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

JAN 18 1985

Docket No.: 50-423

Mr. William G. Counsil
Senior Vice President
Nuclear Engineering and Operations
Northeast Nuclear Energy Company
P. O. Box 270
Hartford, Connecticut 06141-0270

Dear Mr. Counsil:

Subject: Request for Additional Information for Millstone Nuclear Power Station, Unit No. 3

Reference: (1) Letter from Mr. W. G. Counsil to Mr. B. J. Youngblood, Millstone Nuclear Power Station, Unit No. 3 Control Room Design Review Implementation Plan, dated November 10, 1983.

(2) Letter from Mr. B. J. Youngblood to Mr. W. G. Counsil, Request for Additional Information for Millstone Nuclear Power Station, Unit 3 Enclosure 2, dated May 25, 1984.

(3) Letter from Mr. W. G. Counsil to Mr. B. J. Youngblood, Millstone Nuclear Power Station, Unit No. 3 Supplement 1 to NUREG-0737 Control Room Design Review Summary Report, dated November 1, 1984.

(4) Letter from Mr. W. G. Counsil to Mr. B. J. Youngblood, Millstone Nuclear Power Station, Unit No. 3 Supplement 1 to NUREG-0737, Safety Parameter Display System, dated December 7, 1984.

(5) Verrelli, David M., to Mr. E. E. Utley, Meeting Summary - Report No. 50-400/84-40, dated November 26, 1984.

(6) Letter from Mr. H. C. Schmidt to Mr. B. J. Youngblood, Comanche Peak Steam Electric Station (CPSES) Confirmatory Ultrasonic Examination of Cast Stainless Weld at CPSES, dated July 6, 1984.

(7) NUREG-0876 Safety Evaluation Report Supplement No. 5 related to the operation of Byron Station, Units 1 and 2 Commonwealth Edison Company, Sections 5.2.4, 6.6 and Appendix I, dated October 1984.

Enclosure 1 contains requests for additional information which the staff requires to complete its evaluation of your application for an operating license for Millstone 3. These requests for information are related to the staff's Equipment Qualification Branch, Structural and Geotechnical Engineering Branch and Auxiliary Systems Branch review of your FSAR.

DCRDR Audit

Enclosed for your use are the results of the staff's In-Progress Audit of Millstone Nuclear Power Station Unit 3 Detailed Control Room Design Review. This audit was conducted during the period from August 7 to August 10, 1984. The results of the audit showed that your program was significantly modified from the process outlined in the Program Plan (Reference 1). The changes have enhanced the overall DCRDR process and resolved the staff concerns discussed in its comments on the Program Plan (Reference 2). During the audit your staff agreed to document all changes and deviations in the Control Room Design Review Summary Report which you submitted on November 1, 1984 (Reference 3). The staff is currently reviewing this summary report. Since you have indicated that certain portions of the DCRDR have not been completed and a Supplemental Summary Report will be issued, the staff plans to postpone its decision on whether or not to perform an on-site pre-implementation audit.

Phased Implementation of Millstone 3 SPDS

Enclosure 3 contains the staff's reply to your request for NRC concurrence with the concept of a two-phased implementation for the Millstone 3 SPDS (Reference 4). The staff concurs with the described two-phase implementation except for the items described in the enclosure. The staff's concurrence with the concept of phased implementation is based on the assumption that the Phase I SPDS; features to be operable by fuel load, will be designed to satisfy the provisions of Supplement 1 to NUREG-0737. The staff plans to perform an on-site audit of the proposed design to ensure that an acceptable design can be implemented by fuel load.

Ultrasonic Inspection (UT) Demonstration

Enclosure 4 contains two summaries of observations made by the staff's contractors, Battelle and EG&G Idaho during the UT demonstration. Related information (References 5 and 6) is also included in Enclosure 4 for your use. Should you find it necessary to request relief from certain preservice inspection examination requirements in the future you should refer to the information contained in SER Supplement No. 5 for Byron when preparing your relief request (Reference 7). The staff is preparing a request for additional information which will address this subject. Conclusions will be reported in Section 5.2.4 of a future SER Supplement.

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SER Supplement 1

As discussed with your licensing staff, the NRC staff plans to issue the SER Supplement 1 for Millstone 3 in March 1985. Enclosure 5 contains two SER sections which the staff proposes to include in the supplement. This is being transmitted for your information and does not require a response.

Please inform the Millstone 3 Project Manager of your schedule for responding to these requests where appropriate.

For further information or clarification, please contact the Project Manager, Elizabeth L. Doolittle at (301) 492-4911.

Sincerely,

ORIGINAL SIGNED BY:

B. J. Youngblood, Chief
Licensing Branch No. 1
Division of Licensing

Enclosures: As stated

cc: See next page

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ENCLOSURE 1

REQUEST FOR ADDITIONAL INFORMATION

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3

NORTHEAST NUCLEAR ENERGY COMPANY

DOCKET NO. 50-423

REQUEST FOR ADDITIONAL INFORMATION

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3

NORTHEAST NUCLEAR ENERGY COMPANY

DOCKET NO. 50-423

220.0 Structural and Geotechnical Engineering Branch, Structural Engineering Section

220.39 Please provide detailed analysis and calculations performed to evaluate the structural supports (concrete corbel type structures) for the P-1 snubber in the A and B cubicles. The analysis and calculations should contain assumptions and their bases, description of analytical techniques, results and conclusions. This information is needed to complete our review of the information you submitted in a letter to Mr. B. J. Youngblood from Mr. W. G. Counsil, dated October 18, 1984.

