NORTHEAST UTILITIES

THE CONNECTICUT LIGHT AND POWER COMPANY
WESTERN MASSACHUSETTS ELECTRIC COMPANY
HOLYOKE WATER POWER COMPANY
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September 14, 1984

Docket No. 50-336
B11286
A03722

Director of Nuclear Reactor Regulation
Attn: Mr. James R. Miller
Operating Reactors Branch #3
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

References: (1) K. L. Heitner letter to W. G. Counsil, dated December 30, 1983.

(2) W. G. Counsil letter to J. R. Miller, dated February 1, 1984.

Gentlemen:

Millstone Nuclear Power Station, Unit No. 2
Followup Actions to Amendment No. 90
to Operating License No. DPR-65

The NRC Staff forwarded Amendment No. 90 to Facility Operating License No. DPR-65 in Reference (1). The amendment authorized Cycle 6 operation for Millstone Unit No. 2. The NRC Safety Evaluation Report (SER) accompanying the amendment addressed various aspects of the core reload including the accident analysis evaluations submitted to support Cycle 6 operation. The Staff documented several concerns relating to the large break loss-of-coolant, steam line break and steam generator tube rupture accident evaluations submitted to support Cycle 6 operation of the plant following the core reload and thermal shield removal.

Northeast Nuclear Energy Company (NNECO) addressed the majority of the concerns identified in the Cycle 6 SER in Reference (2). The Staff concerns were further discussed at length with Mr. Jack Guttman and representatives from Argonne National Laboratories (ANL) during a meeting at NNECO corporate offices in February, 1984.

The purpose of this submittal is to address the concerns identified in Reference (1), as clarified in the February meeting with the NRC and ANL, and to provide the information which NNECO committed to submit during the February meeting. Specifically, the attachment hereto addresses the following issues for Millstone Unit No. 2:

1. Justification for the large break loss-of-coolant evaluation performed to support Cycle 6 operation without the thermal shield.

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2. Discussion of efforts to evaluate the steam line break accident assuming limited mixing.

3. The results of our evaluation of the steam generator tube rupture event considering the effects of potential reactor vessel upper head voiding. (NNECO agreed to provide this information during the February, 1984 meeting with ANL).

4. Conformance to General Design Criterion 17 and its applicability to the steam line break analysis performed to support Cycle 6 operation.

The attached information is intended to resolve the concerns identified by the Staff in Reference (1) and as discussed during the February, 1984 meeting. NNECO intends to review the groundrules document for reload safety analyses with the fuel vendor to ensure that appropriate assumptions are made which reflect the licensing and design bases for Millstone Unit No. 2. Based on the information reviewed to date it is expected that no changes will be required to the reload safety analyses assumptions to reflect the bases on which Millstone Unit No. 2 was licensed to operate.

We wish to point out that differences identified by either ourselves or the Staff between current licensing criteria and historical practices at Millstone Unit No. 2 are not automatically deemed appropriate for inclusion in the Millstone Unit No. 2 licensing basis. In any event, the Cycle 7 reload reanalysis effort will reflect the appropriate plant design and licensing bases.

We trust you find this information satisfactory.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

W. G. Counsil

W. G. Counsil
Senior Vice President

Docket No. 50-336

Attachment

Millstone Nuclear Power Station, Unit No. 2

Additional Information Supporting

the Cycle 6 Reload Safety Analyses

August, 1984

1. <u>Impact of Thermal Shield Removal on the Cycle 6 Large Break LOCA Evaluation</u>

The Reference (1) report provided the results of analyses and evaluations of operation of Millstone Unit No. 2 without a thermal shield. Our fuel vendor performed a detailed review of the large break LOCA model for Millstone Unit No. 2 to determine what impacts the removal of the thermal shield would have on the limiting large break LOCA analysis results. This review coupled with the fuel vendor's extensive experience in the area of accident analyses formed the basis for the conclusions documented in Reference (1). The results of the review of the large break LOCA model for Millstone Unit No. 2 did not justify a need to perform a reanalysis of the accident.

