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

April 12, 1993

Docket No. 50-220

Mr. B. Ralph Sylvia
Executive Vice President, Nuclear
Niagara Mohawk Power Corporation
301 Plainfield Road
Syracuse, New York 13212

Dear Mr. Sylvia:

SUBJECT: ISSUANCE OF AMENDMENT FOR NINE MILE POINT NUCLEAR STATION UNIT NO. 1 (TAC NO. M79135)

The Commission has issued the enclosed Amendment No. 140 to Facility Operating License No. DPR-63 for the Nine Mile Point Nuclear Station Unit No. 1 (NMP-1). The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated November 20, 1990, which was superseded by an application dated February 7, 1992. The February 7, 1992, application was supplemented by letters dated June 22, 1992, and January 29, 1993. The TSs and supporting information in Attachment 2 of the February 7, 1992, application and supplemental letters dated June 22, 1992, and January 29, 1993, were replaced by information provided in a letter dated February 18, 1993, and supplemented by a letter dated March 29, 1993.

The amendment revises TSs 3.2.7.1, 3.3.3, 4.3.3, and 3.3.4 and associated Bases to update these TSs to conform to the requirements of 10 CFR Part 50, Appendix J, and NRC Safety Evaluations (SEs) dated May 6, 1988, and November 9, 1988.

A copy of the related SE is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

Donald S. Brinkman

Donald S. Brinkman, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 140 to DPR-63
2. Safety Evaluation

cc w/enclosures:
See next page

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CP-1

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Niagara Mohawk Power Corporation

Nine Mile Point Nuclear Station
Unit No. 1

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DATED: April 12, 1993

AMENDMENT NO. 140 TO FACILITY OPERATING LICENSE NO. DPR-63-NINE MILE POINT UNIT 1

Docket File
NRC & Local PDRs
PDI-1 Reading
S. Varga, 14/E/4
J. Calvo, 14/A/4
R. Capra
C. Vogan
D. Brinkman
R. Barrett, 8/H/7
C. McCracken, 8/D/1
OGC-WF
D. Hagan, 3302 MNBB
G. Hill (2), P1-22
Wanda Jones, P-370
C. Grimes, 11/F/23
D. Shum, 8/D/1
J. Pulsipher, 8/D/1
ACRS (10)
OPA
OC/LFMB
PD plant-specific file
C. Cowgill, Region I

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

NIAGARA MOHAWK POWER CORPORATION

DOCKET NO. 50-220

NINE MILE POINT NUCLEAR STATION UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 140
License No. DPR-63

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

   A. The application for amendment by Niagara Mohawk Power Corporation (the licensee) dated November 20, 1990, as superseded February 7, 1992, as supplemented June 22, 1992, January 29, 1993, February 18, 1993, and March 29, 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 I;

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

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

   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. DPR-63 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 140, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance to be implemented within 90 days.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Capra

Robert A. Capra, Director
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical Specifications

Date of Issuance: April 12, 1993

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 140 TO FACILITY OPERATING LICENSE NO. DPR-63

DOCKET NO. 50-220

Revise Appendix A as follows:

Remove Pages	Insert Pages
118	118
118a	118a
119	119
-	119a (added page)
-	119b (added page)
120	120
120b	120b
135	135
136	136
137	137
138	138
139	139
140	140
-	140a (added page)
-	140b (added page)
-	140c (added page)
141	141
142	142
143	143
-	143a (added page)
144	144
146	146
147	147
148	148
-	148a (added page)
-	148b (added page)
149	149

LIMITING CONDITIONS FOR OPERATION
Table 3.2.7

REACTOR COOLANT SYSTEM ISOLATION VALVES

Line or System	No. of Valves (Each Line)	Location Relative to Primary Containment	Normal Position	Motive Power*	Maximum Oper. Time (Sec)	Action on Initiating Signal	Initiating Signal (All Valves have Remote Manual Backup)
Main Steam[1]	1	Inside	Open	AC Motor	10	Close	Reactor water level low-low or low reactor pressure, (with mode switch in run) or main steam line high radiation, or main steam line high flow, or low-low-low condenser vacuum, or high temperature in the steam tunnel
(Two Lines)	1	Outside	Open	Pn/DC Solenoid	10	Close	
Feedwater[1]	1	Outside	Open	AC Motor	60	---	Remote Manual
(Two Lines)	1	Outside	Open	Self Act. Ck.	---	---	---
Emergency Cooling							
Steam Leaving Reactor[1]	1	Outside	Open	AC Motor	38	Close	High emergency cooling system flow
(Two Lines)	1	Outside	Open	DC Motor	38	Close	
Condensate Return to Reactor[1]	1	Inside	Closed	Self Act. Ck.	---	---	---
(Two Lines)	1	Outside	Closed	Pn/DC Solenoid	60	Close	High emergency cooling system flow
					60	Open	Reactor water level low-low or high reactor pressure

LIMITING CONDITIONS FOR OPERATION
Table 3.2.7 (Continued)

REACTOR COOLANT SYSTEM ISOLATION VALVES

Line or System	No. of Valves (Each Line)	Location Relative to Primary Containment	Normal Position	Motive Power*	Maximum Oper. Time (Sec)	Action on Initiating Signal	Initiating Signal (All Valves Have Remote Manual Backup)
Reactor Cleanup							
Water Leaving Reactor(1)	1	Inside	Open	AC Motor	18	Close	Reactor water level low-low or high area temperature or liquid poison initiation
(One Line)	1	Outside	Open	DC Motor	18	Close	
Water Return to Reactor(1)	1	Inside	Open	AC Motor	18	Close	
(One Line)	1	Outside	Open	Self Act. Ck.	---	---	---
Shutdown Cooling							
Water Leaving Reactor(1)	1	Inside	Closed	AC Motor	40	Close	Reactor water level low-low, or high area temperature
(One Line)	1	Outside	Closed	DC Motor	40	Close	
Water Return to Reactor(1)	1	Inside	Closed	AC Motor	40	Close	
(One Line)	1	Outside	Closed	Self Act. Ck.	---	---	---

LIMITING CONDITIONS FOR OPERATION
Table 3.2.7 (Continued)

REACTOR COOLANT SYSTEM ISOLATION VALVES

Line or System	No. of Valves (Each Line)	Location Relative To Primary Containment	Normal Position	Motive Power*	Maximum Oper. Time (Sec)	Action on Initiating Signal	Initiating Signal (All Valves Have Remote Manual Backup)
Liquid Poison(1) (One Line)	1 1	Inside Outside	Closed Closed	Self Act. Ck. Self Act. Ck.	--- ---	--- ---	--- ---
Control Rod Drive Hydraulic(2) (One Line)	1 1	Inside Outside	Open Open	Self Act. Ck. Self Act. Ck.	--- ---	--- ---	--- ---
Scram Discharge Volume(1) System Vent** (One Line)	2	Outside	Open	Pn/AC Solenoid	10	Close	Automatic or manual reactor scram
Scram Discharge Volume(1) System Drain** (One Line)	2	Outside	Open	Pn/AC Solenoid	10	Close	
Core Spray							
Core Spray Injection(3) (Two Lines)	2 1	Inside Outside	Closed Open	AC Motor AC Motor	22.5 22.5	Open Open	Reactor water level low-low or high drywell pressure coincident with reactor vessel pressure less than 365 psig
Core Spray High Point Vent(4) (Two Lines)	1 1	Inside Outside	Closed Closed	AC Motor Pn/DC Solenoid	27 27	Close Close	Reactor water level low-low or high drywell pressure
Core Spray Condensate Supply(5) (Keep Fill) (Two Lines)	2	Outside	Open	Self Act. Ck.	---	---	---
Core Spray System Valves(5) (Two Lines)	1	Outside	Closed	Self Act. Ck.	---	---	---
Core Spray Pump Discharge(4) (Two Test Lines to Suppression Chamber)	1	Outside	Closed	AC Motor	27	Close	Reactor water level low-low or high drywell pressure

LIMITING CONDITIONS FOR OPERATION
Table 3.2.7 (Continued)

REACTOR COOLANT SYSTEM ISOLATION VALVES

Line or System	No. of Valves (Each Line)	Location Relative To Primary Containment	Normal Position	Motive Power*	Maximum Oper. Time (Sec)	Action on Initiating Signal	Initiating Signal (All Valves Have Remote Manual Backup)
Post Accident Reactor Sampling[1][6] (One Line)	1	Outside	Open	Self Act. Flow Fuse	---	---	---
	1	Outside	Closed	Pn/DC Solenoid	30	Close	Reactor water level low-low or main steam line high radiation or low-low-low condenser vacuum or reactor low pressure, (with mode switch in run) or high temperature in the steam tunnel or main steam line high flow
Reactor Recirculation System Sampling[1] (One Line)	1	Inside	Closed	AC Motor	20	Close	
	1	Outside	Closed	DC Motor	20	Close	

Notes:

* Pn - Pneumatically Operated
** Section 3.1.1e for LCO Requirements

(1) These valves do not have to be vented during the Type A test. However, Type C leakage from these valves is added to the Type A test results, if not vented.

