July 12, 2001

Mr. John H. Mueller
Chief Nuclear Officer
Niagara Mohawk Power Corporation
Nine Mile Point Nuclear Station
Operations Building, Second Floor
Lycoming, NY 13093

SUBJECT: NINE MILE POINT NUCLEAR STATION UNIT NO. 2 - ISSUANCE OF AMENDMENT RE: EXCESS FLOW CHECK VALVES SURVEILLANCE TESTING (TAC NO. MB0301)

Dear Mr. Mueller:

The Commission has issued the enclosed Amendment No. 96 to Facility Operating License No. NPF-69 for the Nine Mile Point Nuclear Station, Unit No. 2 (NMP2). The amendment consists of changes to the Technical Specifications in response to your application transmitted by letter dated February 5, 2001, as supplemented by letter dated April 19, 2001.

The amendment revises Section 3.6.1.3, "Primary Containment Isolation Valves," those portions regarding requirements for excess flow check valve surveillance testing.

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

Sincerely,

/RA/
Peter S. Tam, Senior Project Manager, Section 1
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket No. 50-410

Enclosures: 1. Amendment No. 96 to NPF-69
2. Safety Evaluation

cc w/encls: See next page

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Accession No.: ML011800115

OFC	PM:PDI-1	LA:PDI-1	BC(A):PRAB	OGC	SC(A):PDI-1
NAME	PTam	SLittle	MReinhart*	NSt.Amour	RCorreia
DATE	6/29/01	6/29/01	5/15/01	7/10/01	7/12/01

OFFICIAL RECORD COPY
*Memo of 5/15/01 used essentially as-is.

DATED: July 12, 2001

AMENDMENT NO. 96 TO FACILITY OPERATING LICENSE NO. NPF-69, NINE MILE POINT UNIT 2

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S. Little
P. Tam
OGC
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PD plant-specific file
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cc: Plant Service list

NIAGARA MOHAWK POWER CORPORATION

DOCKET NO. 50-410

NINE MILE POINT NUCLEAR STATION, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 96
License No. NPF-69

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

   A. The application for amendment by Niagara Mohawk Power Corporation (NMPC) dated February 5, 2001, as supplemented by letter dated April 19, 2001, 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. 96 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 and shall be implemented prior to startup from Refueling Outage 8, currently scheduled for approximately spring 2002.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

Richard P. Correia, Acting Chief, Section I
Project Directorate I
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment:
Changes to the
Technical Specifications

Date of Issuance: July 12, 2001

ATTACHMENT TO LICENSE AMENDMENT NO. 96

TO FACILITY OPERATING LICENSE NO. NPF-69

DOCKET NO. 50-410

Replace the following page of Appendix A, Technical Specifications, with the attached revised page. The revised page is identified by amendment number and contains a marginal line indicating the area of change.

Remove Page	Insert Page
3.6.1.3-12	3.6.1.3-12

The Technical Specifications Bases document is controlled by the licensee under Technical Specification 5.5.10, “Technical Specification (TS) Bases Control Program.” The NRC staff recognizes that the licensee will issue retyped pages to reflect the changes indicated in the February 5, 2001, application for amendment. These pages are:

B 3.6.1.3-17
B 3.6.1.3-19

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 96 TO FACILITY OPERATING LICENSE NO. NPF-69

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT NUCLEAR STATION UNIT NO. 2

DOCKET NO. 50-410

1.0 INTRODUCTION

By letter dated February 5, 2001, as supplemented by letter dated April 19, 2001, Niagara Mohawk Power Corporation (NMPC), the licensee for Nine Mile Point Nuclear Station, Unit No. 2 (NMP2), submitted a request for changes to the NMP2 Technical Specification (TS). The requested changes would revise the surveillance test requirements for excess flow check valves (EFCVs). By a letter dated March 27, 2001, the NRC requested additional information. NMPC responded by a letter dated April 19, 2001. The April 19, 2001, supplemental letter provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

2.0 BACKGROUND

Excess flow check valves (EFCV) are installed in boiling-water reactor (BWR) instrument lines penetrating the primary containment boundary to limit the release of fluid in the event of an instrument line break. Regulatory Guide (RG) 1.11, “Instrument Lines Penetrating Primary Reactor Containment,” provides guidance on the implementation of General Design Criteria (GDC) 55 and 56 for instrumentation lines that penetrate primary reactor containment and are part of the reactor coolant pressure boundary. As stated by RG 1.11, EFCVs in combination with flow restricting features (line size or orifice) satisfy the requirements of GDC 55 and 56 for automatic isolation capability, maintain the reliability of the connected instrumentation, and ensure the functional performance of secondary containment in the event of an instrumentation line rupture. Examples of EFCV installations include reactor pressure vessel (RPV) level and pressure instrumentation, main steam line flow instrumentation, recirculation pump suction pressure, and reactor core isolation cooling steam line flow instrumentation. EFCVs are not required to close in response to a containment isolation signal and are not required to operate under post loss-of-coolant accident (LOCA) conditions.

