UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

NIAGARA MOHAWK POWER CORPORATION

DOCKET NO. 50-410

NINE MILE POINT NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 44
License No. NPF-69

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

   A. The application for amendment by Niagara Mohawk Power Corporation (the licensee) dated December 30, 1992, as supplemented May 19, 1993, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter 1;

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

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

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

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. NPF-69 is hereby amended to read as follows:

(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. 44 are hereby incorporated into this license. Niagara Mohawk Power Corporation shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of the date of its issuance to be implemented within 30 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Capra

Robert A. Capra, Director
Project Directorate I-1
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Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 21, 1993

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 44 TO FACILITY OPERATING LICENSE NO. NPF-69

DOCKET NO. 50-410

Revise Appendix A as follows:

Remove Pages	Insert Pages
3/4 4-12	3/4 4-12
-	3/4 4-12a (added page)
3/4 4-13	3/4 4-13
3/4 4-14	3/4 4-14
-	3/4 4-14a (added page)
B3/4 4-3	B3/4 4-3
B3/4 4-3a	B3/4 4-3a

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

LIMITING CONDITIONS FOR OPERATION

3.4.3.1 The following reactor coolant system leakage detection systems shall be OPERABLE:

a. The primary containment airborne particulate radioactivity monitoring system,

b. The primary containment airborne gaseous radioactivity monitoring system,

c. The drywell floor drain tank fill rate monitoring system, and

d. Drywell equipment drain tank fill rate monitoring system.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With the primary containment airborne particulate radioactivity monitoring system or the primary containment airborne gaseous radioactivity monitoring system inoperable, operation may continue for up to 30 days provided grab samples of the containment atmosphere are obtained and analyzed at least once per 12 hours; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the drywell equipment drain tank fill rate monitoring system inoperable, operation may continue for up to 30 days provided that the drywell equipment drain tank fill rate is determined via alternate methods; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With the drywell floor drain tank fill rate monitoring system inoperable, operation may continue for up to 30 days provided that the drywell floor drain tank fill rate is determined via alternate methods; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With both drywell floor drain and the drywell equipment drain tank fill rate monitoring systems inoperable, restore either system to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

REACTOR COOLANT SYSTEM

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

LEAKAGE DETECTION SYSTEMS

SURVEILLANCE REQUIREMENTS

4.4.3.1 The reactor coolant system leakage detection systems shall be demonstrated OPERABLE by:

a. Primary containment atmosphere particulate and gaseous monitoring systems-performance of a CHANNEL CHECK at least once per 12 hours, a SOURCE CHECK at least once per 31 days, a CHANNEL FUNCTIONAL TEST at least once per 184 days and a CHANNEL CALIBRATION at least once per 18 months.

b. Primary containment sump flow monitoring system-performance of a CHANNEL FUNCTIONAL TEST at least once per 31 days and a CHANNEL CALIBRATION TEST at least once per 18 months.

REACTOR COOLANT SYSTEM LEAKAGE

OPERATIONAL LEAKAGE

LIMITING CONDITIONS FOR OPERATION

3.4.3.2 Reactor coolant system (RCS) leakage shall be limited to:

a. No PRESSURE BOUNDARY LEAKAGE.
b. 5 gpm UNIDENTIFIED LEAKAGE.
c. 25 gpm IDENTIFIED LEAKAGE averaged over any 24-hour period.
d. 0.5 gpm leakage per nominal inch of valve size up to a maximum 5 gpm at an RCS pressure of 1020 $\pm$ 20 psig from any reactor coolant system pressure isolation valve specified in Table 3.4.3.2-1.
e. 2 gpm increase in UNIDENTIFIED LEAKAGE within any 24-hour period in Mode 1.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

a. With any PRESSURE BOUNDARY LEAKAGE, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. With any RCS leakage greater than the limits in Specification 3.4.3.2.b and/or c (above), reduce the leakage rate to within the limits within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With any RCS pressure isolation valve leakage greater than the above limit, isolate the high-pressure portion of the affected system from the low-pressure portion within 4 hours by use of at least two other closed (manual or deactivated automatic or check*) valves, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With one or more of the high/low-pressure interface valve leakage pressure monitors shown in Table 3.4.3.2-2 inoperable, restore the inoperable monitor(s) to OPERABLE status within 7 days or verify the pressure to be less than the alarm setpoint at least once per 12 hours; restore the inoperable monitor(s) to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

*Which have been verified not to exceed the allowable leakage limit at the last refueling outage.

LIMITING CONDITIONS FOR OPERATION

e. With one or more of the required interlocks shown in Table 3.4.3.2-3 inoperable, restore the inoperable interlock to OPERABLE status within 7 days or isolate the affected heat exchanger(s) from the RCIC steam supply by closing and deenergizing heat exchanger valves 2 RHS*MOV22A and 2RHS*MOV80A or 2RHS*MOV22B and 2RHS*MOV80B, as appropriate.

f. With any reactor coolant system leakage greater than the limit in 3.4.3.2.e above, identify the source of leakage within 4 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.4.3.2.1 The RCS leakage shall be demonstrated to be within each of the above limits by:

a. Monitoring the primary containment airborne particulate radioactivity at least once per 12 hours,

b. Monitoring the drywell floor drain tank and equipment drain tank fill rate at least once per 8 hours,

c. Monitoring the primary containment airborne gaseous radioactivity at least once per 12 hours, and

d. Monitoring the reactor vessel head flange leak detection system at least once per 24 hours.