(2) These valves have flow through them during and following an accident (a water seal) and receive a water leak rate test in accordance with the IST Program.

(3) The inside core spray injection isolation valves are water sealed during and after an accident. These valves are leak rate tested with water in accordance with the IST Program. The outside core spray injection isolation valves are open with their breakers locked in the off position. Therefore, the outside core spray injection isolation valves do not have to be tested under the IST or Appendix J Leakage Program.

(4) These valves are provided with a water seal. Valves shall be tested during each refuel outage not to exceed two years consistent with Appendix J water seal testing requirements. Leakage rates shall be limited to 0.5 gpm per nominal inch of valve diameter up to a maximum of 5 gpm.

(5) These valves are tested in accordance with Section 4.2.7.1a.

(6) The self actuating flow fuse is tested in accordance with Section 4.3.4c.

BASES FOR 3.2.7 AND 4.2.7 REACTOR COOLANT SYSTEM ISOLATION VALVES

Double isolation valves are provided in lines which connect to the reactor coolant system to assure isolation and minimize reactor coolant loss in the event of a line rupture. The specified valve requirements assure that isolation is already accomplished with one valve shut or provide redundancy in an open line with two operative valves. Except where check valves are used as one or both of a set of double isolation valves, the isolation valves shall be capable of automatic initiation and the closure times presented in Table 3.2.7. These closure times were selected to minimize coolant losses in the event of the specific line rupturing. Using the longest closure time on the main-steam-line valves following a main-steam-line break (Section XV C.1.0)[1], the core is still covered by the time the valves close. Following a specific system line break, the cleanup and shutdown cooling closing times will upon initiation from a low-low level signal limit coolant loss such that the core is not uncovered. Feedwater flow would quickly restore coolant levels to prevent clad damage. Closure times are discussed in Section VI-D.1.0[1].

The valve operability test intervals are based on periods not likely to significantly affect operations, and are consistent with testing of other systems. Results obtained during closure testing are not expected to differ appreciably from closure times under accident conditions as in most cases, flow helps to seal the valve.

The test interval of once per operating cycle for automatic initiation results in a failure probability of 1.1×10^{-7} (Fifth Supplement, p. 115)[2] that a line will not isolate. More frequent testing for valve operability results in a more reliable system.

(1) UFSAR
(2) FSAR

TABLE 3.2.7.1

PRIMARY COOLANT SYSTEM PRESSURE ISOLATION VALVES

	System	Valve No.	Maximum[a] Allowable Leakage
1.	Core Spray System	40-03	≤5.0 gpm
		40-13	≤5.0 gpm
2.	Condensate Supply to Core Spray (Keep Fill System)	40-20	≤5.0 gpm
		40-21	≤5.0 gpm
		40-22	≤5.0 gpm
		40-23	≤5.0 gpm

Footnote:

(a) 1. Leakage rates shall be limited to 0.5 gpm per nominal inch of valve diameter up to a maximum of 5 gpm.

2. Test differential pressure shall not be less than 150 psid.

3. The observed leakage at test differential pressure shall be adjusted to the functional maximum pressure differential.

LIMITING CONDITION FOR OPERATION

3.3.3 LEAKAGE RATE

Applicability:

Applies to the allowable leakage rate of the primary containment system.

Objective:

To assure the capability of the containment in limiting radiation exposure to the public from exceeding values specified in 10CFR100 in the event of a loss-of-coolant accident accompanied by significant fuel cladding failure and hydrogen generation from a metal-water reaction.

To assure that periodic surveillances of reactor containment penetrations and isolation valves are performed so that proper maintenance and repairs are made during the service life of the containment, and systems and components penetrating primary containment.

Specification:

Whenever the reactor coolant system temperature is above 215F the primary containment leakage rate shall be within the limits of 4.3.3.b.

SURVEILLANCE REQUIREMENT

4.3.3 LEAKAGE RATE

Applicability:

Applies to the primary containment system leakage rate.

Objective:

To verify that the leakage from the primary containment system is maintained within specified values.

Specification:

a. Integrated Primary Containment Leakage Rate - Type A Test

   (1) Integrated leak rate tests shall be performed at the test pressure (P_t) of 22 psig.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	Containment pressure shall not be permitted to decrease more than one (1) psi below P_t.
	(2) Type B and C tests should be completed prior to each Type A test. Type B and C leakages (penalties) not accounted for in the Type A test shall be incorporated as minimum pathway additions to the Upper Confidence Limit (UCL) to determine the overall as left integrated leakage rate.
	(3) If the leakage rate exceeds the acceptance criterion, corrective action shall be required. If, during the performance of a Type A test, excessive leakage occurs through locally testable penetrations or isolation valves to the extent that it would interfere with the satisfactory completion of the test, these leakage paths may be isolated and the Type A re-test continued until completion. The Type A test shall be considered a failed test. A local leakage test shall be performed at P_t before and after the repair of each isolated leakage path. The sum of the post repaired local leakage rates and the UCL shall be less than 75 percent of the maximum allowable leakage rate, L_t (22). Local leakage rates shall not be subtracted from the Type A test results to determine the acceptability of a test. The as found and as left leakage data values of excessive leakage areas beyond acceptance criteria shall be provided to the NRC.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

(4) Closure of the containment isolation valves for the purpose of the test shall be accomplished by the means provided for normal operation of the valves.

(5) A Type A test shall last a minimum of eight (8) hours with leakage rates calculated based on "Total Time" method. If a twenty-four (24) hour test is performed the "Mass Point" method will be used to calculate leakage rates. A verification test shall be performed following each Type A test. The verification test provides a method for assuring that systematic error or bias is given adequate consideration. During the verification test, containment pressure may not decrease more than one (1) psi below P_t.

b. <u>Acceptance Criteria - Type A Test</u>

(1) The maximum allowable leakage rate L_t (22) shall not exceed 1.19 weight percent of the contained air per 24 hours at the test pressure of 22 psig (P_t).

(2) The maximum allowable operational leakage, rate L_{to} (22) which shall be met prior to power operation following a Type A test

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

(either as measured or following repairs and retest) shall not exceed 0.75 L_t (22) (0.892 weight percent per day).

(3) When adding the leakage rate measured during a Type C test to the results of a Type A test, the leakage rate shall be determined using minimum pathway analysis.

c. Frequency

(1) Three Type A tests shall be conducted during each ten year service interval at approximately equal intervals. The third test will be conducted when the plant is shutdown for the 10 year inservice inspections.

