NORTHEAST UTILITIES

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October 5, 1979

Docket No. 50-336

Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

References: (1) H. R. Denton letter to W. G. Counsil, dated September 17, 1979.
(2) Amendments 17, 27 and 28 to the FSAR, transmitting the High Energy Pipe Break analysis.

Dear Mr. Denton:

Millstone Nuclear Power Station, Unit No. 2
System Interaction

This letter responds to your September 17, 1979 letter on the subject of a "potential unreviewed safety question on interaction between non-safety grade systems and safety-grade systems". This potential problem was further addressed in IE Information Notice 79-22, issued September 14, 1979.

In conjunction with Combustion Engineering, we have reviewed the specific non-safety grade systems listed in IE Information Notice 79-22, as well as others, for potential interactions that could constitute a substantial safety hazard. This review consisted of the evaluations listed in Attachment I and required a considerable expenditure of resources. Our findings are summarized in Attachment II. Wherever necessary, plant procedures have been modified to ensure operator awareness and operator action that results in acceptable consequences, as further discussed in Attachment II. While, in some cases, we have identified variations from the FSAR, the basic conclusion, that these events do not constitute an undue risk to the health and safety of the public, remains unchanged.

The four interaction scenarios identified in Mr. Denton's letter of September 17, 1979 involved postulated high energy line breaks. EPRI has recently performed an evaluation, determining that the probability of a high energy line break resulting in severe consequences for a typical plant is in the vicinity of 10^{-7} per reactor year. The EPRI report is expected to be sent to the NRC during the week of October 8, 1979 and is entitled, "Probabilistic Analysis of IE Information Notice 79-22 Scenarios". Further, such breaks are more likely to be small cracks

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rather than abrupt failures so that the resulting adverse environment builds up over a period of time providing the potential for detection prior to component failure. Additionally, our review recognized the difference between a demonstrated deficiency (e.g., determination that a control component would operate in a fashion not within the limits presented in the safety analysis) and a potential, unreviewed question. As previously stated, we have not identified any events that would change the conclusions of the FSAR.

As you must recognize, our investigation within the time frame required by your September 17 letter considered generic evaluations coupled with plant-specific, detailed analyses, where required. Based on our initial investigation, continued operation is warranted.

As a result of the Three Mile Island accident, there are a significant number of industry, governmental, and regulatory investigations under way examining the licensing bases and the operating procedures of nuclear generating facilities. These investigations are already identifying areas where studies may result in the consideration of new or revised events as part of the bases for assuring the continued safety of nuclear plants. NUREG-0578 outlines several such events and suggests remedies.

NUREG-0578 requirements for analyses of potential safety problems envision the kinds of scenarios identified by Westinghouse, including the subject of IE Information Notice 79-22. Section 3.2, Page 17 states in part,

> ". . . The NRC requirements for non-safety systems are generally limited to assuring that they do not adversely affect the operation of safety systems . . ."

Further, on Page A-45 of NUREG-0578,

> "Consequential failures shall also be considered . . ."

We, therefore, believe that the scope of the action required by IE Information Notice 79-22 is completed. However, the requirements of NUREG-0578 and any further evaluations required by the NRC will, therefore, be integrated with the planned response sequence for compliance with NUREG-0578.

We are aware of the need to establish a priority of consideration of new issues based upon their potential impact upon the health and safety of the public. Such a priority is required so that the industry's and regulatory resources of skilled engineers and analysts can be applied to the review of the most important concerns first. While the talent resources are extensive, they are limited. We suggest that you consider the establishment of a priority system for TMI-related considerations so that we can assure the public that adequate resources will be available to address those concerns important to safety.

Very truly yours, 1202 271

NORTHEAST NUCLEAR ENERGY COMPANY

W. G. Counsil

W. G. Counsil
Vice President

Attachment

ATTACHMENT 1

OUTLINE OF COMPLETED 79-22 EVALUATIONS

(1) Identification of classes of breaks (location, size fluid, etc.), Reference (2).

(2) Identification of non-safety systems/components which could affect required safety system/component performance.

(3) Determination if failures in these systems could cause the adverse affect identified in Step (2).

(4) Identification of the failure mode of the non-safety grade component and determination as to whether such a failure could be caused by the spectrum of line breaks considered.

(5) Assessment of the magnitude of the detrimental effect.

(6) Determination of the most severe effect for each class of break to identify the limiting consequence.

(7) a. Elimination of adverse consequences by determining that the consequences are acceptable, or

b. Modification of plant procedures to ensure operator awareness and operator action that results in acceptable consequences.

MILLSTONE UNIT NO. 2

CONTROL FUNCTION/EVENT MATRIX

Function	SLB Inside Containment	SLB Outside Containment	FWLB Inside Containment	FWLB Outside Containment	CEA Ejection	LOCA	Comment
Pressurizer Level/Pressure (Heaters, Spray Letdown Charging)	*	--	*	--	--	*	
Power Operated Relief Valve	S	--	S	--	--	S	
CEA position (RRS, CDES)	--	--	--	--	--	--	Manual Control Only
Feedwater Control System	S	X	S	*	--	S	
Boron Control	--	--	--	--	--	--	
Turbine Control	--	*	--	*	--	--	
Turbine Bypass	--	*	--	*	--	--	
Steam Dump to Atmosphere	--	X	--	X	--	--	
Steam Generator Blowdown	--	--	--	--	--	--	
Condenser	--	--	--	--	--	--	

X = Potential Interaction

* = Potential Interaction but not adverse to safety analysis

S = Safety Related Components, No Failure

-- = No Interaction

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MILLSTONE UNIT NO. 2

ATTACHMENT II

Millstone Unit No. 2 Conclusions of the Effects of High Energy Pipe Breaks on Non-Safety Related Control Systems

NNECO has reviewed the effects of high energy pipe breaks on non-safety related control systems, which could possibly effect the results of the plant safety analysis, if these systems were to fail in an adverse direction due to adverse accident environment. As a result of this review, NNECO has concluded that although some variations in the safety analysis results may exist, these variations do not adversely effect the results and conclusions of the safety analyses.

