OCT 2 4 1975

Docket No. 50-220

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Niagara Mohawk Power Corporation
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Gentlemen:

The Commission has issued the enclosed Amendment No. 3 to Facility License No. DPR-63. This amendment includes Change No. 3 to the Technical Specifications and is in response to your request dated September 17, 1975 and staff discussions.

This amendment requires operability and surveillance of hydraulic snubbers required to protect the primary coolant system and all other safety related systems and components.

Copies of the related Safety Evaluation and the Federal Register Notice are also enclosed.

Sincerely,

/s/

George Lear, Chief
Operating Reactors Branch #3
Division of Reactor Licensing

Enclosures:
1. Amendment No. 3
   w/Change No. 3
2. Safety Evaluation
3. Federal Register Notice

cc w/encls:
See next page

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

NIAGARA MOHAWK POWER CORPORATION

DOCKET NO. 50-220

NINE MILE POINT NUCLEAR STATION, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 3
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 September 17, 1975, 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; and

   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.

2. Accordingly, the license is amended by a change to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.(C).2 of Facility 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, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications, as revised by issued changes thereto through Change No. 3."

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter A. Paulson

for George Lear, Chief
Operating Reactors Branch #3
Division of Reactor Licensing

Attachment:
Change No. 3 to the
Technical Specifications

Date of Issuance: OCT 2 4 1975

ATTACHMENT TO LICENSE AMENDMENT NO. 3

CHANGE NO. 3 TO THE TECHNICAL SPECIFICATIONS

FACILITY OPERATING LICENSE NO. DPR-63

DOCKET NO. 50-220

Replace page iii with the attached revised page. Add pages 241a through 241j.

3.6.4 Hydraulic Snubbers

Applicability

Applies to the operational status of the hydraulic shock suppressors (snubbers).

Objective

To assure the capability of the snubbers to:

Prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient, and

Allow normal thermal motion during startup and shutdown.

Specification

a. During all modes of operation except Cold Shutdown and Refuel, all hydraulic snubbers which are required to protect the primary coolant system or any other safety related system or component shall be operable except as noted in 3.6.4.b through 3.6.4.d below. These safety related hydraulic snubbers are listed in Table 3.6.4.

4.6.4 Hydraulic Snubbers

Applicability

Applies to the periodic testing requirement for the hydraulic shock suppressors (snubbers).

Objective

To assure the operability of the snubbers to perform their intended functions.

3

Specification

The following surveillance requirements apply to all hydraulic snubbers listed in Table 3.6.4.

a. All hydraulic snubbers whose seal material has been demonstrated by operating experience, lab testing or analysis to be compatible with the normal operating environment (<150F) shall be visually inspected to verify their operability in accordance with the following schedule:

Number of Snubbers Found Inoperable During Inspection or During Inspection Interval	Next Required Inspection Interval
0	18 months ± 25%
1	12 months ± 25%
2	6 months ± 25%

3,4	124 days ± 25%
5,6,7	62 days ± 25%
≥ 8	31 days ± 25%

The required inspection interval shall not be lengthened more than one step at a time.

Snubbers may be categorized in two groups, "accessible" or "inaccessible" based on their accessibility for inspection during reactor operation. These two groups may be inspected independently according to the above schedule.

b. From and after the time that a hydraulic snubber is determined to be inoperable, continued reactor operation is permissible only during the succeeding 72 hours unless the snubber is sooner made operable.

b. All hydraulic snubbers whose seal materials have not been demonstrated to be compatible with the normal operating environment (≤150F) shall be visually inspected for operability every 31 days.

c. If the requirements of 3.6.4.a and 3.6.4.b cannot be met, an orderly shutdown shall be initiated and the reactor shall be in a cold shutdown condition within 36 hours.

c. The initial inspection shall be performed within 6 months from the date of issuance of these specifications. For the purpose of entering the schedule in Specification 4.6.4.a, it shall be assumed that the facility had been on a 12 month inspection interval.

Once each refueling cycle, a representative sample of 10 snubbers or approximately 10% of the snubbers, whichever is less, shall be functionally tested for operability including verification of proper piston movement and lock up. For each unit and subsequent unit found inoperable, an additional 10% or ten snubbers shall be so tested until no more failures are found or all units have been tested.