(2) Retesting

(a) If a Type A test fails to meet the acceptance criteria of 4.3.3.b.(1), a Corrective Action Plan that focuses attention on the cause of the problem shall be developed and implemented. A Type A test that meets the requirements of

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	4.3.3.a.(3) and 4.3.3.b.(2) is required prior to plant start-up. A report of the Corrective Action following the failed Type A shall be submitted to the NRC for review and approval with the Containment Leak Test Report. (b) If any periodic Type A test fails to meet the acceptance criteria of 4.3.3.b.(1), the test schedule for subsequent Type A tests will be reviewed and approved by the NRC. (c) If two consecutive periodic Type A tests (not including an immediate retest under (a)) fail to meet the acceptance criteria of 4.3.3.a.(3), 4.3.3.b.(1) or 4.3.3.b.(2), notwithstanding the periodic retest schedule of 4.3.3.c.(1), a Type A test must be performed at each refueling outage or every 18 months, whichever occurs first, unless alternative leak test requirements are accepted by the NRC by means of specific exemption from Appendix J per 10CFR50.12. This testing shall be performed until two

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	consecutive periodic Type A tests (not including an immediate retest under (a)) meet the acceptance criteria of 4.3.3.a.(3), 4.3.3.b.(1) and 4.3.3.b.(2), then the retest schedule specified in 4.3.3.c.(1) should be resumed. d. <u>Local Leak Rate-Type B and Type C Tests</u> (1) Primary containment testable penetrations and isolation valves required to be Type B or Type C tested by regulatory requirements, shall be tested at a pressure of 35.0 psig (P_a) each major refueling outage, not to exceed two years, except as provided in (a) and (b) below. (a) Bolted double gasketed seals which shall be tested whenever the seal is closed after being opened and at least at each refueling outage not to exceed a two year interval. (b) Type B tests for primary containment penetrations employing a continuous leakage monitoring system shall be conducted at intervals not to exceed three years.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	(2) When system pressure (Psys) on the opposite side of the isolation valve under test cannot be reduced to atmospheric pressure, then the test pressure shall not be less than P_a + Psys. (3) Personnel airlocks shall be leak tested in accordance with the following: (a) The airlocks shall be tested at a test pressure of 35 psig following a refueling outage or maintenance outage requiring drywell access prior to primary containment integrity being required. (b) Airlocks opened during periods when primary containment integrity is required shall be tested within three days after being opened. For airlock doors opened more frequently than once every three days, the airlocks shall be tested at least once every three days. (c) The airlocks shall be tested every six months at a test pressure of 35 psig. (d) Leakage rate for airlocks shall not exceed $0.05L_a$ at 35 psig.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	(4) Primary containment penetrations and isolation valves that are not defined as Type B or Type C test components (e.g., seal welded cold instrument lines, CRD lines, drywell to wetwell connections, etc.) shall not be individually tested. The penetrations will be considered as integral parts of the Type A test. e. Acceptance Criteria - Type B and Type C Tests The combined leakage rate for penetrations and valves subject to Type B and C tests determined by maximum pathway analysis shall be less than 0.60 L_a. If this value is exceeded, repairs and retests shall be performed to correct the condition. f. Continuous Leak Rate Monitor (1) When the primary containment is inerted, the containment shall be monitored for gross leakage by a weekly review of the inerting system makeup requirements. (2) This monitoring system may be taken out of service for the purpose of maintenance or testing but shall be returned to service as these activities are completed.

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT
	g. <u>Inspection</u> The accessible interior surfaces of the primary containment shall be visually inspected each operating cycle for evidence of deterioration.

BASES FOR 3.3.3 AND 4.3.3 LEAKAGE RATE

The primary containment preoperational test pressures are based upon the calculated primary containment pressure response in the event of a loss-of-coolant accident. The peak drywell pressure would be 35 psig which would rapidly reduce to 22 psig within 100 seconds following the pipe break. The total time the drywell pressure would be above 22 psig is calculated to be about 10 seconds. Following the pipe break, the suppression chamber pressure rises to 22 psig within 10 seconds, equalizes with drywell pressure and thereafter rapidly decays with the drywell pressure decay. [1]

The design pressures of the drywell and suppression chamber are 62 psig and 35 psig, respectively.[2] As pointed out above, the pressure response of the drywell and suppression chamber following an accident would be the same after about 10 seconds. Based on the calculated primary containment pressure response discussed above and the suppression chamber design pressure; primary containment preoperational test pressures were chosen. Also, based on the primary containment pressure response and the fact that the drywell and suppression chamber function as a unit, the primary containment will be tested as a unit rather than testing the individual components separately.

The design basis loss-of-coolant accident was evaluated at the primary containment maximum allowable accident leak rate of 1.9%/day at 35 psig. The analysis showed that with this leak rate and a standby gas treatment system filter efficiency of 90 percent for halogens, 95 percent for particulates, and assuming the fission product release fractions stated in TID-14844, the maximum total whole body passing cloud dose is about 6.0 rem and the maximum total thyroid dose is about 150 rem at the site boundary considering fumigation conditions over an exposure duration of two hours. The resultant doses that would occur for the duration of the accident at the low population distance of 4 miles are lower than those stated due to the variability of meteorological conditions that would be expected to occur over a 30-day period. Thus, the doses reported are the maximum that would be expected in the unlikely event of a design basis loss-of-coolant accident. These doses are also based on the assumption of no holdup in the secondary containment resulting in a direct release of fission products from the primary containment through the filters and stack to the environs. Therefore, the specified primary containment leak rate and filter efficiency (Specification 4.4.4) are conservative and provide margin between expected off-site doses and 10 CFR 100 guideline limits.

The maximum allowable leakage rate (L_a) is 1.5%/day at a pressure of 35 psig (P_a). This value for the test condition was derived from the maximum allowable accident leak rate of about 1.9%/day when corrected for the effects of containment environment under accident and test conditions. In the accident case, the containment atmosphere initially would be composed of steam and hot air depleted of oxygen whereas under test conditions the test medium would be air or nitrogen at ambient conditions. Considering the differences in mixture composition and temperatures, the appropriate correction factor applied was 0.8 and determined from the guide on containment testing.[(3)]

Although the dose calculations suggest that the allowable test leak rate could be allowed to increase to about 3.0%/day before the guideline thyroid dose limit given in 10 CFR 100 would be exceeded, establishing the limit at 1.5%/day provides an adequate margin of safety to assure the health and safety of the general public. It is further considered that the allowable leak rate should not deviate significantly from the containment design value to take advantage of the design leak-tightness capability of the structure over its service lifetime. Additional margin to maintain the containment in the "as built" condition is achieved by establishing the allowable operational leak rate. The operational limit is derived by multiplying the allowable test leak rate by 0.75 thereby providing a 25% margin to allow for leakage deterioration which may occur during the period between leak rate tests.

A reduced pressure test program is used for the integrated test. The test pressures are based on loss-of-coolant accident conditions. The peak primary containment pressure following a loss-of-coolant accident would be 35 psig. This would rapidly reduce to 22 psig. The total time drywell pressure would be above 22 psig would be about 10 seconds. Preoperational integrated leak tests were performed at test pressures at 35 psig and 22 psig. Subsequent integrated tests are performed at a test pressure of 22 psig.

Closure of the containment isolation valves for the purpose of the test is accomplished by the means provided for normal operation of the valves. The reactor is vented to the containment atmosphere during testing.

The acceptance criteria states that the maximum allowable leakage rate (L_t) shall not exceed 1.19 weight percent of the contained air in 24 hours at 22 psig (P_t). This corresponds to the maximum allowable leakage rate (L_a) of 1.5 weight percent at 35 psig (P_a). The maximum allowable test leak rate L_t (at 22 psig) shall not exceed the 1.5%/day times the square root of the ratio of the pressures P_t (at 22 psig) and P_a (at 35 psig), respectively since the ratio of measured leakages for Nine Mile Point Unit 1 is 0.735. The allowable operational leakage rate, L_{to} (at 22 psig) shall not exceed 75 percent of L_t (at 22 psig) and shall be met prior to resumption of power operation following a test.

The primary containment leak rate test frequency is based on maintaining adequate assurance that the leak rate remains within the specification. The leak rate test frequency is based on 10 CFR 50 Appendix J.

The penetration and air purge piping leakage test frequency, along with the containment leak rate tests, is adequate to allow detection of leakage trends. Whenever a double-gasketed penetration (primary containment head equipment hatches and the suppression chamber access hatch) is broken and remade, the space between the gaskets is pressurized to determine that the seals are performing properly. The test pressure of 35 psig is consistent with the accident analyses and the maximum preoperational leak rate test pressure. It is expected that the majority of the leakage from valves, penetrations and seals would be into the reactor building. However, it is possible that leakage into other parts of the facility could occur. Such leakage paths that may affect significantly the consequences of accidents are to be minimized. If the leakage rates of the double-gasketed seal penetrations, testable penetration isolation valves, containment air purge inlets and outlets and the vacuum relief valves are at the maximum specified, they will total 90 percent of the allowed leak rate.[2] Hence, 10 percent margin is left for leakage through walls and untested components.