REQUEST FOR ADDITIONAL INFORMATION
MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3
NORTHEAST NUCLEAR ENERGY COMPANY
DOCKET NO. 50-423

271.0 Equipment Qualification Branch, Seismic and Dynamic Qualification

271.1 Demonstration of operability of the containment purge and vent valves and the ability of these valves to close during a design basis accident is necessary to assure containment isolation. This demonstration of operability is required by NUREG-0737, "Clarification of TMI Action Plan Requirements," II.E.4.2 for containment purge and vent valves which are not sealed closed during operational conditions 1, 2, 3 and 4.

1. For each purge and vent valve covered in the scope of this review, the following documentation demonstrating compliance with the "Guidelines for Demonstration of Operability of Purge and Vent Valves" (attached, Attachment #5) is to be submitted for staff review:

   A. Dynamic Torque Coefficient Test Reports (Butterfly valves only) - including a description of the test setup.

   B. Operability Demonstration or In-situ Test Reports (when used)

   C. Stress Reports

   D. Seismic Reports for Valve Assembly (valve and operator) and associated parts.

   E. Sketch or description of each valve installation showing the following (Butterfly valves only):

      1. direction of flow

      2. disc closure direction

      3. curved side of disc, upstream or downstream (asymetric discs)

      4. orientation and distance of elbows, tees, bends, etc. within 20 pipe diameters of valve

      5. shaft orientation

      6. distance between valves

   F. Demonstration that the maximum combined torque developed by the valve is below the actuator rating.

2. The applicant should respond to the "Specific Valve Type Questions" (attached) which relate to his valve.

REQUEST FOR ADDITIONAL INFORMATION
MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3
NORTHEAST NUCLEAR ENERGY COMPANY
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3. Analysis, if used, should be supported by tests which establish torque coefficients of the valve at various angles. As torque coefficients in butterfly valves are dependent on disc shape aspect ratio, angle of closure flow direction and approach flow, these things should be accurately represented during tests. Specifically, piping installations (upstream and downstream of the valve) during the test should be representative of actual field installations. For example, non-symetric approach flow from an elbow upstream of a valve can result in fluid dynamic torques of double the magnitude of those found for a valve with straight piping upstream and downstream.

4. In-situ tests, when performed on a representative valve, should be performed on a valve of each sinze/type which is determined to represent the worst case load. Worst case flow direction, for example, should be considered.

   For two valves in series where the second valve is a butterfly valve, the effect of non-symetric flow from the first valve should be considered if the valves are within 15 pipe diameters of each other.

5. If the applicant takes credit for closure time vs. the buildup of containment pressure, he must demonstrate that the method is conservative with respect to the actual valve closure rate. Actual valve closure rate is to be determined under both loaded and unloaded conditions and periodic inspection under tech. spec. requirements should be performed to assure closure rate does not increase with time or use.

REQUEST FOR ADDITIONAL INFORMATION
MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3
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The following considerations apply when testing is chosen as a means for demonstrating valve operability:

Bench Testing

A. Bench testing can be used to demonstrate suitability of the in-service valve by reason of its traceability in design to a test valve. The following factors should be considered when qualifying valves through bench testing.

   1. Whether a valve was qualified by testing of an identical valve assembly or by extrapolation of data from a similarly designed valve.

   2. Whether measures were taken to assure that piping upstream and downstream and valve orientation are simulated.

   3. Whether the following load and environmental factors were considered

      a. Simulation of LOCA
      b. Seismic loading
      c. Temperature soak
      d. Radiation exposure
      e. Chemical exposure
      d. Debris

B. Bench testing of installed valves to demonstrate the suitability of the specific valve to perform its required function during the postulated design basis accident is acceptable.

   1. The factors listed in items A.2 and A.3 should be considered when taking this approach.

In-Situ Testing

In-situ testing of purge and vent valves may be performed to confirm the suitability of the valve under actual conditions. When performing such tests, the conditions (loading, environment) to which the valve(s) will be subjected during the test should simulate the design basis accident.

NOTE: Post test valve examination should be performed to establish structural integrity of the key valve/actuator components.

REQUEST FOR ADDITIONAL INFORMATION
CONCERNING POST-FIRE SAFE SHUTDOWN
MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3
NORTHEAST NUCLEAR ENERGY COMPANY
DOCKET NO. 50-423

410.0 Auxiliary Systems Branch

410.32 A recent plant inspection at another facility revealed that for a fire in the control room, isolation transfer switches for certain hot shutdown systems/components had to be switched to the alternate or isolated position prior to damage occurring to these circuits. If this were not accomplished in time, fuses would have to be replaced in order to make the safe shutdown system/component operable. For most of the transfer switches, the situation did not cause a problem since the desired effect after isolation was the deenergization of power. In other instances where the system/component had to be operable or where operation might be required to override a spurious actuation (such as a motor operated valve) replacement of fuses would be required if blown.

Although the present isolation switches at Millstone 3 isolate the required equipment or component from the control room, it has not been demonstrated whether or not it is necessary to replace fuses in order to place the equipment/component in the desired mode of operation or position. In order for us to complete our review, we need to determine whether fuse replacement is necessary for the operation of a safety system after a control room fire; therefore, please provide the following:

a. The results of your review of electrical design drawings for the existing isolation transfer switches to determine where and if this situation exists.

b. If the Millstone 3 design necessitates the changing of fuses to achieve and maintain hot shutdown after a control room fire, provide modifications to existing switches and/or install new isolation switches where necessary to provide redundant fusing such that a blown fuse will not require replacement to achieve and maintain hot shutdown.

c. Provide typical electrical drawings of transfer schemes for process control and instrumentation.

ENCLOSURE 2

RESULTS OF IN-PROGRESS AUDIT OF

MILLSTONE NUCLEAR POWER STATION, UNIT NO. 3

DETAILED CONTROL ROOM DESIGN REVIEW

DOCKET NO. 50-423