NMP2 TS Surveillance Requirement (SR) 3.6.1.3.9 currently requires verification of the actuation capability of each reactor instrumentation line EFCV every 24 months. The SR specifies that each reactor instrumentation line EFCV be operable by verifying that the valve actuates to the isolation position on an actual or simulated instrument line break. The proposed change revises TS SR 3.6.1.3.9 to relax the 24-month EFCV surveillance frequency by limiting the number of tests to a “representative sample” every 24 months such that each EFCV will be tested at least once every 10 years (nominal). The “representative sample” consists of approximately equal numbers of EFCVs being tested every 24 months such that each EFCV is tested at least once every 10 years.

The basis for the proposed change is the high degree of reliability shown by the EFCVs and the low consequences of an EFCV failure. The supporting analysis for the licensee's conclusion is based on General Electric Nuclear Energy (GENE) Topical Report NEDO-32977-A, "Excess Flow Check Valve Testing Relaxation" dated June 2000, which was prepared for the BWR Owner's Group (BWROG). The topical report provided: (1) an estimate of steam release frequency into the reactor building due to a break in an instrument line concurrent with an EFCV failure to close, and (2) an assessment of the radiological consequences of such a release. The BWROG concluded that EFCVs testing intervals could be extended up to 10 years based on the reported reliability and consequence analysis without significantly affecting plant risk. The BWROG suggested a staggered test interval based on actual valve performance with each valve being tested at least once every 10 years. The NRC staff accepted the generic applicability of the topical report by a safety evaluation report (SER) dated March 14, 2000, and agreed that the EFCV test interval could be extended to as much as 10 years. The staff also noted that licensees adopting the topical report must have a failure feedback mechanism and corrective action program to ensure that EFCV performance continues to be bounded by the topical report results. Additionally, each licensee who adopts the topical report is required to perform a plant-specific radiological dose assessment and EFCV failure rate and release frequency analysis to confirm that its facility is bounded by the generic analysis of the topical report.

Meanwhile, the licensee's proposed change would adopt the staff's approved Technical Specification Task Force (TSTF) change to the Improved Standard Technical Specifications (ISTS, NUREG-1433), TSTF-334, Revision 2, "Relaxed Surveillance Frequency for Excess Flow Check Valves Testing." TSTF-334 was approved by the staff on October 31, 2000, by a letter from W. D. Beckner to A. R. Pietrangelo of the Nuclear Energy Institute. It approved specific changes to the ISTS, providing guidance for licensees implementing the extended EFCV surveillance test intervals proposed in the topical report. TSTF-334 is applicable only for those plants for which NEDO-32977-A is applicable and are subject to EFCV performance and corrective action criteria to be developed by the licensee.

3.0 EVALUATION

The staff reviewed the licensee's submittals for conformance to the March 14, 2000, SER to Topical Report NEDO-32977-A, and the guidance of approved TSTF-334, Revision 2. The staff's detailed evaluation follows.

3.1 EFCV Failure Rate and Release Frequency

In NEDO-32977-A, EFCV reliability was evaluated based on testing experience provided by 12 different BWR plants. The composite data indicated that EFCVs are very reliable. The data represented 12,424.5 valve years of operation with a total of 11 failures noted. The EFCV composite failure rate was 1.67E-07/hour and was referenced as the "upper limit" failure rate in the topical report.