Leakage from airlocks is measured under accident pressures in accordance with 10 CFR 50 Appendix J.

Monitoring the nitrogen make-up requirements of the inerting system provides a method of observing leak rate trends. This instrumentation equipment must be periodically removed from service for test and maintenance, but this out-of-service time will be kept to a practical minimum.

The test program follows the guidelines stated in the Bechtel Topical Report.[4] This program provides adequate assurance that the test results realistically estimates the degree of containment leakage following a loss-of-coolant accident. The containment leakage rate is calculated using the Absolute Methodology.[8] Containment leakage results are presented in the test report as calculated using the Total Time and Mass Point techniques. The results of local leak rate tests, including, "as-found" and "as-left" leakages, are also included in the containment leak test report.

The specific treatment of selective valve arrangements including the acceptability of the interpretations of 10 CFR 50 Appendix J requirements are given in References 5, 6, and 7. They serve as the bases for alternative test configurations (e.g., reverse accident, multi-valve, water leakage flow tests) as well as relaxations from previous leakage limits or constraints.

References:

(1) FSAR, Volume II, Appendix E

(2) UFSAR, Section VI B.2.1

(3) TID-20583, Leakage Characteristics of Steel Containment Vessels and the Analysis of Leakage Determinations

(4) BN-TOP-1 "Testing Criteria for Integrated Leakage Rate Testing of Primary Containment Structures for Nuclear Power Plants," Revision 1, Bechtel Corporation, November 1, 1972

(5) NRC Safety Evaluation Report dated May 6, 1988, "Regarding Proposed Technical Specifications and Exemption Requests Related to Appendix J."

(6) Niagara Mohawk Letter dated July 28, 1988, "Clarifications, Justifications & Conformance with 10 CFR 50 Appendix J SER."

(7) NRC Letter dated November 9, 1988, "Review of the July 28, 1988 Letter on Appendix J Containment Leakage Rate Testing at Nine Mile Point Unit 1."

(8) ANSI/ANS - 56.8 - 1987, "Containment System Leakage Testing Requirements."

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT

3.3.4 PRIMARY CONTAINMENT ISOLATION VALVES

Applicability:

Applies to the operating status of the system of isolation valves on lines open to the free space of the primary containment.

Objective:

To assure that potential leakage paths from the primary containment in the event of a loss-of-coolant accident are minimized.

Specification:

a. Whenever the reactor coolant system temperature is greater than 215F, all containment isolation valves on lines open to the free space of the primary containment shall be operable except as specified in 3.3.4b below.

b. In the event any isolation valve becomes inoperable the system shall be considered operable provided that within 4 hours at least one valve in each line having an inoperable valve is in the mode corresponding to the isolated condition.

4.3.4 PRIMARY CONTAINMENT ISOLATION VALVES

Applicability:

Applies to the periodic testing requirements of the primary containment isolation valve system.

Objective:

To assure the operability of the primary containment isolation valves to limit potential leakage paths from the containment in the event of a loss-of-coolant accident.

Specification:

The primary containment isolation valves surveillance shall be performed as indicated (See Table 3.3.4)

a. At least once per operating cycle the operable isolation valves that are power operated and automatically initiated shall be tested for automatic initiation and closure times.

b. At least once per quarter all normally open power operated isolation valves shall be fully closed and reopened.

LIMITING CONDITION FOR OPERATION

Table 3.3.4

PRIMARY CONTAINMENT ISOLATION VALVES
LINES ENTERING FREE SPACE OF THE CONTAINMENT

Line or System	No. of Valves (Each Line)	Location Relative To Primary Containment	Normal Position	Motive Power*	Maximum Oper. Time (Sec)	Action on Initiating Signal	Initiating Signal (All Valves Have Remote Manual Backup)
Drywell Vent & Purge							
N_2 Connection (One Line)	1	Outside	Closed	Pn/DC Solenoid	15	Close	Reactor water level low-low or high drywell pressure or high radiation at stack monitoring
	1	Outside	Closed	AC Motor	30	Close	
Air Connection (One Line)	1	Outside	Closed	Pn/DC Solenoid	15	Close	
	1	Outside	Closed	AC Motor	30	Close	
Suppression Chamber Vent & Purge							
N_2 Connection (One Line)	1	Outside	Closed	Pn/DC Solenoid	15	Close	Reactor water level low-low or high drywell pressure or high radiation at stack monitoring
	1	Outside	Closed	AC Motor	30	Close	
Air Connection (One Line)	1	Outside	Closed	Pn/DC Solenoid	15	Close	
	1	Outside	Closed	AC Motor	30	Close	
Drywell N_2 Makeup (One Line)	2	Outside	Closed	Pn/DC Solenoid	60	Close	Reactor water level low-low or drywell high pressure
Suppression Chamber N_2 Makeup (One Line)	2	Outside	Closed	Pn/DC Solenoid	60	Close	Reactor water level low-low or drywell high pressure
Drywell Equipment Drain Line(1) (One Line)	1	Inside	Open	AC Motor	60	Close	Reactor water level low-low or drywell high pressure
	1	Outside	Open	Pn/DC Solenoid	60	Close	
Drywell Floor Drain Line(1) (One Line)	1	Inside	Open	AC Motor	60	Close	
	1	Outside	Open	Pn/DC Solenoid	60	Close	
Vacuum Relief							
Atmosphere to Pressure Suppression System (Three Lines)	1	Outside	Closed	Pn/DC Solenoid	5	Open	Negative pressure relative to atmosphere
	1	Outside	Closed	Self Act. Ck.	---	---	---
Reactor Cleanup System Relief Valve(2) Discharge (One Line to Suppression Chamber)	2	Outside	Closed	Self Act. Ck.	---	---	---

LIMITING CONDITIONS FOR OPERATION

Table 3.3.4 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES
LINES ENTERING FREE SPACE OF THE CONTAINMENT

Line or System	No. of Valves (Each Line)	Location Relative To Primary Containment	Normal Position	Motive Power*	Maximum Oper. Time (Sec)	Action on Initiating Signal	Initiating Signal (All Valves Have Remote Manual Backup)
H_2O_2 #11 Sampling							
Drywell Supply (Two Lines)	2	Outside	Open	Pn/DC Solenoid	60	Close	Reactor Water level low-low or high drywell pressure
Suppression Chamber Supply (One Line)	2	Outside	Open	Pn/DC Solenoid	60	Close	Reactor Water level low-low or high drywell pressure
Drywell Return (One Line)	2	Outside	Open	Pn/DC Solenoid	60	Close	Reactor Water level low-low or high drywell pressure
Suppression Chamber Return (One Line)	2	Outside	Open	Pn/DC Solenoid	60	Close	Reactor Water level low-low or high drywell pressure
H_2O_2 #12 Sampling							
Drywell Supply(1) (Three Lines)	2	Outside	Open	Pn/DC Solenoid	60	Close	Reactor water level low-low or high drywell pressure
Suppression Chamber Supply(1) (One Line)	2	Outside	Open	Pn/DC Solenoid	60	Close	Reactor water level low-low or high drywell pressure
Drywell Return(1) (One Line)	2	Outside	Open	Self Act. Ck.	---	---	---
Suppression Chamber Return(1) (One Line)	2	Outside	Open	Self Act. Ck.	---	---	---

LIMITING CONDITION FOR OPERATION

Table 3.3.4 (continued)

PRIMARY CONTAINMENT ISOLATION VALVES
LINES ENTERING FREE SPACE OF THE CONTAINMENT

Line or System	No. of Valves (Each Line)	Location Relative To Primary Containment	Normal Position	Motive Power*	Maximum Oper. Time (Sec)	Action on Initiating Signal	Initiating Signal (All Valves Have Remote Manual Backup)
Core Spray							
Pump Suction[3] (Four Lines From Suppression Chamber)	1	Outside	Open	AC Motor	90	---	Remote Manual
Pump Discharge[4] (Two Test Lines to Suppression Chamber)	1	Outside	Closed	AC Motor	27	Close	Reactor water level low-low or high drywell pressure
Condensate Supply[4] (Keep Fill) (Two Lines)	2	Outside	Open	Self Act. Ck.	---	---	---
Core Spray High Point Vent[4] (Two Lines)	1 1	Outside Inside	Closed Closed	Pn/DC Solenoid AC Motor	27 27	Close Close	Reactor water level low-low or high drywell pressure
Containment Spray							
Drywell & Suppression Chamber[2] Common Supply (Four Lines)	1	Outside	Open	Pn/DC Solenoid	60	---	Remote Manual
Drywell Branch[2] (Four Lines)	1	Outside	Closed	Self Act. Ck.	---	---	---
Suppression Chamber Branch[2] (One Branch for Each System)	2**	Outside	Closed	Self Act. Ck.	---	---	---
Pump Suction From Suppression Chamber[3] (Four Lines)	1	Outside	Open	AC Motor	70	---	Remote Manual
Containment Spray Test Line to Torus[2] (One Line)	1	Outside	Closed	AC Motor	60	---	Remote Manual
Emergency Cooling Vent to Torus[2] (One Line)	2	Outside	Closed	AC Motor	---	---	Remote Manual