3

LIMITING CONDITION FOR OPERATION	SURVEILLANCE REQUIREMENT	
d. If a hydraulic snubber is determined to be inoperable while the reactor is in the shutdown or refuel mode, the snubber shall be made operable prior to reactor startup.	d. Once each refueling cycle at least two representative snubbers from a relatively severe environment shall be completely disassembled and examined for damage and abnormal seal degradation.	3

TABLE 3.6.4

Inside Containment

Identification No.	System	Elevation
01-HS-1	Main Steam	264'-0½"
01-HS-2	Main Steam	264'-0½"
01-HS-3	Main Steam	264'-0½"
01-HS-4	Main Steam	264'-0½"
01-HS-5	Main Steam	264'-0½"
01-HS-6	Main Steam	264'-0½"
01-HS-7	Main Steam	264'-0½"
01-HS-8	Main Steam	264'-0½"
31-HS-1	Feedwater	263'-6½"
31-HS-2	Feedwater	263'-6½"
32-HS-1	Reactor Recirculation	230'-2 5/8"
32-HS-2	Reactor Recirculation	230'-2 5/8"
32-HS-3	Reactor Recirculation	245'-5 3/8"
32-HS-4	Reactor Recirculation	245'-5 3/8"
32-HS-5	Reactor Recirculation	225'-6"
32-HS-6	Reactor Recirculation	230'-2 5/8"
32-HS-7	Reactor Recirculation	230'-2 5/8"
32-HS-8	Reactor Recirculation	245'-5 3/8"
32-HS-9	Reactor Recirculation	245'-5 3/8"
32-HS-10	Reactor Recirculation	225'-6"
32-HS-11	Reactor Recirculation	230'-2 5/8"
32-HS-12	Reactor Recirculation	230'-2 5/8"
32-HS-13	Reactor Recirculation	245'-5 3/8"
32-HS-14	Reactor Recirculation	245'-5 3/8"
32-HS-15	Reactor Recirculation	225'-6"
32-HS-16	Reactor Recirculation	230'-2 5/8"
32-HS-17	Reactor Recirculation	230'-2 5/8"
32-HS-18	Reactor Recirculation	245'-5 3/8"
32-HS-19	Reactor Recirculation	245'-5 3/8"
32-HS-20	Reactor Recirculation	225'-6"

TABLE 3.6.4

(Continued)

Inside Containment

Identification No.	System	Elevation
32-HS-21	Reactor Recirculation	230'-2 5/8"
32-HS-22	Reactor Recirculation	230'-2 5/8"
32-HS-23	Reactor Recirculation	245'-5 3/8"
32-HS-24	Reactor Recirculation	245'-5 3/8"
32-HS-25	Reactor Recirculation	225'-6"
33-HS-1	Cleanup	263'-6½"
33-HS-2	Cleanup	263'-6½"
33-HS-3	Cleanup	263'-6½"
33-HS-4	Cleanup	263'-6½"
37-HS-1	Reactor Vent and Drain	263'-6"
38-HS-1	Shutdown Cooling	260'-10 5/8"
38-HS-2	Shutdown Cooling	260'-10 5/8"
38-HS-3	Shutdown Cooling	269'-3"
39-HS-9	Emergency Cooling	269'-3"
39-HS-10	Emergency Cooling	269'-3"
39-HS-11	Emergency Cooling	269'-3"
39-HS-12	Emergency Cooling	269'-3"
40-HS-1	Core Spray	240'-0"
40-HS-2	Core Spray	240'-0"
40-HS-3	Core Spray	240'-0"
40-HS-4	Core Spray	261'-6"
40-HS-5	Core Spray	261'-6"
40-HS-6	Core Spray	261'-6"
66-HS-1	Main Steam	265'-3"
66-HS-2	Main Steam	260'-1"
66-HS-3	Main Steam	260'-1"
66-HS-4	Main Steam	232'-10"

3

TABLE 3.6.4

(Continued)

Inside Containment

Identification No.	System	Elevation
66-HS-5	Main Steam	232'-10"
66-HS-6	Main Steam	232'-6"
66-HS-7	Main Steam	232'-6"
66-HS-8	Main Steam	232'-6"
66-HS-9	Main Steam	232'-6"
66-HS-10	Main Steam	232'-6"
66-HS-11	Main Steam	232'-6"
66-HS-12	Main Steam	232'-6"
66-HS-13.	Main Steam	232'-6"
66-HS-14	Main Steam	232'-6"
66-HS-15	Main Steam	232'-6"
66-HS-16	Main Steam	232'-6"
66-HS-17	Main Steam	253'-1"