Section 8.2.3 of Reference (1) included a discussion of the impact of removing the thermal shield on the large break LOCA analysis. The thermal hydraulic phenomena associated with the transient were evaluated for the system design change and the results of expected system response changes with respect to peak clad temperature were determined to be negligible or of a beneficial nature (a lower ultimate peak clad temperature). We refer you to Reference (1) for the detailed discussion of these effects. Several effects were identified which will benefit the PCT while other effects were identified which will adversely effect the PCT. The <u>integrated</u> effects, however, would be expected to result in a negligible beneficial change in the PCT calculated for the large break LOCA analysis. The evaluation of the thermal hydraulic phenomena was based on numerous LOCA analyses and sensitivity studies performed by our fuel vendor in which the individual blowdown and reflood hydraulic phenomena were observed. These data supported the qualitative assessment documented in Reference (1).

The conclusions presented in Reference (1) are further substantiated by the results of large break LOCA evaluations performed for another plant of essentially identical design which recently removed its thermal shield.

NNECO is confident that the results of the evaluations of the large break LOCA analysis for the plant design change of removing the thermal shield are accurate. A reanalysis of this event solely for this design change is not considered necessary nor a prudent resource expenditure. The plant design change has been incorporated into the reload safety analysis checklist and will be reflected in future safety analyses for Millstone Unit No. 2.

This information is considered responsive to the Reference (2) request for additional information regarding large break LOCA PCT.

2. <u>Steam Line Break Accident Assuming Limited Mixing</u>

An evaluation of the steam line break event for Millstone Unit No. 2 is underway to address the NRC Staff concern regarding mixing factors assumed for the broken loop. Preliminary results of this evaluation indicate no significant changes to previous conclusions which assumed good mixing.

The analysis is being performed assuming limited mixing between the reactor vessel inlet and outlet loops containing the broken and intact steam generators.

NNECO will provide the Staff with results of this evaluation when they are finalized.

3. Effects of Upper Head Voiding on the Results of the Steam Generator Tube Rupture Event with Increased Safety Valve Blowdown

During the February, 1984 meeting between NNECO, the NRC Staff and ANL representatives, the steam generator tube rupture evaluation was discussed at length. The effects of reactor vessel upper head modeling and steam generator safety valve blowdown characteristics were addressed.

NNECO committed to provide the Staff the results of an evaluation of reactor vessel upper head voiding on the steam generator tube rupture accident assuming a 12% safety valve blowdown. NNECO had previously provided the results of the tube rupture analysis considering upper head voiding in Reference (3); however this evaluation assumed the steam generator safety valves reseat at their opening pressures. The impact of additional safety valve blowdown was addressed in Reference (4). This calculation provides the results considering the combined effects of potential upper head voiding and the worst case safety valve blowdown (12%).

The results of this evaluation are summarized below.

Head voiding would cause a less rapid primary system depressurization and result in an increase in primary-to-secondary system break flow following reactor trip. The slower depressurization is caused by flashing of the hot water contained in the relatively stagnant head region. The calculation was performed with RETRAN02 MOD002. Although this calculation does not specifically represent the upper head in RETRAN, it conservatively bounds the possible increase in flow from the primary system into the steam generator that might be caused by upper head voiding. The maximum post-trip pressure drop between the primary and secondary systems corresponds to the saturation pressure of the upper head region (1600 psia) minus the steam generator safety valve closing pressure (870 psia assuming 12% blowdown). The increased break flow based on these conditions was modeled in RETRAN as a fill junction with a constant flow rate for the time following reactor trip. Since the increased break flow is modeled as a fill junction, the primary system response in this evaluation is essentially the same as that in Reference (4). Prior to reactor trip, the break flow is calculated identically to the Reference (4) calculation. A comparison between the break flow calculated in Reference (4) and this evaluation is shown in Figure 1.