LIMITING CONDITIONS FOR OPERATION

Table 3.3.4 (Continued)

PRIMARY CONTAINMENT ISOLATION VALVES
<u>LINES ENTERING FREE SPACE OF THE CONTAINMENT</u>

Line or System	No. of Valves (Each Line)	Location Relative To Primary Containment	Normal Position	Motive Power*	Maximum Oper. Time (Sec)	Action on Initiating Signal	Initiating Signal (All Valves Have Remote Manual Backup)
<u>Containment Atmosphere Monitoring Supply Line</u> (One Line)	2	Outside	Open	Pn/DC Solenoid	60	Close	Reactor water level low-low or high drywell pressure
<u>Containment Post LOCA Vent</u> (Two Lines)	2	Outside	Closed	Pn/DC Solenoid	60	Close	Reactor water level low-low or high drywell pressure
<u>N2 Purge - TIP Indexers</u>(1) (One Line)	2	Outside	Closed	Self Act. Ck.	---	---	---
<u>Traversing Incore Probe</u>(1) (Four Lines)	1	Outside	Closed	AC Motor	60	Close	Reactor water level low-low or high drywell pressure
<u>Breathing Air Connection</u> (One Line)	1	Inside	Closed	---	---	---	Local Manual
	1	Outside	Closed	---	---	---	Local Manual
<u>Service Water Connection</u>(1) (One Line)	1	Inside	Closed	---	---	---	Local Manual
	1	Outside	Closed	---	---	---	Local Manual
<u>LINES WITH A CLOSED LOOP INSIDE CONTAINMENT VESSELS</u>							
<u>Recirculation Pump Cooling Water</u>(5)							
Supply Line	1	Outside	Open	Self Act. Ck.	---	---	---
Return Line	1	Outside	Open	DC Motor	60	---	Remote Manual
<u>Drywell Cooler Water</u>(5)							
Supply Line	1	Outside	Open	Self Act. Ck.	---	---	---
Return Line	1	Outside	Open	DC Motor	60	---	Remote Manual

Notes:

* Pn - Pneumatically Operated
** One valve in each separate line and one valve in each common line.

(1) These valves do not have to be vented during the Type A test. However, Type C leakage from these valves is added to the Type A test results, if not vented.

(2) These valves are provided with a water seal capability. No Appendix J or IST leakage rate testing is required.

(3) These valves are water leak rate tested and acceptance criteria are established in accordance with the IST Program.

(4) These valves are provided with a water seal. Valves shall be tested during each refuel outage not to exceed two years consistent with Appendix J water seal testing requirements. Leakage rates shall be limited to 0.5 gpm per nominal inch of valve diameter up to a maximum of 5 gpm.

(5) These valves do not meet the requirements of 10CFR50 Appendix J, Section II-H. No testing required.

Double isolation valves are provided on lines penetrating the primary containment and open to the free space of the containment. Closure of one of the valves in each line would be sufficient to maintain the integrity of the pressure suppression system. Except where check valves are used as one or both of a set of double isolation valves, the isolation closure times are presented in Table 3.3.4. Automatic initiation is required to minimize the potential leakage paths from the containment in the event of a loss-of-coolant accident. Details of the isolation valves are discussed in Section VI-D.[1] For allowable leakage rate specification, see Section 3.3.3/4.3.3.

For the design basis loss-of-coolant accident fuel rod perforation would not occur until the fuel temperature reached 1700F which occurs in approximately 100 seconds.[2] A required closing time of 60 seconds for all primary containment isolation valves will be adequate to prevent fission product release through lines connecting to the primary containment.

For reactor coolant system temperatures less than 215F, the containment could not become pressurized due to a loss-of-coolant accident. The 215F limit is based on preventing pressurization of the reactor building and rupture of the blowout panels.

The test interval of once per operating cycle for automatic initiation results in a failure probability of 1.1×10^{-7} that a line will not isolate (Fifth Supplement, p. 115).[3] More frequent testing for valve operability results in a more reliable system.

In addition to routine surveillance as outlined in Section VI-D.1.0[1] 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 the bypass valve. Repressurizing of the individual instruments assures that flow-check valves have reset to the open position.

An in-depth review of the NMP-1 design and operation relative to Appendix J requirements has evaluated the various system/valving configurations.[4] The results of the evaluation and subsequent clarifications[5] are reflected in this specification and its bases.

(1) UFSAR
(2) Nine Mile Point Nuclear Generation Station Unit 1 Safer/Corecool/GESTR-LOCA Loss of Coolant Accident Analysis, NEDC-31446P, Supplement 3, September, 1990.
(3) FSAR
(4) NRC Safety Evaluation Report, dated May 6, 1988, "Regarding Proposed Technical Specifications and Exemption Requests Related to Appendix J."
(5) Niagara Mohawk Letter dated July 28, 1988, "Clarifications, Justifications & Conformance with 10CFR50 Appendix J SER."

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 140 TO FACILITY OPERATING LICENSE NO. DPR-63

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR STATION UNIT NO. 1

DOCKET NO. 50-220

1.0 INTRODUCTION

By letter dated November 20, 1990, as superseded February 7, 1992, as supplemented June 22, 1992, January 29, 1993, February 18, 1993, and March 29, 1993, Niagara Mohawk Power Corporation (the licensee) submitted a request for changes to the Nine Mile Point Nuclear Station Unit No. 1, Technical Specifications (TS). The requested changes would revise the following sections of the TS: (1) Table 3.2.7, Reactor Coolant System Isolation Valves; (2) Table 3.2.7.1, Primary Coolant System Pressure Isolation Valves; (3) 3.3.3/4.3.3, Leakage Rate; (4) 3.3.4, Primary Containment Isolation Valves; (5) Table 3.3.4, Primary Containment Isolation Valves Lines Entering Free Space of the Containment; and (6) associated Bases. The requested changes would update the reactor coolant system and the primary containment isolation valve tables and the containment leakage rate testing requirements to reflect the NRC staff's conclusions as described in the NRC staff's safety evaluation (SE) dated May 6, 1988, as supplemented by the licensee's letter dated July 28, 1988, and the NRC staff's letter dated November 9, 1988, regarding compliance with the requirements of 10 CFR Part 50, Appendix J, at Nine Mile Point Unit 1 (NMP-1). The NRC staff reviewed the requested changes and additional information requested by letter dated November 30, 1992. The licensee responded by letters dated January 29, 1993; February 18, 1993; and March 29, 1993. The discussions below provide the NRC staff's evaluation of the requested changes, including responses to the request for additional information. The licensee's February 7, 1992, application superseded an application from the licensee dated November 20, 1990. The June 22, 1992, January 29, 1993, February 18, 1993, and March 29, 1993, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 EVALUATION

TS Table 3.2.7, Reactor Coolant System Isolation Valves

TS Table 3.2.7 regarding reactor coolant system isolation valves has been revised to incorporate the following:

(1) The initiating signals for the main steam line and the emergency cooling high point vent to main steam line isolation valves have been identified as reactor water level low-low or low reactor pressure (with mode switch in run) or main steam line high radiation, or main steam line high flow, or low-low-low condenser vacuum, or high temperature in the steam tunnel.

(2) The initiating signals for the isolation valves of the emergency cooling steam line drain to main steam line have been identified as high system flow or reactor water level low-low or main steam line isolation.

(3) Two emergency cooling high point vent lines which will be isolated on reactor water level low-low or main steam line isolation have been added.