4.4.3.2.2 Each RCS pressure isolation valve specified in Table 3.4.3.2-1 shall be demonstrated OPERABLE by leak testing pursuant to Specification 4.0.5 as outlined in the ASME Code Section XI, paragraph IWV-3427(b) and verifying the leakage of each valve to be within the specified limit:

a. At least once per 18 months, and
b. Before returning the valve to service following maintenance, repair, or replacement work on the valve.

The provisions of Specification 4.0.4 are not applicable for entry into OPERATIONAL CONDITION 3.

REACTOR COOLANT SYSTEM

REACTOR COOLANT SYSTEM LEAKAGE

OPERATIONAL LEAKAGE

SURVEILLANCE REQUIREMENTS

4.4.3.2.3 The high/low-pressure interface valve leakage pressure monitors shall be demonstrated OPERABLE with setpoints per Table 3.4.3.2-2 by performance of a:

a. CHANNEL FUNCTIONAL TEST at least once per 31 days, and
b. CHANNEL CALIBRATION at least once per 18 months.

4.4.3.2.4 The high/low-pressure interface interlock for the steam condensing mode bypass valve shall be demonstrated OPERABLE with trips setpoints per Table 3.4.3.2-3 by performance of:

a. CHANNEL FUNCTIONAL TEST at least once per 92 days, and
b. CHANNEL CALIBRATION at least once per 18 months.

RECIRCULATION SYSTEM

3/4.4.1 (Continued)

recirculation pump and recirculation nozzles. Sudden equalization of a temperature difference $\geq$ 145°F between the reactor vessel bottom head coolant and the coolant in the upper region of the reactor vessel by increasing core flow rate would cause undue stress in the reactor vessel bottom head.

3/4.4.2 SAFETY/RELIEF VALVES

The safety/relief valves operate during a postulated ATWS event to prevent the reactor coolant system from being pressurized above a design allowable value of 1375 psig in accordance with the ASME Code. A total of 16 OPERABLE safety/relief valves is required to limit local pressure at active components to within ASME III allowable design values (Service Level A). All other appropriate ASME III limits are also bounded by this requirement. Specified surveillance intervals have been determined in accordance with GENE-770-06-1, "Bases for Changes to Surveillance Test Intervals and Allowed Out-Of-Service Times for Selected Instrumentation Technical Specification," as approved by the NRC and documented in the SER (letter to R. D. Binz IV from C. E. Rossi dated July 21, 1992).

The safety/relief valve lift settings will be demonstrated only during shutdown in accordance with the provisions of Specification 4.0.5.

3/4.4.3 REACTOR COOLANT SYSTEM LEAKAGE

3/4.4.3.1 LEAKAGE DETECTION SYSTEMS

The RCS leakage detection systems required by this specification are provided to monitor and detect leakage from the reactor coolant pressure boundary. These detection systems are consistent with the recommendations of RG 1.45, "Reactor Coolant Pressure Boundary Leakage Detection Systems," May 1973.

Generic Letter 88-01, Supplement 1, allows for alternate methods to determine drywell leakage when the normal leakage monitoring systems are inoperable. These alternate methods may be used for up to 30 days, provided their suitability with regard to accuracy and inspectability is demonstrated. The alternate methods given in the Generic Letter Supplement are to manually pump down the drain tank, or to measure the differences in tank level.

3/4.4.3.2. OPERATIONAL LEAKAGE

The allowable leakage rates from the reactor coolant system have been based on the predicted and experimentally observed behavior of cracks in pipes. The background leakage normally expected to result from equipment design and the detection capability of the instrumentation for determining system leakage were also considered. The evidence obtained from experiments suggests that for leakage somewhat greater than that specified for UNIDENTIFIED LEAKAGE, the probability is small

REACTOR COOLANT SYSTEM

BASES

3/4.4.3.2 OPERATIONAL LEAKAGE (Continued)

that the imperfection or crack associated with such leakage would grow rapidly. An UNIDENTIFIED LEAKAGE increase of > 2 gpm within the previous 24 hour period indicates a potential flaw in the Reactor Coolant Pressure Boundary and must be quickly evaluated to determine the source and extent of the leakage. The increase is measured relative to the steady state value; temporary changes in leakage rate as a result of transient conditions (e.g., startup) are not considered. As such, the 2 gpm increase limit is only applicable in MODE 1 when operating pressures and temperatures are established. However, in all cases, if the leakage rates exceed the values specified or the leakage is located and known to be PRESSURE BOUNDARY LEAKAGE, the reactor will be shut down to allow further investigation and corrective action.

The Surveillance Requirements for RCS pressure isolation valves provide added assurance of valve integrity, thereby reducing the probability of gross valve failure and consequent intersystem LOCA.