This analysis predicts that the increase in steam flow to the environment caused by head voiding is bounded by 1.2 percent compared to the worst-case analysis presented in Reference (4).

The radiological dose for the worst case presented in Reference (4) is less than 1 Rem. The increase in offsite dose caused by an increase in steam release to the atmosphere of 1.2 percent is small and well within the 10CFR100 limits. It is therefore concluded that the effects of head voiding on the analyzed consequences of the Millstone Unit No. 2 steam generator tube rupture are negligible.

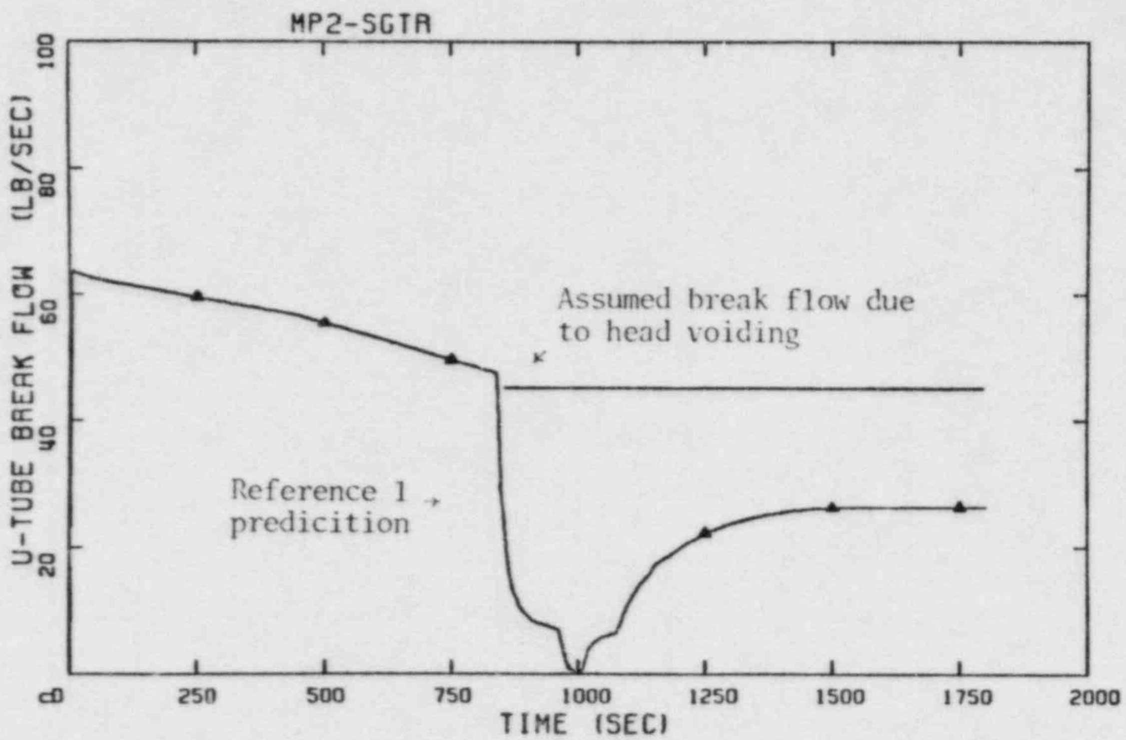
MP2-SGTR
U-TUBE BREAK FLOW (LB/SEC)
100
80
60
40
20
0
250
500
750
1000
1250
1500
1750
2000
TIME (SEC)
Assumed break flow due
to head voiding
Reference 1
predicition

Figure 1

4. <u>Conformance to General Design Criterion 17 for the Steam Line Break Event</u>

Millstone Unit No. 2 was granted an operating license in August, 1975. The NRC Staff Safety Evaluation Report (SER) supporting the operating license was issued on May 10, 1974. Based on a review of the original SER, the NRC Staff (at the time the Atomic Energy Commission Staff) assessed the design of Millstone Unit No. 2 against the General Design Criteria (GDC) in effect as of February 20, 1971. The staff concluded that the plant design conformed to the intent of the GDC.