The staff noted in its review of the report that the BWROG assumed the EFCV failure rate was constant over time and did not account for potential age-related degradation in the EFCV failure rate. Additionally, the staff questioned the use of an instrument line break frequency based on the NRC report WASH-1400, "Reactor Safety Study: Assessment of Accident Risks in U. S. Commercial Nuclear Power Plants," and not on more current data. The BWROG response included an updated instrument line failure frequency of 3.52E-05 failures/year/valve based on the Electric Power Research Institute's (EPRI) Technical Report No. 100380, "Pipe Failures in U.S. Commercial Nuclear Power Plants," July 1992. This value is 6.6 times greater than the value calculated in NEDO-32977-A using WASH-1400 data. The BWROG response also assumed the observed EFCV failures were five times the actual observed number (55 vs. 11) listed in the topical report. The additional impact of an increase in instrument line failure frequency and a five-fold increase in EFCV failures assumed by the BWROG response demonstrated that release frequencies remained low with limited impact on release frequency.

To estimate the release frequency initiated by an instrument line break, two factors are considered: (1) the instrument line break frequency downstream of the EFCV, and (2) the probability of the EFCV failing to close. The NMP2 data was found to be consistent both in time sampled and EFCV reliability (2 EFCV failures, 87 valves per unit and 1075 valve years operating time) when compared to the topical report data. For the current NMP2 surveillance interval of 24 months, an instrument line break frequency of 5.53E-03/year, and a total plant EFCV failure frequency of 3.06E-03/year, the NMP2 instrument line release (i.e., instrument line break with failure of EFCV to close) frequency is estimated to be 1.69E-05/year. For a surveillance interval of 10 years, the instrument line release frequency estimate increases to 8.47E-05/year. Thus, the relaxed surveillance leads to an increase of release frequency by 6.77E-05/ year. This increase is consistent with the staff SER on NEDO-32977-A, which concluded that an increase in release frequency of 7.3E-05/year was not significant. The NMP2 plant-specific EFCV failure and release rates are also comparable with industry data and the results given in the topical report.

Based on the above, the staff does not consider the estimated increase in release frequency due to the proposed relaxation of EFCV surveillance to be significant.

3.2 Failure Feedback Mechanism and Corrective Action Program

The staff noted that Topical Report NEDO-32977-A does not provide a specific failure feedback mechanism, but does state that a plant's corrective action program must evaluate equipment failures and establish appropriate corrective actions. During review of the topical report, the BWROG responded to the staff's question concerning failure feedback by stating that each licensee who adopts the relaxed surveillance intervals recommended by the topical report should ensure that an appropriate feedback mechanism responsive to EFCV failure trends is in place.

The licensee stated that the NMP2 Maintenance Rule (10 CFR 50.65) Program will be revised to provide a means to track the performance of the EFCVs. To ensure EFCV performance remains consistent with the extended test interval, a minimum performance criterion has been established by the licensee. The criterion specifies less than or equal to 1 functional failure on a 24-month rolling average to ensure that EFCV performance remains consistent with the extended surveillance interval assumptions, and adverse trends in EFCV performance are identified.

Accordingly, the staff considers the licensee's program to account for potential changes in EFCV failure rates to be acceptable and satisfies TSTF-334 performance and corrective action criteria.

3.3 Operational Impact

The operational impact of an EFCV failing to close during the rupture of an instrument line connected to the RPV boundary is the environmental effects of a steam release in the vicinity of the instrument racks in the reactor building. The topical report stated that the magnitude of release through an instrument line would be within the pressure control capacity of reactor building ventilation systems, and that the integrity and functional performance of the secondary containment and Standby Gas Treatment System (SGTS) following an instrument line break would continue to be met. The licensee confirmed that if an EFCV should fail, the restricting orifice limits the steam release to within the pressure control capability of the normal reactor building ventilation system such that an instrument line rupture outside the primary containment will not result in over-pressurizing the secondary containment. The NMP2 Updated Safety Analysis Report (USAR) notes that operator action would be required for plant shutdown and depressurization to terminate the event. The separation of divisional instrument lines and equipment in the reactor building is expected to minimize the operational impact of an instrument line break on other equipment due to jet impingement.

The staff thus found that the operational impact of an EFCV failing to close during rupture of an instrument line has been acceptably addressed by existing plant design.

3.4 Radiological Consequences

The radiological consequences for an instrument line break have been previously evaluated by the licensee in the USAR, Section 15.6.2.5. The analysis does not credit the EFCVs for isolating the break and assumes a discharge of reactor water through an instrument line with a 1/4 inch restricting orifice for the 2-hour duration of the event. No credit is taken for the secondary containment (including the normal reactor building ventilation system) or operation of the SGTS. The postulated radiological consequences of the failure of small lines carrying primary coolant outside containment at NMP2 will continue to be less than a small fraction of the dose guidelines of 10 CFR Part 100 and less than the dose criteria of 10 CFR Part 50, Appendix A, GDC 19. The resulting offsite exposures are a small fraction of the 10 CFR Part 100 limits.