(4) The initial signals for the cleanup system have been revised to show isolation valve closure on reactor water level low-low or high area temperature or liquid poison initiation.

The TS changes proposed in items (1), (2), (3), and (4) above will provide diversity in the parameters sensed for the initiation of containment isolation in accordance with the requirements as described in Item II.E.4.2 of NUREG-0737. Therefore, the NRC staff finds them acceptable.

(5) A new footnote (1) indicating the isolation valves which do not have to be vented during the Type A test has been added to the main steam, feedwater, reactor cleanup, liquid poison, shutdown cooling, scram discharge volume system vent, scram discharge volume drain, post accident sampling, reactor recirculation system sampling, emergency cooling steam leaving reactor, and emergency cooling condensate return to reactor lines. This footnote is consistent with the NRC staff's conclusion described in its SE dated May 6, 1988. Therefore, the NRC staff finds it acceptable.

(6) Remote manual capability for initiating the AC motor operated isolation valve closure has been added to the feedwater lines. The NRC staff finds that for each feedwater line, there are two containment isolation valves (one self-acting check valve and one AC motor-operated gate valve). During a loss-of-coolant accident (LOCA), it is desirable to maintain reactor coolant make-up from all sources of supply. This AC motor operated valve will not automatically actuate for isolation upon receipt of the signal from the protection system. However, if there is a need for system or containment isolation, this AC motor-operated valve will be actuated remote-manually. Thus, the NRC staff finds this proposed remote manual capability for initiating the AC motor-operated isolation valve closure acceptable.

(7) An automatic initiating signal for opening the emergency condenser condensate return valves following a LOCA has been added. However, TS Table 3.2.7 also indicates that these valves will close upon receipt of the initiating signal from the reactor protection system. The closure signal is initiated by high flow in the emergency cooling system (emergency condenser), and that closure signal takes precedence if both the open and close signals should occur simultaneously. It is appropriate for the closure signal to take precedence over the opening signal. The parameter initiating closure, high emergency condenser system flow, would indicate a breach in the condenser integrity outside containment. If the emergency condenser continued to operate, reactor coolant could continue to be released outside containment. Thus even if an opening signal also occurred in this situation (e.g., on low-low reactor water level), the containment isolation valve should close. The NRC staff finds the proposed TS changes for this item to be acceptable.

(8) The reactor head spray line has been deleted. The licensee indicated that this line was cut and capped and would be Type B tested. The NRC staff finds this proposed TS change acceptable.

(9) Footnote (2) has been added to indicate that there will be flow (which forms a water seal) through the control rod drive hydraulic line isolation valves. Therefore, these isolation valves will receive a water leak rate test in accordance with the inservice test (IST) program. This proposed footnote is consistent with the NRC staff's conclusion described in its SE dated May 6, 1988, therefore; the NRC staff finds it acceptable.

(10) The penetrations, with footnote (5), for core spray condensate supply and core spray system valves have been added to the table. Footnote (5) requires these valves be tested in accordance with TS 4.2.7.1a. The NRC staff finds the addition of these penetrations acceptable.

(11) Footnote (3) for the penetrations of core spray injection isolation valves indicates that the inside core spray injection isolation valves are water sealed during and after an accident and are leak rate tested with water in accordance with the IST program. This part of the proposed footnote is consistent with the NRC staff's conclusion described in its SE dated May 6, 1988; therefore, the NRC staff finds it acceptable. The footnote further indicates that the outside core spray injection isolation valves are electrically locked open with their breakers locked in the off position, and, therefore, these isolation valves do not have to be tested under the IST or Appendix J leakage test programs. The NRC staff finds that the locked open, outside core spray injection valves need not be Type C tested because, in effect, they are not containment isolation valves. The second valve isolation function is provided by further out system valves (40-03 or 40-13) which are included in TS Table 3.2.7.

(12) Footnote (4) has been added to indicate that the isolation valves for the core spray high point vent and core spray pump discharge lines which are water sealed shall be tested during each refueling outage not to exceed two years. Leakage rates shall be limited to 0.5 gpm per nominal inch of valve diameter up to a maximum of 5 gpm. Based on its review, the NRC staff finds this footnote in compliance with the Appendix J requirements and, therefore, acceptable.

(13) A common footnote (1) for the post-accident reactor sampling and reactor recirculation system sampling penetrations indicates that the isolation valves do not have to be vented during Type A test. However, Type C leakage from these valves will be added to the Type A test results. This common footnote for the above penetrations is consistent with the NRC staff's conclusion described in its SE dated May 6, 1988. Therefore, the NRC staff finds it acceptable.

An additional footnote (6) was provided for the post-accident reactor sampling penetration to indicate that the first isolation valve which is a self-actuating check valve will be tested in accordance with Section 4.3.4(c) of the TS. This TS section indicates that the above isolation valve will only be tested for operability at least once per operating cycle. It is difficult, if not impossible, to perform a standard Type C test on an excess-flow check valve with typical Type C testing equipment. The problem is making the valve go closed, which only happens when flow through it exceeds a certain amount. Of course, if a valve is not closed, one cannot measure its leak rate. Therefore, excess-flow check valves are typically not Type C tested, but rather given a functional test of the type that would be required by the proposed footnote. Additionally, the redundant containment isolation valve in this line is included in TS Table 3.2.7, will be Type C tested, and its leak rate added to the Type A test result, in accordance with the NRC staff's SE dated May 6, 1988.

Based on the above considerations, the NRC staff finds the testing of this penetration to be appropriate and the proposed TS to be acceptable.

TS Table 3.2.7.1, Primary Coolant System Pressure Isolation Valves

These valves also serve as containment isolation valves and are also listed in TS Table 3.2.7. The footnotes on TS Table 3.2.7.1 have been revised to specify leakage rates which are in compliance with Appendix J requirements; therefore, the proposed changes are acceptable.

TSs 3.3.3 and 4.3.3, Leakage Rate

The intent of the proposed changes in these TS sections is to make the surveillance requirements for Types A, B, and C leak tests consistent with the

NRC staff's conclusion described in its SE dated May 6, 1988, and Appendix J requirements. The NRC staff's evaluation of the significant proposed changes is provided below:

TS 3.3.3, Leakage Rate - Objectives

The following paragraph has been added to this section as part of the objective for primary containment leakage rate tests:

> "To assure that periodic surveillance of reactor containment penetrations and isolation valves are performed so that proper maintenance and repairs are made during the service life of the containment, and systems and components penetrating primary containment."

The NRC staff finds this proposed addition to the objectives of leakage rate testing acceptable.

TS 4.3.3, Leakage Rate - Specifications

TS 4.3.3.a, Integrated Primary Containment Leakage Rate - Type A Test

The following specifications have been proposed to replace the previous specifications:

(1) Integrated leak rate tests shall be performed at the test pressure (P_t) of 22 psig. Containment pressure shall not be permitted to decrease more than 1 psi below P_t.

(2) Type B and C tests should be completed prior to each Type A test. Type B and C leakages (penalties) not accounted for in the Type A test shall be incorporated as minimum pathway additions to Upper Confidence Limit (UCL) to determine the overall as left integrated leakage rate.

The above proposed TS changes (Items (1) and (2)) are in compliance with the requirements of Appendix J. Therefore, the NRC staff finds them acceptable.

(3) If the leakage rate exceeds the acceptance criterion, corrective action shall be required. If, during the performance of a Type A test, excessive leakage occurs through locally testable penetrations or isolation valves to the extent that it would interfere with satisfactory completion of the test, these leakage paths may be isolated and the Type A re-test continued until completion. The Type A test shall be considered a failed test. A local leakage test shall be performed at P_t before and after the repair of each isolated leakage path. The sum of the post

repaired local leakage rates and the UCL shall be less than 75 percent of the maximum allowable leakage rates, L_t (22). Local leakage rates shall not be subtracted from the Type A test results to determine the acceptability of a test. The as found and as left leakage data values of excessive leakage areas beyond acceptance criteria shall be provided to NRC.

Based on its review, the NRC staff finds the above proposed TS acceptable since it is consistent with the NRC staff's conclusion described in its SE dated May 6, 1988, and the Appendix J requirements.