3

TABLE 3.6.4

Outside Containment

Identification No.	System	Elevation
02-HS-1	Main Steam	243'-6"
02-HS-2	Main Steam	243'-6"
02-HS-3	Main Steam	243'-6"
02-HS-4	Main Steam	243'-6"
02-HS-5	Main Steam	252'-8"
02-HS-6	Main Steam	252'-8"
02-HS-7	Main Steam	252'-8"
02-HS-8	Main Steam	252'-8"
02-HS-9	Main Steam	287'-3"
02-HS-10	Main Steam	287'-3"
02-HS-11	Main Steam	287'-3"
29-HS-1	Feedwater	279'-0"
29-HS-2	Feedwater	301'-0"
29-HS-3	Feedwater	301'-0"
29-HS-4	Feedwater	301'-0"
29-HS-5	Feedwater	301'-0"
29-HS-6	Feedwater	301'-0"
29-HS-7	Feedwater	301'-0"
29-HS-8	Feedwater	305'-6"
29-HS-9	Feedwater	305'-6"
29-HS-10	Feedwater	305'-6"
29-HS-11	Feedwater	301'-0"
29-HS-12	Feedwater	291'-6"
29-HS-13	Feedwater	291'-6"
29-HS-14	Feedwater	291'-6"
29-HS-15	Feedwater	291'-6"
29-HS-16	Feedwater	303'-9"
29-HS-17	Feedwater	303'-9"

3

TABLE 3.6.4

(Continued)

Outside Containment

Identification No.	System	Elevation
30-HS-1	Feedwater	325'-6"
30-HS-2	Feedwater	325'-6"
30-HS-3	Feedwater	325'-6"
30-HS-4	Feedwater	325'-6"
30-HS-5	Feedwater	325'-6"
38-HS-4	Shutdown Cooling	268'-1"
38-HS-5	Shutdown Cooling	268'-1"
38-HS-6	Shutdown Cooling	268'-1"
38-HS-7	Shutdown Cooling	271'-4"
38-HS-8	Shutdown Cooling	270'-10"
39-HS-1	Emergency Cooling	333'-0"
39-HS-2	Emergency Cooling	333'-0"
39-HS-3	Emergency Cooling	318'-0"
39-HS-4	Emergency Cooling	318'-0"
39-HS-5	Emergency Cooling	318'-0"
39-HS-6	Emergency Cooling	318'-0"
39-HS-7	Emergency Cooling	305'-9"
39-HS-8	Emergency Cooling	305'-9"
39-HS-13	Emergency Cooling	308'-2 3/4"
39-HS-14	Emergency Cooling	315'-0"
39-HS-15	Emergency Cooling	315'-0"
39-HS-16	Emergency Cooling	325'-10½"
39-HS-17	Emergency Cooling	334'-6"
39-HS-18	Emergency Cooling	334'-6"
39-HS-19	Emergency Cooling	334'-6"

3

TABLE 3.6.4

(Continued)

Outside Containment

Identification No.	System	Elevation
39-HS-20	Emergency Cooling	341'-3"
39-HS-21	Emergency Cooling	341'-3"
39-HS-22	Emergency Cooling	341'-3"
39-HS-23	Emergency Cooling	341'-3"
39-HS-24	Emergency Cooling	308'-4 3/4"
39-HS-25	Emergency Cooling	315'-0"
39-HS-26	Emergency Cooling	315'-0"
39-HS-27	Emergency Cooling	325'-10½"
39-HS-28	Emergency Cooling	334'-6"
39-HS-29	Emergency Cooling	334'-6"
39-HS-30	Emergency Cooling	334'-6"
39-HS-31	Emergency Cooling	341'-3"
39-HS-32	Emergency Cooling	341'-3"
39-HS-33	Emergency Cooling	341'-3"
39-HS-34	Emergency Cooling	341'-3"
51-HS-1	Feedwater	301'-0"
51-HS-2	Feedwater	313'-6"
51-HS-3	Feedwater	313'-6"
51-HS-4	Feedwater	313'-6"
51-HS-5	Feedwater	291'-0"
51-HS-6	Feedwater	291'-0"
51-HS-7	Feedwater	291'-0"
51-HS-8	Feedwater	291'-0"
51-HS-9	Feedwater	296'-1"
51-HS-10	Feedwater	291'-6"
51-HS-11	Feedwater	291'-6"
51-HS-12	Feedwater	291'-6"

3

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient, while allowing normal thermal motion during startup and shutdown. The consequence of an inoperable snubber is an increase in the probability of structural damage to piping as a result of a seismic or other event initiating dynamic loads. It is therefore required that all hydraulic snubbers required to protect the primary coolant system or any other safety system or component be operable during reactor operation.

