NORTHEAST UTILITIES

THE CONNECTICUT LIGHT AND POWER COMPANY
WESTERN MASSACHUSETTS ELECTRIC COMPANY
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December 8, 1983

Docket No. 50-423
B10961

Director of Nuclear Reactor Regulation
Mr. B. J. Youngblood
Licensing Branch No. 1
Division of Licensing
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

References: D. G. Eisenhut letter to W. G. Counsil, Acceptance Reivew of the Application for an Operating License for Millstone 3, dated January 31, 1983.

Dear Mr. Youngblood:

Millstone Nuclear Power Station Unit No. 3
Response to Acceptance Review Question 440.7

Attached is the response to Acceptance Review question 440.7.

The response contained herein is being provided as it will appear in Amendment 6 which is scheduled to be submitted approximately mid-January, 1984.

If you have any concerns related to commitments contained herein or any questions related to our responses, please contact our Licensing representative directly.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY, ET AL

By NORTHEAST NUCLEAR ENERGY COMPANY, Their Agent

W. G. Counsil

W. G. Counsil
Senior Vice President

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STATE OF CONNECTICUT)
) ss. Berlin
COUNTY OF HARTFORD)

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

Lorraine J. D'Amico
Notary

My Commission Expires March 31, 1988

Question 440.7

For each of the accidents analyzed in Chapter 15, Accident Analyses, show that the most limiting single active component failure or single operator error as defined by the corresponding Chapter 15 SRP in NUREG-0800 has been assumed in the Analysis of each accident.

Response to Question 440.7

All of the ANS condition II transients in Chapter 15.0 are analyzed assuming the most limiting single failure (e.g., loss of one protection signal or SI train failure). Table 440.7 lists the limiting single failures for each ANS condition II event.

The incidents of moderate frequency were analyzed consistent with the acceptance criteria given in the Standard Review Plan concerning peak pressure (less than 110 percent of design), fuel integrity (DNBR limit), generation of more serious plant conditions, and single active failures.

Pressure transients for each event are provided in the FSAR and demonstrate that the pressure remains below 110 percent of design pressure. Fuel cladding integrity is demonstrated for each case by showing that the DNBR remains above the limit value. This is discussed in the results and conclusions sections for each event.

For each transient, its associated worst single failure within the protection system assumed in the FSAR analyses is given in Table 440.7-1. The protection system is defined as those safety functions required to mitigate the consequences of the event. This includes not only the Solid State Protection System (SSPS), but also the Engineered Safeguards Features (ESF) and pressurizer and steam generator safety valves.

These single failures were selected based on the requirements of 10CFR50 Appendix A, the SRP, and Reg. Guide 1.53 (which addresses IEEE-279 and IEEE-379). A single failure is defined as "...an occurrence which results in the loss of capability of a component to perform its intended safety functions." (10CFR50 App. A). The single failure criterion states that a "single failure within the

protection system shall not prevent proper protective action at the system level when required" (IEEE-279).

The single failures which are considered are active failures, consistent with the SRP acceptance criteria. Failures in portions of the protection system which are not required to mitigate the consequences of an accident are not considered. These are failures of systems which are not challenged during the transient and are not active failures. Such failures are independent failures and are therefore not within the scope of incidents of moderate frequency.

For each event listed in Table 440.7-1, a brief discussion of the assumed single failure is provided below. The purpose of these discussions is to justify that the single failure assumed is indeed the worst single failure. These failures are failures at the system level and consider the failure of a protective function. The cause of mechanical nature of the failure which causes the system failure is not discussed, since these are addressed in the FMEA's of the SSPS and ESF and in Chapters 6, 7, and 9 of the FSAR. Therefore, further detail beyond the systems level single failure of loss of one protection train is not provided.

Operation of the steam generator safety valves may be required to prevent an overpressurization of the secondary system. Except where it is already stated in the FSAR, the steam generator valves are not challenged or required to mitigate the consequences of the event. Failures of these valves are not considered since they are not active failures. These independent failures are not applicable. Therefore, failure of these valves is not discussed below unless they are actuated as stated in the FSAR.