The Staff review, as documented in the SER, concluded that the onsite and offsite electric power system conformed to the requirements of GDC 17 and 18. Clearly, this conclusion was based upon a combination of information provided by NNECO as well as the Staff's independent review.

It is obvious that the regulatory review criteria have increased in scope and depth since Millstone Unit No. 2 was licensed in 1975. In fact, the Systematic Evaluation Program (SEP) was initiated, in part, as a result of the changes in licensing criteria and eleven of the oldest nuclear power plants were reviewed to determine their level of conformance to current licensing criteria. Millstone Unit No. 2 was not evaluated in the SEP. Plant licensing and safety analyses performed to support reloads as well as other plant design changes have reflected bases established on the docket since the original Staff SER was issued. The volume of documentation is extensive and includes five refuelings, a power uprating, and a change in fuel vendors. The power uprating and fuel vendor change both involved a reanalysis of the transients and accidents included in the licensing basis. In all cases, favorable NRC Staff review and approval for continued operation was obtained, and the bases for those conclusions was clearly articulated.

NNECO's position regarding conformance to the GDC is predicated on the original plant design, as documented in the FSAR, and the Staff review of that design and analysis as documented in the SER, as amended and supplemented in docketed correspondence.

In accordance with 10CFR50.59, NNECO makes changes to the plant design and operation without NRC Staff approval if the changes do not involve an unreviewed safety question or require a change to the Technical Specifications. The tests delineated in 10CFR50.59(a)(2) are performed against the docketed FSAR, as updated annually. When a reload is evaluated under 10CFR50.59, the bases for the conclusions regarding the unreviewed safety question determinations are the transient and accident analyses included <u>in the FSAR</u>. No attempt is automatically made to evaluate how SRP revisions or other regulatory guidance <u>might</u> impact the determinations <u>if</u> they were applied.

The Cycle 6 reload safety analyses were reviewed against the FSAR to determine, in accordance with 10CFR50.59, if the refueling constituted an unreviewed safety question. The Cycle 6 safety analyses were performed in a manner consistent with past practices and methodologies which have been reviewed and approved by the NRC Staff.

Subsequently, the Staff has questioned the appropriateness of assumptions utilized in the Cycle 6 steam line break analysis regarding the availability of off-site power. A review of the impact of assuming loss of off-site power coincident with a steam line break has been performed. The purpose of this review was not to alter the Cycle 6 submittal, but to provide assurance that NNECO has reasonably bounded the worst case for Millstone Unit No. 2. This review has utilized an evaluation which was performed for the Cycle 5 analysis. Because both Cycle 5 and Cycle 6 have very similar boundary conditions, the conclusions of the Cycle 5 evaluation are fully applicable to Cycle 6.

In the case of a steam line break with a loss of off-site power, an additional delay is assumed following the generation of a safety injection signal to start the diesel generators and to commence loading of the safety injection equipment. Since the reactor coolant pumps are coasting down with a loss of off-site power, the ability of the affected steam generator to extract heat from the reactor coolant system is reduced. Considering moderator feedback, a reduction in reactor coolant flow will result in a slower reactivity insertion rate due to the cooldown since the rate of water density increase would be reduced. The minimum approach to criticality will occur later in the transient and the core power increase will be slower than in the similar case with off-site power. In both cases, significant margin to DNB exists. The studies with and without off-site power available, which were performed for Cycle 5, were not carried to completion due to the significant DNB margin available.

The steam line break analysis submittal for Millstone Unit No. 2 Cycle 6 is reasonably representative of the limiting case. The Cycle 5 study indicates that ample margin exists for cases with and without off-site power available and that the impact of assuming loss of off-site power is small.

NNECO has attached a chronology of correspondence which demonstrates the level of review performed for current analysis methodologies and assumptions for the steam line break accident. In each instance, the Staff documents their acceptance of the assumptions regarding off-site power for the steam line break accident. NNECO has accepted these documents for the purpose of establishing an appropriate licensing basis.