Because the snubber protection is required only during relatively low probability events, a period of 72 hours is allowed for repairs or replacements. In case a shutdown is required, the allowance of 36 hours to reach a cold shutdown condition will permit an orderly shutdown consistent with standard operating procedures. Since plant startup should not commence with knowingly defective safety related equipment, Specification 3.6.4.d prohibits startup with inoperable snubbers.

All safety related hydraulic snubbers are visually inspected for overall integrity and operability. The inspection shall include verification of proper orientation, adequate hydraulic fluid level and proper attachment of snubber to piping and structures.

The inspection frequency is based upon maintaining a constant level of snubber protection. Thus the required inspection interval varies inversely with the observed snubber failures. The number of inoperable snubbers found during a required inspection determines the time interval for the next required inspection. Inspections performed before that interval has elapsedshallbe used as a new reference point to determine the next inspection. However, the results of such early inspections performed before the original required time interval has elapsed (nominal time less 25%) shall not be used to lengthen the required inspection interval. Any inspection whose results require a shorter inspection interval will override the previous schedule.

Experience at our operating facility has shown that the required surveillance program should assure an acceptable level of snubber performance because the seal materials have demonstrated their compatibility with the normal operating environment. To date, six years operating experience, has resulted in no seal failures. Also, laboratory test data are available on seal material for all snubbers installed as of September 1975. These data show their compatibility with the normal operating environment.

To increase the assurance of snubber reliability, functional tests shall be performed once each refueling cycle. These tests will include stroking of the snubbers to verify proper piston movement and lock-up. Ten percent or ten snubbers, whichever is less, represents an adequate sample for such tests. Observed failures on these samples should require testing of additional units. Snubbers in high radiation areas or those especially difficult to remove need not be selected for functional tests provided operability was previously verified. All snubbers inside the primary containment are not accessible for surveillance testing during normal reactor operating and are defined as "inaccessible". Only those snubbers in low radiation areas outside of the primary containment are considered "accessible" during normal reactor operation. To complement the visual external inspections, disassembly and internal examination for component damage and abnormal seal degradation shall be performed. The examination of two units, each refueling cycle, selected from relatively severe environments should adequately serve this purpose. Any observed wear, breakdown or deterioration will provide a basis for additional inspections.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 3 TO FACILITY OPERATING LICENSE NO. DPR-63

(CHANGE NO. 3 TO THE TECHNICAL SPECIFICATIONS)

NIAGARA MOHAWK POWER CORPORATION

NINE MILE POINT UNIT 1

DOCKET NO. 50-220

INTRODUCTION

During the summer of 1973, inspections at two reactor facilities revealed a high incidence of inoperable hydraulic shock suppressors (snubbers) manufactured by Bergen Paterson Pipesupport Corporation. As a result of those findings, the Office of Inspection and Enforcement required each operating reactor licensee to immediately inspect all Bergen Paterson snubbers utilized on safety systems and to reinspect them 45 to 90 days after the initial inspection. Snubbers supplied by other manufacturers were to be inspected on a lower priority basis.

Since a long term solution to eliminate recurring failures was not immediately available, the Division of Reactor Licensing sent a letter, dated October 1, 1973, to Niagara Mohawk Power Corporation (NMPC) specifying continuing snubber surveillance requirements and requesting a submittal of proposed Technical Specifications for a snubber surveillance program. On November 18, 1974, NMPC proposed Technical Specifications for hydraulic snubbers at Nine Mile Point Unit 1. Concurrent with our review of NMPC's November 18, 1974 submittal, the NRC staff developed model technical specifications which provide additional assurance of satisfactory snubber operation. The model technical specifications reflect the accumulated industry-wide experience with respect to snubber performance and reliability. By letter dated July 25, 1975, we sent NMPC a copy of the model technical specifications for hydraulic snubbers and requested that NMPC adapt the model to Nine Mile Point Unit 1. On September 17, 1975, NMPC submitted an application for a license amendment which would incorporate the model technical specifications for hydraulic snubbers into Appendix A of Facility Operating License No. DPR-63 for Nine Mile Point Unit 1. Selected modifications to the proposed changes were made with mutual concurrence of the NRC staff and the licensee.

EVALUATION

Snubbers are designed to prevent unrestrained pipe motion under dynamic loads as might occur during an earthquake or severe transient while allowing normal thermal movement during startup and shutdown.