(4) A Type A test shall last a minimum of eight (8) hours with leakage rates calculated based on "Total Time" method. If a twenty-four (24) hour test is performed the "Mass Point" method will be used to calculate leakage rates. A verification test shall be performed following each Type A test. The verification test provides a method for assuring that systematic error or bias is given adequate consideration. During the verification test, containment pressure may not decrease more than one (1) psi below P_t.

Based on its review, the NRC staff finds this proposed TS and its Bases acceptable.

TS 4.3.3.b, Acceptance Criteria - Type A Test

The following acceptance criteria have been proposed to replace the previous acceptance criteria for Type A tests:

(1) The maximum allowable leakage rate L_t (22) shall not exceed 1.19 weight percent of the contained air per 24 hours at the test pressure of 22 psig (P_t).

(2) The maximum allowable operational leakage rate L_{to} which shall be met prior to power operation following a Type A test (either as measured or following repairs and retest) shall not exceed 0.75 L_t (0.892 weight percent per day).

(3) When adding the leakage rate measured during a Type C test to the results of a Type A test, the leakage rate shall be determined using minimum pathway analysis.

Based on its review, the NRC staff finds the above proposed TS changes (Items (1), (2) and (3)) consistent with the NRC staff's conclusion described in its SE dated May 6, 1988 and, therefore, acceptable.

TS 4.3.3.c, Frequency

The following frequency has been proposed to replace the previous frequency for Type A tests:

(1) Three Type A tests shall be conducted during each ten year service interval at approximately equal intervals. The third test will be conducted when the plant is shutdown for the 10 year inservice inspections.

Based on its review, the NRC staff finds that this proposed TS is consistent with the requirements of Appendix J and, therefore, acceptable.

(2) Retesting

(a) If a Type A test fails to meet the acceptance criteria of 4.3.3.b.(1), a Corrective Action Plan that focuses attention on the cause of the problem shall be developed and implemented. A Type A test that meets the requirements of 4.3.3.a.(3) and 4.3.3.b.(2) is required prior to plant start-up. A report of the corrective action following the failed Type A shall be submitted to the NRC for review and approval with the Containment Leak Test Report.

Based on its review, the NRC staff finds the proposed TS consistent with the requirements of Appendix J and, therefore, acceptable.

(b) If any periodic Type A test fails to meet the acceptance criteria of 4.3.3.b.(1), the test schedule for subsequent Type A tests will be reviewed and approved by the NRC.

Based on its review, the NRC staff finds this proposed TS in compliance with the requirements of the Appendix J and, therefore, acceptable.

(c) If two consecutive periodic Type A tests (not including an immediate retest under (a)) fail to meet the acceptance criteria of 4.3.3.a.(3), 4.3.3.b.(1) or 4.3.3.b.(2), not withstanding the periodic retest schedule of 4.3.3.c.(1), a Type A test must be performed at each refueling outage or every 18 months, whichever occurs first, unless alternative leak test requirements are accepted by the NRC by means of specific exemption from Appendix J per 10 CFR 50.12. This testing shall be performed until two consecutive periodic Type A tests (not including an immediate retest under (a)) meet the acceptance criteria of 4.3.3.a.(3), 4.3.3.b.(1) and 4.3.3.b.(2), then the retest schedule specified in 4.3.3.c.(1) should be resumed.

Based on its review, the NRC staff finds the proposed TS consistent with the requirements of Appendix J and, therefore, acceptable.

4.3.3.d, Local Leak Rate - Type B and Type C Tests

The following TS have been proposed to replace the current TS related to Types B and C tests:

(1) Primary containment testable penetrations and isolation valves required to be Type B or Type C tested by regulatory requirements, shall be tested at a pressure of 35.0 psig (P_a) each major refueling outage, not to exceed two years, except as provided below:

* Bolted double gasket seals which shall be tested whenever the seal is closed after being opened and at least at each refueling outage not to exceed a two year interval.

* Type B tests for primary containment penetrations employing a continuous leakage monitoring system shall be conducted at intervals not to exceed three years.

Based on its review, the NRC staff finds the above proposed TS in accordance with the requirements of Appendix J and, therefore, acceptable.

(2) When system pressure (P_{sys}) on the opposite side of the isolation valve under test cannot be reduced to atmospheric pressure, then the test pressure shall not be less than P_a + P_{sys}.

The NRC staff finds this proposed test pressure in compliance with the requirement of Appendix J and, therefore, acceptable.

(3) Personnel airlocks shall be leak tested in accordance with the following:

(a) The airlocks shall be tested at a test pressure of 35 psig following a refueling outage or maintenance outage requiring drywell access prior to primary containment integrity being required.

Appendix J states, in part, that airlocks opened during periods when containment is not required by the plant's TS shall be tested at the end of such periods at not less than P_a. Therefore, assuming refueling and maintenance outages are the only periods when containment integrity is not required by the TS, the NRC staff finds this proposed TS in compliance with the requirement of Appendix J and acceptable.

(b) Airlocks opened during periods when primary containment integrity is required shall be tested within three days after being opened. For airlock doors opened more frequently than once every three days, the airlocks shall be tested at least once every three days.

Based on its review, the NRC staff finds this proposed TS in accordance with the requirements of Appendix J and, therefore, acceptable.

(c) The airlocks shall be tested every six months at a test pressure of 35 psig.

Based on its review, the NRC staff finds this proposed TS in accordance with the requirements of Appendix J and, therefore, acceptable.

(d) Leakage rate for airlocks shall not exceed 0.05 L_a at 35 psig.

The proposed acceptance criterion of 0.05 L_a at 35 psig is an appropriate number, consistent with standard TS and the TS of many operating plants. Therefore, the NRC staff finds the proposed TS to be acceptable.

(4) Primary containment penetrations and isolation valves that are not defined as Type B or Type C test components (e.g., seal welded cold instrument lines, CRD lines, drywell to wetwell connections, etc.) shall not be individually tested. The penetrations will be considered as integral parts of the Type A test.

Based on its review, the NRC staff finds this proposed TS consistent with the NRC staff's conclusion described in its SE dated May 6, 1988, and the requirements of Appendix J; therefore, it is acceptable.

4.3.3.e, Acceptance Criteria (for Type B and C tests)

The licensee has proposed the following acceptance criteria for Types B and C tests:

> The combined leakage rate for penetrations and valves subject to Type B and C tests determined by maximum pathway analysis shall be less than 0.60 L_a. If this value is exceeded, repairs and retests shall be performed to correct the condition.

The NRC staff finds these proposed acceptance criteria for Types B and C tests in compliance with the requirements described in Appendix J and, therefore, acceptable.

4.3.3.f, Continuous Leak Rate Monitor

(1) The licensee proposes to change the phrase, "...the containment shall be continuously monitored for gross leakage by review of the inerting system makeup requirements...." to "...the containment shall be monitored for gross leakage by a weekly review of the inerting system makeup requirements...."

Based on its review, the NRC staff finds this proposed TS change acceptable.

(2) The licensee proposes to change the sentence, "This monitoring system may be taken out of service for the purpose of maintenance or testing but shall be returned to service as soon as practical." to "This monitoring system may be taken out of service for the purpose of maintenance or testing but shall be returned to service as these activities are completed."

Based on its review, the NRC staff finds this proposed TS change acceptable.

4.3.3.g, Inspection

The licensee proposes to change the sentence, "The accessible interior surface of the drywell shall be visually inspected each operating cycle for evidence of deterioration." to "The accessible interior surfaces of the primary containment shall be visually inspected each operating cycle for evidence of deterioration."

Based on its review, the NRC staff finds this proposed TS change acceptable.

In addition to the above proposed TS changes, the licensee has updated the Bases associated with the above TS Sections 3.3.3 and 4.3.3 to reflect the NRC staff's conclusion described in its SE dated May 6, 1988, and Appendix J requirements. Based on its review, the NRC staff offers no objection to the proposed changes.

3.3.4, Primary Containment Isolation Valves

A. TS 3.3.4b. has been revised to allow 4 hours to isolate a line when an isolation valve becomes inoperable. The TS previously required an isolation valve to be closed if it become inoperable without specifying a time for completing this action.

Based on its review, the NRC staff finds that this 4-hour time limit is consistent with the time allowed by the standard TS. Therefore, the NRC staff finds this proposed TS change acceptable.

