in operation. 2. Not required to be performed until 24 hours after > 23% RTP. --	
	Verify at least two of the following criteria (a, b, and c) are satisfied for each operating recirculation loop: a. Jet pump loop flow versus flow control valve position differs by ≤ 10% from established patterns. b. Jet pump loop flow versus recirculation loop drive flow differs by ≤ 10% from established patterns. c. Each jet pump diffuser to lower plenum differential pressure differs by ≤ 20% from established patterns.	24 hours

3.7 PLANT SYSTEMS

3.7.5 Main Turbine Bypass System

LCO 3.7.5 The Main Turbine Bypass System shall be OPERABLE.

<u>OR</u>

LCO 3.2.2, "MINIMUM CRITICAL POWER RATIO (MCPR)," limits for an inoperable Main Turbine Bypass System, as specified in the COLR, are made applicable.

APPLICABILITY: THERMAL POWER ≥ 23% RTP.

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. Requirements of the LCO not met.	A.1 Satisfy the requirements of the LCO.	2 hours
B. Required Action and associated Completion Time not met.	B.1 Reduce THERMAL POWER to < 23% RTP.	4 hours

SURVEILLANCE REQUIREMENTS

SURVEILLANCE		FREQUENCY
SR 3.7.5.1	Perform a system functional test.	24 months
SR 3.7.5.2	Verify the TURBINE BYPASS SYSTEM RESPONSE TIME is within limits.	24 months

4.0 DESIGN FEATURES (continued)

4.3 Fuel Storage

4.3.1 Criticality

4.3.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1.2 of the USAR;

b. A nominal 6.18 inch center to center distance between fuel assemblies placed in the storage racks;

c. Fuel assemblies having a maximum k-infinity of 1.32 in the normal reactor core configuration at cold conditions; and

d. Fuel assemblies having a maximum U-235 enrichment of 4.9 weight percent.

4.3.1.2 The new fuel storage racks are designed and shall be maintained with:

a. $k_{eff} \leq 0.95$ if fully flooded with unborated water, which includes an allowance for uncertainties as described in Section 9.1.1 of the USAR;

b. $k_{eff} \leq 0.98$ with all but one of the non-combustible storage vaults covers in place when optimum moderation (foam, spray, fogging, or small droplets) is assumed;

c. A nominal 7.00 inch center to center distance between fuel assemblies placed within a storage rack and a nominal 12.25 inch center to center distance between fuel assemblies in adjacent racks;

d. Fuel assemblies having a maximum k-infinity of 1.34 in the normal reactor core configuration at cold conditions; and

e. Fuel assemblies having a maximum U-235 enrichment of 4.9 weight percent.

(continued)

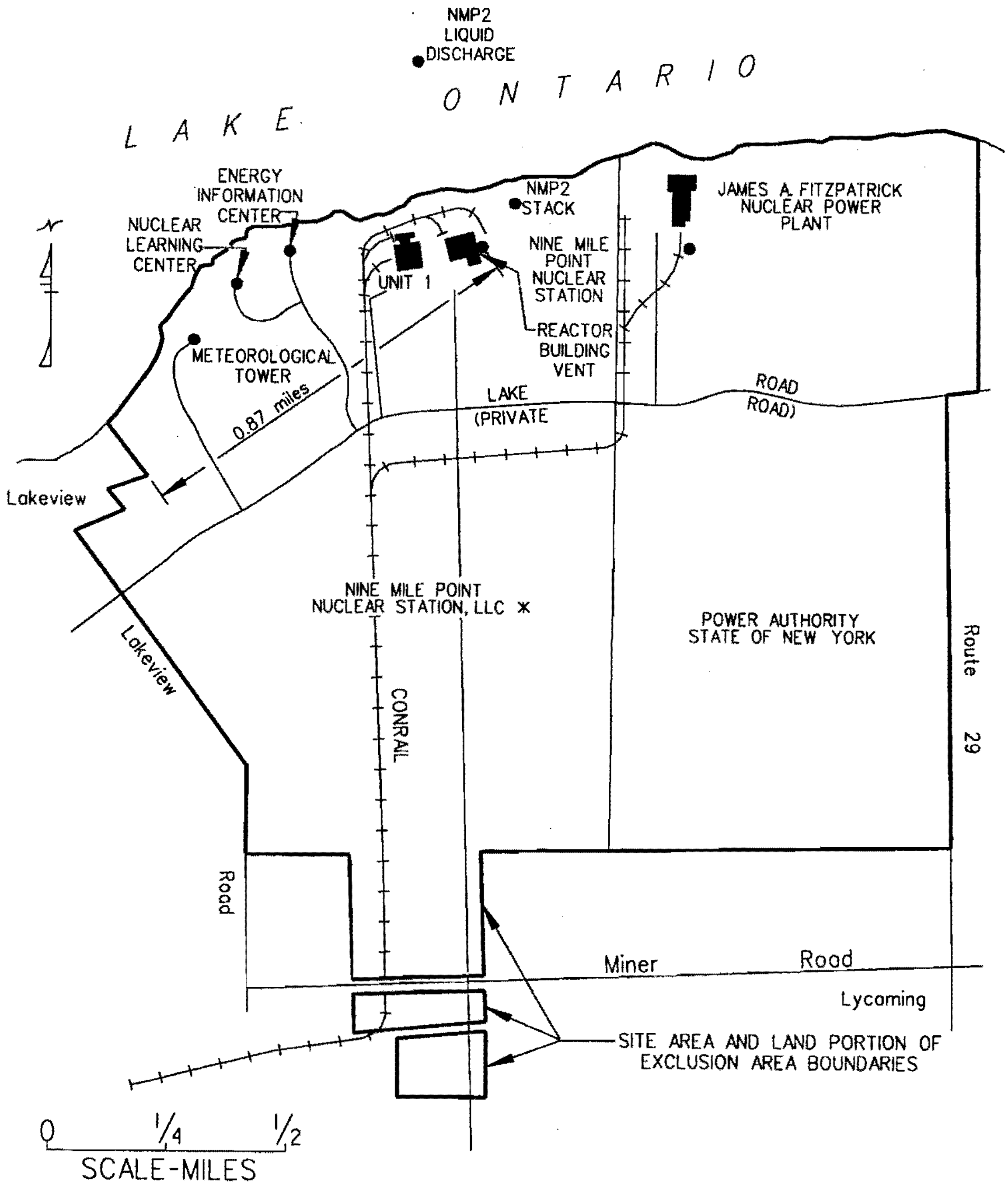
4.0 DESIGN FEATURES (continued)

4.3.2 Drainage

The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 329 ft 7 inches.

4.3.3 Capacity

The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 4049 fuel assemblies.



* Niagara Mohawk Power Corporation retains ownership in certain transmission line and switchyard facilities within the exclusion area boundary. Access and usage are controlled by Nine Mile Point Nuclear Station, LLC by Agreement.

Figure 4.1-1 (Page 1 of 1)
Site Area and Land Portion of Exclusion Area Boundaries