Not withstanding the previous discussion, NNECO fully understands and accepts its responsibility to assure conformance to all applicable regulations, including GDC-17. We hereby reiterate our previous position that Millstone Unit No. 2 complies with GDC-17. As we have confirmed in previous discussions, with the Staff, conformance to the GDCs can be achieved in various ways, with many of the variations being a direct function of the age and vintage of the plant in question. For instance, the volume of docketed correspondence demonstrating our Millstone 3 unit's conformance to the regulations is significantly larger than Millstone Unit No. 2. There exists a higher degree of conformance with SRP criteria at Unit 3 as compared to Unit 2. These differences are not, a priori, indicative of any deficiencies at Millstone Unit No. 2. Qualitative, technically-sound, engineering evaluations represent one suitable alternative to quantitative calculations generated using NRC-approved models. A portion of our basis for compliance with GDC-17 utilizes the former technique. NNECO periodically reviews the bases for the transient and accident analyses which support operation of all the Millstone Units. The licensing bases for the three Units are varied with one unit being of SEP vintage and one being an NTOL with an FSAR Chapter 15 consistent with the SRP. As information is made available to NNECO which has a potentially significant impact on the safety analyses assumptions or results for any of the units, NNECO will revise the assumptions utilized in reload safety analyses to conform with the Plant's regulatory basis, if deemed appropriate.

Correspondence Chronology
Steam Line Break Evaluations

1. W. G. Counsil letter to R. Reid, dated March 6, 1980.

   Northeast Nuclear Energy Company (NNECO) dockets the Basic Safety Report (BSR) for Millstone Unit No. 2. The BSR is intended to serve as the reference fuel assembly and safety analysis report for use of Westinghouse fuel assemblies at Millstone Unit No. 2

2. R. A. Clark letter to W. G. Counsil, dated September 18, 1981.

   Additional information requested on the transient and accident analyses presented in the BSR. Question 5a requested NNECO to provide a qualitative discussion of the steam line rupture event assuming a loss of offsite power.

3. W. G. Counsil letter to R. A. Clark, dated October 27, 1981.

   NNECO's response to the September 18, 1981 request for additional information on the BSR transient and accident analyses.

4. W. G. Counsil letter to R. Reid, dated January 25, 1980.

   NNECO dockets the steam line break analysis with automatic auxiliary feedwater initiation. Analysis assumes off-site power available.

5. T. M. Novak letter to W. G. Counsil, dated January 14, 1981.

   Amendment No. 63 to DPR-65 is issued. NRC reviews steam line break analysis supporting automatic auxiliary feedwater initiation and concludes that off-site power availability is a conservative assumption.

6. W. G. Counsil letter to R. A. Clark, dated December 17, 1981.

   NNECO dockets the results of the Cycle 5 steam line rupture accident evaluation assuming offsite power is available.

7. R. A. Clark letter to W. G. Counsil, dated January 12, 1982.

   The NRC Staff documents their review of the transient and accident analyses submitted in the BSR. The Staff concludes that the events analyzed, including the steam line rupture, are acceptable.

8. E. L. Conner letter to W. G. Counsil, dated March 5, 1982.

   The NRC staff issues Amendment No. 74 to DPR-65 for Millstone Unit No. 2 authorizing Cycle 5 operation. Section 2.4.7 of the accompanying safety evaluation report addresses the steam line rupture evaluation for Cycle 5 and concludes that the appropriate analysis has been performed and that the results are acceptable.

References

1. W. G. Counsil letter to J. R. Miller, dated December 12, 1983 (B10968).

2. K. L. Heitner letter to W. G. Counsil, dated December 30, 1983.

3. W. G. Counsil letter to J. R. Miller, dated July 15, 1983.

4. W. G. Counsil letter to J. R. Miller, dated December 12, 1983 (B10960).