Finally, a loss of offsite power is not considered as a single failure for these events. The SRP does not require consideration of a loss of offsite power for the accidents listed in Table 440.7-1 (loss of AC power, 15.2.6, is by definition an exception). Furthermore, no single active failure will cause a loss of offsite power to the emergency buses. Therefore, consideration of this failure is not applicable.

Feedwater Temperature Reduction (15.1.1)

As stated in 15.1.1.1, this event is similar to the effect of increasing steam flow. This is bounded by the events in 15.1.2 and 15.1.3, as stated in 15.1.1.3.

Excessive Feedwater Flow (15.1.2)

As seen in Figure 15.1-1, the pressurizer pressure decreases until the time of turbine trip. The pressure spike results from the choice of a conservative delay between turbine trip and reactor trip; however, the pressurizer PORV's and safety valves do not open. Since they are not required to mitigate the consequences of the event, a single failure in these valves is not applicable and has no impact. Failure of an FIV to close will have no impact since the DNBR is already increasing by the time the FIV closes (Table 15.1-1). The engineered safeguards features are not required for this event; therefore, a single failure of the ESF is not applicable and has no impact. Therefore, the failure of one protection train as listed in Table 440.7-1 is the limiting single active failure.

Excessive Steam Flow (15.1.3)

As stated in 15.1.3.2, the plant reaches a stabilized condition. No reactor trip is required, no pressurizer relief valves are required to reduce pressure (Figures 15.1-3, 15.1-5, 15.1-7, 15.1-9), and no ESF actuation occurs. Since the protection system is not required to function for this event, a single failure does not apply and has no impact.

Inadvertent Secondary Depressurization (15.1.4)

As stated in 15.1.4.1, the failure (opening) of a steam dump, relief, or safety valve initiates the transient. As seen in Figure 15.1-3, this is a depressurization event, therefore pressure relieving functions of the protection system are not challenged nor required to mitigate the consequences of the event. The only portion of the protection system required is the safety injection portion of the ESF. A single failure in a protection train of the signals which actuate SI (15.1.4.1 Item 1) will have no impact due to the redundancy, diversity and independence of the SI actuation signals. The failure of one SI train (listed in

Table 440.7-1) is the limiting single failure since it reduces SI flow, delays the injection of boron to the core, and, consequently allows a "closer" return to criticality. This is the single failure assumed in the FSAR as stated in 15.1.4.2. For this event, the DNB design basis is met by demonstrating that there is no return to criticality (15.1.4.3).

Loss of External Load (15.2.2)

This is bounded by the event described in 15.2.3, as stated in 15.2.2.3 and 15.2.3.1.

Turbine Trip (15.2.3)

Unlike a depressurization transient, for this analysis, the ability to maintain RCS pressure below 110 percent of design per the SRP criterion must be explicitly addressed. Since the DNBR increases with pressure (assuming all other variables are held constant), the event is analyzed with and without pressure control to address both peak pressure and DNBR concerns. As stated in 15.2.3.2, both the pressurizer and steam generator safety valves may be required to operate. Assumptions relative to their operation are described under Items 4 and 5 in the FSAR.

If the pressurizer relief/safety valves fail to close once the pressure has been reduced, there will be no impact on the minimum DNBR. This is because the valves are not required to close until after the time of reactor trip, at which point the DNBR is rising and is very high (see Figures 15.2-1 through 15.2-8). As stated in 15.2.3.2, Item 4, steam relief is obtained by the steam generator safety valves. However, these or any other steam relief valves would not be required to close until after reactor trip, when both the RCS pressure and DNBR are past their maximum and minimum values respectively. Therefore, failure to close would have no impact. Although the ESF may be required to function to supply auxiliary feedwater, a failure in the ESF would have no impact since credit for auxiliary feedwater is not taken (15.2.3.2 Item 6). Therefore, the limiting single failure is one protection train (Table 440.7-1).