B. TS Table 3.3.4 regarding primary containment isolation valves has been revised to incorporate the followings:

(1) The maximum operating time for the drywell and suppression chamber vent and purge valves has been changed from 60 seconds to 15 and 30 seconds for the Pn/DC solenoid (air) and motor-operated valves, respectively. The licensee stated that these closure times are consistent with the NRC staff's requirements to resolve Multi-Plant Action (MPA) Item B-24.

As a part of MPA Item B-24, the NRC staff evaluated the radiological consequences of a LOCA during containment purging at NMP-1. Based on the reduction of the purge isolation valve closure time from the previous value of 60 seconds to 15 seconds, the NRC staff in a SE, dated December 8, 1983, concluded that the radiological consequences of a LOCA during purging at NMP-1 to be within the 10 CFR Part 100 dose guideline values. The NRC staff further stated that, "The staff will, therefore, require NMP-1 to reduce the technical specification limit on purge/vent valves isolation system response time to 15 seconds or less."

In a memorandum for W. R. Butler, thru J. Kudrick, from M. Fields, subject: Motor Operated Containment Isolation Valves in Purge and Vent Lines, dated January 30, 1985, the NRC staff considered the fact that plants with motor-operated purge/vent valves might not achieve full closure of those valves within the 15-second criterion set forth in MPA Item B-24, during a design-basis accident (DBA) with simultaneous loss of offsite power. The NRC staff surveyed approximately 50 operating plants, including NMP-1, and discovered that 11 had motor-operated valves (MOVs) in their purge/vent systems. Several, including NMP-1, were identified as having an air-operated valve in series with each MOV, and the NRC staff concluded that the MPA Item B-24 criterion of complete valve closure within 15 seconds following onset of a DBA LOCA was satisfied for these plants.

The implied basis for this conclusion was that the air-operated valve, with its rapid (15 second) closure time, is sufficient by itself to satisfy the 15-second criterion, despite the fact that the redundant MOV may take longer than 15 seconds to close. Plants were built with one air-operated valve and one MOV in each purge/vent line presumably in an effort to provide diversity and avoid potential common-mode failures that could prevent both valves in a line from isolating at the same time. Further, the NRC staff recognizes that, in fact, large MOVs that close in 15 seconds or less are not available.

Considering the above, the NRC staff finds that the 15-second and 30-second closure times for, respectively, the air-operated and motor-operated purge/vent valves are acceptable. Therefore, the NRC staff finds the proposed TS to be acceptable.

(2) Footnote (1) has been added to the drywell equipment drain line, drywell floor drain line, #12 H_2/O_2 sampling lines, the service water connection line, the N_2 Purge-Tip indexers, and traversing incore probe lines. This footnote indicates that the isolation valves for these lines do not have to be vented during Type A test. However, Type C test leakage from these valves will be added to the Type A test results.

This footnote is consistent with the NRC staff's conclusion described in its SE, dated May 6, 1988; therefore, the NRC staff finds it acceptable.

(3) The suppression chamber water makeup line has been deleted. The licensee indicated that this line had been flanged and designated as a penetration subjected to Type B only. The NRC staff finds this proposed TS change acceptable.

(4) Footnote (2) has been added to indicate that the isolation valves for the reactor cleanup system relief valve discharge, containment spray drywell and suppression chamber common supply, drywell branch and suppression chamber branch, containment spray test line to torus, and emergency cooling vent to torus lines are provided with a water seal capability. No Appendix J testing is required.

This footnote is consistent with the NRC staff's conclusion described in its SE, dated May 6, 1988. Therefore, the NRC staff finds it acceptable.

(5) Footnote (3) has been added to indicate that the penetrations for core spray pump suction and containment spray pump suction from suppression chamber lines are water leak rate tested in accordance with the IST program. The proposed footnote is consistent with the NRC staff's conclusion described in its SE, dated May 6, 1988; therefore, the NRC staff finds it acceptable.

(6) Footnote (4) has been added to indicate that the isolation valves for the core spray high point vent, core spray pump discharge, and condensate supply lines shall be tested during each refuel outage not to exceed 2 years to be consistent with Appendix J water seal testing requirements. Leakage rates shall be limited to 0.5 gpm per nominal inch of valve diameter up to a maximum of 5 gpm. Based on its review, the NRC staff finds this footnote consistent with the NRC staff's conclusion described in its SE, dated May 6, 1988, and therefore, acceptable.

(7) The maximum operating time for the core spray pump discharge test line to the suppression chamber has been reduced from 90 to 27 seconds. The licensee indicated that this proposed TS change is consistent with the Appendix K reload analysis for core spray initiation and flow requirement. Based on its review, the NRC staff finds this proposed TS change acceptable.

(8) The initiating signals for isolating the core spray discharge test line to the suppression chamber has been revised to include the high drywell pressure signal. The NRC staff finds this proposed TS change acceptable.

(9) Footnote (5) has been added to indicate that the valves in closed loops inside containment do not meet the requirements of Appendix J, Section II.H. The NRC staff finds this footnote consistent with the NRC staff's conclusion described in its SE, dated May 6, 1988, and therefore, acceptable.

(10) The maximum operating time for the recirculation pump cooling and drywell cooler water return line DC motor isolation valves has been increased from 30 to 60 seconds. This proposed TS change is within the containment isolation valve closure time limit as described in Section 6.2.4, "Containment Isolation System", of the Standard Review Plan. Therefore, the NRC staff finds it acceptable.

(11) The NRC staff required an editorial modification to Note (2) of Table 3.3.4 of the proposed TS to clarify that no Appendix J or IST leakage rate testing of the affected valves is required. However, the affected valves do require periodic full stroke exercising and stroke time testing in accordance with the requirements of the NMP-1 IST program. This modification to Note (2), which changed "IST testing" to "IST leakage rate testing," was discussed with and agreed to by Mr. Nick Spagnoletti of the licensee's staff during a telephone discussion on March 30, 1993.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New York State official was notified of the proposed issuance of the amendment. The State official had no comments.

4.0 ENVIRONMENTAL CONSIDERATION

The amendment changes a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and changes surveillance requirements. The NRC staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (56 FR 4866 and 57 FR 11111). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendment.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributors:
D. Shum
J. Pulsipher
D. Brinkman

Date: April 12, 1993

Docket No. 50-220

April 12, 1993

Mr. B. Ralph Sylvia
Executive Vice President, Nuclear
Niagara Mohawk Power Corporation
301 Plainfield Road
Syracuse, New York 13212

Dear Mr. Sylvia:

SUBJECT: ISSUANCE OF AMENDMENT FOR NINE MILE POINT NUCLEAR STATION UNIT NO. 1 (TAC NO. M79135)

The Commission has issued the enclosed Amendment No. 140 to Facility Operating License No. DPR-63 for the Nine Mile Point Nuclear Station Unit No. 1 (NMP-1). The amendment consists of changes to the Technical Specifications (TSs) in response to your application transmitted by letter dated November 20, 1990, which was superseded by an application dated February 7, 1992. The February 7, 1992, application was supplemented by letters dated June 22, 1992, and January 29, 1993. The TSs and supporting information in Attachment 2 of the February 7, 1992, application and supplemental letters dated June 22, 1992, and January 29, 1993, were replaced by information provided in a letter dated February 18, 1993, and supplemented by a letter dated March 29, 1993.

The amendment revises TSs 3.2.7.1, 3.3.3, 4.3.3, and 3.3.4 and associated Bases to update these TSs to conform to the requirements of 10 CFR Part 50, Appendix J, and NRC Safety Evaluations (SEs) dated May 6, 1988, and November 9, 1988.

A copy of the related SE is enclosed. A Notice of Issuance will be included in the Commission's next regular biweekly Federal Register notice.

Sincerely,

Original signed by:

Donald S. Brinkman, Senior Project Manager
Project Directorate I-1
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Enclosures:
1. Amendment No. 140 to DPR-63
2. Safety Evaluation

cc w/enclosures:
See next page

LA:PDI-1	PM:PDI-1 DB	BC:SCSB	OGC	D:PDI-1	
CVogan	DBrinkman:smm	RBarrett	EHolcon	RACapra	
4/6/93 4/12	4/6/93	4/7/93	4/8/93	4/12/93	

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