The consequence of an inoperable snubber is an increase in the probability of structural damage to piping resulting from a seismic or other postulated event which initiates dynamic loads. It is, therefore, necessary that snubbers installed to protect safety system piping be operable during reactor operation and be inspected at appropriate intervals to assure their operability.

Examination of defective snubbers at reactor facilities has shown that the high incidence of failures observed in the summer of 1973 was caused by severe degradation of seal materials and subsequent leakage of the hydraulic fluid. The basic seal materials used in Bergen Paterson snubbers were two types of polyurethane; a millable gum polyester type containing plasticizers and an unadulterated molded type. Material tests performed at several laboratories (Reference 1) established that the millable gum polyurethane deteriorated rapidly under the temperature and moisture conditions present in many snubber locations. Although the molded polyurethane exhibited greater resistance to these conditions, it also may be unsuitable for application in the higher temperature environments. Data are not currently available to precisely define an upper temperature limit for the molded polyurethane. The investigation indicated that seal materials are available, primarily ethylene propylene compounds, which should give satisfactory performance under the most severe conditions expected in reactor installations.

An extensive seal replacement program has been carried out at many reactor facilities. Experience with ethylene propylene seals has been very good with no serious degradation reported thus far. Although the seal replacement program has significantly reduced the incident of snubber failures, some failures continue to occur. These failures have generally been attributed to faulty snubber assembly and installation, loose fittings and connections and excessive pipe vibrations. The failures have been observed in both PWRs and BWRs and have not been limited to units manufactured by Bergen Paterson. Because of the continued incidence of snubber failures, we have concluded that snubber operability and surveillance requirements should be incorporated into the Technical Specifications. We have further concluded that these requirements should be applied to all safety related hydraulic snubbers, regardless of manufacturer, in all light water cooled reactor facilities.

(1) Report, H. R. Erickson, Bergen Paterson to K. R. Goller, NRC, October 7, 1974, Subject: Hydraulic Shock Sway Arrestors

We have developed the attached Technical Specifications and Bases to provide additional assurance of satisfactory snubbers performance and reliability. The specifications require that snubbers be operable during reactor operation and prior to startup. Because snubber protection is required only during relatively low probability events, a period of 72 hours is allowed for repair or replacement of defective units before the reactor must be shut down.

An inspection program is specified to provide additional assurance that the snubbers remain operable. The inspection frequency is based upon maintaining a constant level of snubber protection. Thus the required inspection interval varies inversely with the observed snubber failures. The longest inspection interval allowed in the Technical Specifications after a record of no snubber failures has been established is nominally 18 months. Experience at operating facilities has shown that the required surveillance program should provide an acceptable level of snubber performance provided that the seal materials are compatible with the operating environment. Snubbers containing seal material which has not been demonstrated to be compatible with the operating environment are required to be inspected every 31 days until the compatibility is established or an appropriate seal change is completed.

To further increase the level of snubber reliability, the Technical Specifications require functional tests and internal inspections of snubbers at least once each refueling cycle. The tests will verify proper piston movement and lock up, and the internal inspections will monitor for wear, breakdown and deterioration that cannot be observed by the external inspections.

CONCLUSIONS

We have concluded that the proposed additions to the Technical Specifications, as modified, increase the probability of successful snubber performance, increase reactor safety and we therefore find them acceptable.

We have concluded, based on the considerations discussed above, that: (1) because the change does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the change does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: OCT 2 4 1975

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-220

NIAGARA MOHAWK POWER CORPORATION

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

Notice is hereby given that the U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 3 to Facility Operating License No. DPR-63 issued to Niagara Mohawk Power Corporation which revised Technical Specifications for operation of the Nine Mile Point Nuclear Station, Unit 1, located in Oswego County, New York. The amendment is effective as of its date of issuance.

The amendment requires operability and surveillance of hydraulic snubbers required to protect the primary coolant system and all other safety related systems and components.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment is not required since the amendment does not involve a significant hazards consideration.

For further details with respect to this action, see (1) the application for amendment dated September 17, 1975, (2) Amendment No. 3 to License No. DPR-63, with Change No. 3, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Oswego City Library, 120 E. Second Street, Oswego, New York 13126.

A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Reactor Licensing.

Dated at Bethesda, Maryland, this 21st day of October, 1975.

FOR THE NUCLEAR REGULATORY COMMISSION

Walter A. Paulson

Walter A. Paulson, Acting Chief
Operating Reactors Branch #3
Division of Reactor Licensing