Inadvertent Closure of MSIV (15.2.4)

This is bounded by 15.2.3 as stated in the FSAR.

Loss of Condenser Vacuum (15.2.5)

This is bounded by 15.2.3 as stated in the FSAR.

Loss of AC Power (15.2.6)

For this event, the ability of the protection system to provide long term cooling is verified. The loss of one auxiliary feedwater pump of the ESF is the limiting single failure, as stated in Table 440.7-1. A reduction of auxiliary feedwater capacity reduces the capability of the auxiliary feedwater to provide long term cooling. This results in a higher primary side heatup and pressure. The pressure transient of Figure 15.2-9 shows that the pressurizer safety valves are actuated for this event. Failure of the valves to close would have no impact since the auxiliary feedwater is adequately removing the decay heat by that time (Table 15.2-1). For the case where the single active failure is the failure of the pressurizer PORV or safety valve to close, credit can be taken for complete auxiliary feedwater capability. This would reduce the peak pressure and cause the time at which decay heat equals heat removal capability to be sooner. As stated in 15.2.6.1, the steam generator safety and relief valves are used to dissipate decay heat during long term cooling. Since it is desirable to have these valves open, failure to close has no impact, especially since the emergency feedwater supplies sufficient heat removal capability. Single failures which result in loss of signals which actuate auxiliary feedwater, reactor trip, or valve openings, have no impact due to system redundancy, diversity and independence. Therefore, the single failure listed in Table 440.7-1 is the limiting single failure.

Loss of Normal Feedwater (15.2.7)

As for the loss of power event, the primary concern for the loss of normal feedwater is long term cooling capability which is provided by the auxiliary feedwater system. Therefore, as for the loss of AC power, the single active failure causing the loss of one auxiliary feedwater pump is the limiting single

failure, as stated in 15.2.7.2.

Loss of Flow (15.3.1 and 15.3.2)

The protection for this event is discussed in Sections 15.3.1.2 and 15.3.2.2. A single failure in the ESF is not applicable since the ESF are not required to mitigate the consequences of the event. As can be seen in Figures 15.3-2 and 15.3-6, the pressurizer PORV's may open. However, failure to close will have no impact since the point of minimum DNBR is passed and the DNBR is rising by the time the valves close (Figures 15.3-4 and 15.3-8). Therefore, the worst single failure is that of one protection train, as stated in Table 440.7-1.

RCCA Bank Withdrawal from Subcritical(15.4.1)

Although the pressure transient is not shown for this transient, an increase in RCS pressure is expected due to the increase in heat flux and temperature. However, if the PORV's opened and failed to close, there would be no impact on the minimum DNBR since credit for the change (increase) in pressure is not taken in the DNBR analysis. The ESF are not required for this accident, therefore, a single failure of the ESF is not applicable. Therefore, a loss of one protection train is the limiting single failure.

RCCA Bank Withdrawal at Power (15.4.2)

This event is primarily a DNB event and demonstrates the adequacy of the overtemperature T and high flux trips, as stated in 15.4.2.3. Typical transients for the RCCA bank withdrawal at power events are provided in Figures 15.4-4 through 15.4-9. Operation of pressure relieving valves would serve to reduce pressure and thus minimize the DNBR. (If no pressure control was available, the maximum pressure would be limited to that which results in a high pressurizer pressure trip. This is a less limiting pressure transient than those events discussed in 15.2). Failure of valves to close would have no impact, since the point of minimum DNBR is passed by the time the pressure begins to fall (after trip) as seen in the transient figures.

As discussed in 15.4.2.2, for some cases, the steam generator safety valves are opened. The result is to minimize the DNBR, as seen in Figures 15.4-11 and 15.4-12. However, failure to close has no impact since the point of minimum DNBR comes right after reactor trip. Failures in the ESF are not applicable since the ESF are not required. Therefore, the worst single failure is one protection train as stated in Table 440.7-1.