During this analysis, the effects of high energy pipe breaks both inside and outside containment as well as CEA ejection and LOCA were reviewed. The review identified those control systems whose malfunctions could affect the safety analysis. These systems are identified below:

Pressurizer Pressure/Level Control
Power Operated Relief Valves
CEA Position
Feedwater Control Systems
Boron Controls
Turbine Controls
Turbine Bypass
Steam Dump to Atmosphere
Steam Generator Blowdown
Condenser

A review of each of the above control systems, postulating failure from the effects of a high energy pipe rupture, has identified the below-listed systems whose failure could affect the safety analysis.

System	Accident
Steam Dump to Atmosphere	Steam or Feedline Break Outside Containment
Feedwater Control System	Small Steam Break Outside Containment

A detailed discussion of the significance of these interactions follows.

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Steam Dump to Atmosphere

In the original safety analysis, the atmospheric dump valves were assumed to remain closed during analyzed accidents. For this review, it was assumed that a steam line break occurred in the unisolable piping run outside of containment upstream of the isolation valve. The environment caused by this condition results in control malfunctions which cause both atmospheric dump valves to open which results in flow from the broken steam line as well as the atmospheric dump valve in the intact steam line.

This event results in a steam flow rate in excess of that assumed in the accident analysis increases by the amount that the atmospheric dump valves can pass. Each dump valve for Millstone Unit No. 2 can pass 7% of full steam and the flow out the break is limited by the flow restrictors to 300% of full flow from that generator. Thus, the increase in the cooldown rate associated with this event is less than 3%. The minor cooldown increase does not significantly affect the results of the accident analysis for the following reasons: (1) The current analysis does not take credit for safety injection from the charging system which is a qualified ECCS subsystem, thus, mitigating return to criticality concerns; (2) The Moderator Temperature Coefficient (MTC) used in the analysis is conservative with respect to the actual MTC; (3) The slightly larger cooldown has a positive affect in that safety injection from the HPSI pumps may occur sooner.

In order to preclude the possibility of a loss of steam generator inventory in the intact steam generator resulting in a loss of steam applied to the steam driven auxiliary feedwater pump, plant procedures have been modified to alert the operator of the potential for a steam line break outside of containment in one steam line and an open atmospheric dump valve in the other steam line. Plant symptoms for this occurrence would be a rapid decrease in level in one steam generator with a slow decrease in level in the other steam generator and no increase in containment pressure. As a backup in determining that a dump valve has opened, the operators could rely on a visual confirmation of which roof stacks are venting steam.

In order to secure an open vent line and initiate auxiliary feed, procedures have been modified to secure power to the main steam line pressure transmitter. This will cause the atmospheric dump valve to close. As a backup to this method, instrument air to the dump valve will be closed causing the valves to fail closed on loss of air.

A single failure of an MSIV was at first postulated but was later considered non-credible since the MSIV air operator fails closed and has redundant solenoid vents on the air cylinder. A double failure is, therefore, required for MSIV operator failure. The valve itself is a reverse acting check valve. Check valve failure is not considered an active failure.

Feedwater Control System

A malfunction of the feedwater control system during a small steam line break outside containment could possibly produce the following scenario. The steam break causes a slight cooldown of the RCS and temperature feedback effects cause the reactor power to increase slightly such that the reactor does not trip on high power level (< 107%). The steam environment causes the feedwater control system to shut the feedwater regulating valves, thus, causing a loss of feedwater.

This particular event could exceed the bounds of the existing loss of feedwater analysis input in that the power level assumed at trip could be as high as 107%. This condition, however, does not significantly degrade the results of the current safety analysis for the following reasons: (1) The decay heat assumption is ANS + 20%. Since steam generator dryout time is a strong function of decay heat input, this assumption is very conservative; (2) The length of time at the higher power level would not be sufficient to cause an increase in the decay heat input following reactor trip. A calculation using the ANS + 20% curve with infinite irradiation at 107% power shows that ten (10) minutes exist for the operator to initiate auxiliary feedwater.

The possibility of this event occurring is further reduced by operator actions. Indications of power level are available in the control room, and thus, the operator will initiate steps to reduce power or trip the plant.

Conclusions

The results of this review of adverse environment affects from a high energy pipe rupture on control-grade equipment demonstrates that any degradation is of a minor nature. The minor effect of these potential changes plus the considerable margin available in the safety analysis for the incidents involved demonstrate that the adverse failure of the control-grade system is not a safety concern. In addition, the procedure changes implemented to further assist the operator in recognizing the specific occurrences and taking appropriate corrective action further ensure equipment reliability and safety. Therefore, NNECO has conclusively determined that continued operation is justified.

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STATE OF CONNECTICUT)
) ss. Berlin Oct. 5, 1979
COUNTY OF HARTFORD)

Then personally appeared before me W. G. Counsil, who being duly sworn, did state that he is Vice President of Northeast Nuclear Energy Company, a Licensee herein, that he is authorized to execute and file the foregoing information in the name and on behalf of the Licensees herein and that the statements contained in said information are true and correct to the best of his knowledge and belief.

Sheila M. Oates
Notary Public
My Commission Expires March 31, 1981

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