In summary, the radiological consequences of a failure of an EFCV to close after an instrument line rupture is bounded by the licensee's previous analysis, i.e., the radiological dose consequences for an instrument line break are not impacted by the proposed surveillance requirement change.

3.5 Conformance of the Proposed TS to Generic TSTF Guidance

NMP2 SR 3.6.1.3.9 currently requires verification that each reactor instrumentation line EFCV be demonstrated OPERABLE at least once every 24 months by verifying the valve actuates to the isolation position on an actual or simulated instrument line break. The licensee proposed to modify the sentence to read, "Verify a representative sample of reactor instrumentation line EFCVs actuate to the isolation position on an actual or simulated line break signal."

The term "representative sample," as proposed by the topical report and TSTF-334 is not defined in the TS itself. However, the BWROG, in response to the staff's question on this issue stated that the term "representative sample" with an accompanying explanation in the TS Bases, is identical to the usage in the Standard Technical Specifications, NUREG-1433, Revision 1. Specifically, NUREG-1433 uses the term “representative” in SR 3.8.6.3 in reference to battery cell testing, and “representative sample” in SR 3.1.4.2 for verification of control rod scram times. The criterion for “representative sample” and the basis for the nominal 10-year testing interval are provided in the licensee’s submittal, which are similar to Insert 1 and Insert 2 stated in the staff’s approved TSTF-334, Revision 2. Therefore, the application of a “representative sample” for the EFCV testing SR, with an accompanying explanation in the TS Bases, is consistent with TSTF-334, Revision 2 to the STS usage and is therefore, acceptable to the staff.

The licensee included in its submittal, for information, the revised text for the SR 3.6.1.3.9 Bases, including a discussion of the EFCV test frequency and the term “representative sample.” It reads:

> This SR requires a demonstration that a representative sample of reactor instrumentation line EFCVs is OPERABLE by verifying that the valves actuate to the isolation position on an actual or simulated instrument line break condition. The representative sample consists of an approximately equal number of reactor instrumentation line EFCVs, such that each EFCV is tested at least once every 5 refueling cycles. In addition, the reactor instrumentation line EFCVs in the sample are representative of the various plant configurations, models, sizes and operating environments. This ensures that any potentially common problem with a specific type or application of reactor instrumentation line EFCV is detected at the earliest possible time. This SR provides assurance that the reactor instrumentation line EFCVs will perform as designed.
>
> The 24 month frequency is based on the need to perform this Surveillance under the conditions that apply during a plant outage and the potential for an unplanned transient if the Surveillance were performed with the reactor at power. The nominal 10-year interval is based on performance testing as discussed in NEDO-32977-A, “Excess Flow Check Valve Testing Relaxation” (Ref. 8). Furthermore, any reactor instrumentation line EFCV failures will be monitored in accordance with the Maintenance Rule Program to ensure overall reliability is maintained. Appropriate corrective actions will be taken if failures exceed the established performance criteria. Operating experience has demonstrated that these components are highly reliable and that failures to isolate are very infrequent. Therefore, testing of a representative sample was concluded to be acceptable from a reliability standpoint.

The staff found that the licensee’s proposed revision to SR 3.6.1.3.9 and the associated Bases consistent with TSTF-334, Revision 2.

3.6 Summary of Staff Evaluation

As delineated above, the staff determined that the revised SR 3.6.1.3.9 is consistent with TSTF-334 and guidance in Topical Report NEDO-32977-A, that the licensee has a program to account for potential changes in EFCV failure rates, that the operational impact of an EFCV failing to close during rupture of an instrument line has been acceptably addressed by existing plant design, and that the radiological consequences of a failure of an EFCV to close after an instrument line rupture is bounded by the licensee’s previous analysis. The proposed amendment is thus acceptable.

4.0 STATE CONSULTATION

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

5.0 ENVIRONMENTAL CONSIDERATION

The amendment changes requirements with respect to use of facility components 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 staff has previously issued a proposed finding that the amendment involves no significant hazards consideration, and there has been no public comment on such finding (66 FR 15927). 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.

6.0 CONCLUSION

The staff 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 Contributor: C. Doutt

Date: July 12, 2001