Dropped RCCA (15.4.3)

The worst single failure for this event is the failure of one NIS channel. This results in fewer dropped RCCA's being detected in order to initiate reactor trip via negative flux rate, but has no impact if no trip is generated (i.e., if credit for trip is not taken because of the failure). As can be seen in Figures 15.4-13 through 15.4-15, the plant reaches a new equilibrium condition, and no further protective action is required. Therefore, consideration of other single failures within the protection system is not applicable.

Statically Misaligned (RCCA) (15.4.3)

As stated in Table 440.7-1, no transient analysis is required. Furthermore, no protective functions are required and single failures have no impact.

Inactive RC Pump Startup (15.4.4)

The pressure transient in Figure 15.4-19 shows that the pressurizer PORV's may be challenged for this event. However, failure to close would have no impact, since the point of minimum DNBR is passed by the time the failure could occur (Figure 15.4-20). Failures in the ESF are not applicable since the ESF is not required to mitigate the consequences of the event. Therefore, the limiting single failure is the failure of one protection train, as stated in 440.7-1.

Inadvertent Actuation of the ECCS (15.5.1)

As stated in 15.5.1.1, it is a failure in the ESF which initiates the event. As seen in Figure 15.5-2, this is initially a depressurization event.

The pressure then rises to the PORV setpoint. The PORV's are capable of maintaining system pressure below 110 percent of design. Failure of the PORV's to close would have no impact on the DNBR, since it is already high and never falls below the initial value (figure 15.5.3). Therefore, the failure listed in Table 440.7-1 is the limiting single failure.

Increase in RCS Inventory (15.5.2)

As stated in the FSAR, this is bounded by 15.5.1.

Inadvertent RCS Depressurization (15.6.1)

As stated in 15.6.1.1, a single failure resulting in the opening of a pressurizer PORV or safety valve initiates the transient. Although ESF features might be actuated, they are not required to mitigate the consequences of the event since the DNBR rises after reactor trip. Therefore, ESF failures are not applicable. Therefore, the worst single failure is failure of one protection train.

Failure of Small Lines (15.6.2)

No transient analysis is involved for this event. The protective system is not required to function, since operator action terminates this event as stated in 15.6.2.

TABLE 440.7-1

SINGLE FAILURES ASSUMED FOR ACCIDENT OF MODERATE FREQUENCY

Event Description	Section	Worst Failure Assumed	Effect
Feedwater temperature reduction	15.1.1	(1)	none
Excessive feedwater flow	15.1.2	One protection train	none
Excessive steam flow	15.1.3	(1)	none
Inadvertent secondary depressurization	15.1.4	One safety injection train	delays boron to core
Loss of external load	15.2.2	One protection train	none
Turbine trip	15.2.3	One protection train	none
Inadvertent closure of MSIV	15.2.4	One protection train	none
Loss of condenser vacuum	15.2.5	One protection train	none
Loss of ac power	15.2.6	One auxiliary feedwater pump	increases primary heatup
Loss of normal feedwater	15.2.7	One auxiliary feedwater pump	
Loss of forced reactor coolant flow	15.3.1 and 15.3.2	One protection train	none
RCCA bank withdrawal from subcritical	15.4.1	One protection train	none
RCCA bank withdrawal at power	15.4.2	One protection train	none
Dropped RCCA, dropped RCCA bank	15.4.3	One nuclear instrumentation System channel	none
Statically misaligned RCCA	15.4.3	(2)	none
Inactive RC pump startup	15.4.4	One protection train	none
Inadvertent ECCS operation at power	15.5.1	One protection train	none
Increase in RCS inventory	15.5.2	One protection train	none
Inadvertent RCS depressurization	15.6.1	One protection train	none
Failure of small lines carrying primary coolant outside containment	15.6.2	(2)	none

(1) No protective action required

(2) No transient analysis involved