ATTACHMENT A

MAIN STEAM LINE RUPTURE
ANALYSES FOR YANKEE ROWE

Yankee Atomic Electric Company
Rowe, Massachusetts,
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I. INTRODUCTION AND SUMMARY

Reference (1) requested a review of the main steam line rupture analyses supporting plant operation to determine if the assumptions made in the analyses regarding feedwater system operation were appropriate. Reference (1) specifies four basic concerns. These four concerns are the following:

1) containment pressure response,

2) feedwater system pump (main, condensate, auxiliary) operability,

3) ability to detect and isolate a damaged steam generator, and

4) the potential for core return-to-power.

The remainder of this report provides the results of our review and responds to each of the items specified in Reference (1).

Reference (1) requires a proposed corrective action and a schedule for completion of the corrective action, if any are required as a result of the review. Yankee has identified two of the concerns identified in Reference (1) as possible concerns at Yankee Rowe. At this time it is not clear that these concerns require a design change. However, Yankee will be implementing two design changes that address the concerns of containment overpressurization and potential for return-to-power. These two design changes are auto tripping of condensate pumps on coincidence high containment pressure and low steam line pressure, and ensured boiler feed pump auto trip at power levels greater than 15 MWe. These two design changes will be implemented during the next refueling outage and before the plant goes back on line, respectively. It is important to note that the design change regarding containment response to a main steam line rupture may not be required to ensure acceptable consequences. The second design change is not necessary to prevent a core return-to-power transient. However, in both cases, Yankee feels these design changes are prudent since the changes would lessen the severity of a main steam line rupture. Additionally, emergency procedures will be modified to provide additional assurance of feedwater termination to a damaged steam generator.

A significant amount of the content of safety analyses contained in more recent LWR license applications were not required when Yankee Rowe was licensed. However, Yankee has taken additional steps to ensure that the concerns expressed in Reference (1), which are beyond those events considered or analyzed during the Yankee Rowe licensing process, have been addressed as part of our continuing obligation to ensure the health and safety of the public. We believe the information presented herein is both accurate and responsive to your request. A second level review of the supporting analyses is in progress and will be completed shortly. If our continued review identifies any items that alter our conclusions, we will inform you of the findings. Additionally, the main steam line rupture analyses, both core response and containment response, are being addressed by the NRC under the Systematic Evaluation Program (SEP). The SEP topics are VI-2.D, VI-3 and XV-2.

Main Steam Line Rupture at Yankee Rowe is discussed in Section II. A historical essay on what has been done in this regard and relevant communications with NRC is contained in Section III. Section IV details the review findings with Section V assessing the impact of the review findings. Section VI contains information and the schedule for implementation of the aforementioned changes in design and procedure which resulted from our review.

II. DISCUSSION OF MAIN STEAM LINE RUPTURE AT YANKEE ROWE

Each of the four main steam lines at Yankee Rowe (YR) has a Non-Return Valve (NRV) in the line outside containment. These valves act as check valves to preclude reverse flow and also can be manually closed to preclude forward or reverse flow.

A steam line rupture inside containment results in the blowdown of secondary fluid from the steam generator connected to the ruptured steam line into the containment atmosphere. Any backflow from the unaffected steam generators would be rapidly terminated by closure of the NRV located outside containment in the intact portion of the ruptured steam line. Reactor protection would be assured by a number of trips including high containment pressure, low pressurizer pressure and high neutron flux levels.

A steam line rupture outside containment and downstream of the NRVs would result in blowdown of all four steam generators. Reactor protection would be assured by a number of trips including low pressurizer pressure and high neutron flux levels.

III. HISTORICAL BACKGROUND INFORMATION

Historical information with regard to steam line rupture inside containment is presented in Section III.1. Similar information with regard to steam line rupture outside containment is presented in Section III.2.

III.1 Main Steam Line Rupture Inside Containment

Section 402 of the Yankee Rowe Final Hazards Summary Report describes the vapor container design criteria. The design pressure of the vapor container is greater than the calculated pressure rise following the complete severance of a 20 inch main coolant line with two open ends and the simultaneous rupture of one secondary main steam line. This analysis was redone in 1971 and submitted to the NRC via Reference (2) in support of Safety Injection System modifications.

A licensing analysis of the containment transient resulting from a main steam line rupture only inside containment has not been performed. However, scoping analyses, as part of the YR Systematic Evaluation Program effort, to determine containment response to a main steam line rupture have been performed by Yankee. These analyses, based on RELAP4 blowdown analysis and CONTEMPT-LT026 containment analysis, indicate that the design containment pressure of 34.5 psig is not exceeded for the most severe steam line rupture. The peak containment pressure calculated (31.7 psig) is conservative as long as the feedwater flow assumptions made remain valid. Section IV.2 details the assumptions made regarding feedwater flow in this scoping analysis.

The results of this analysis were previously reported to the NRC in the following correspondence:

1) In support of SEP, quoted from page 5 Attachment A to Reference (3), is the following:

   "Detailed calculations of the containment transient resulting from the rupture of a main steam line inside containment have not been completed at this time. Scoping analyses have, however, been performed and indicate that the peak containment pressure will be less than that resulting from the LOCA DBE. These scoping analyses also indicate that although the peak vapor space temperature may exceed that occurring during the LOCA DBE due to superheating of the blowdown fluid, it will decrease below the LOCA DBE values in the long-term."

2) In response to the NRC letter, Reference (4) concerning non-safety grade equipment qualification, the following is quoted.

   "B.1.1 Affects on V. C. Pressure

   Section 402 of the Yankee Rowe Final Hazards Summary Report describes the vapor container design criteria. The design pressure of the vapor container is greater than the calculated pressure rise following the complete severance of a 20 inch main coolant line with two open ends and the simultaneous rupture of one secondary main steam line. This analysis was redone in 1971 (see Proposed Change No. 96, dated August 6, 1971).

   A licensing analysis of the containment pressure transient resulting from a main steam line rupture only, inside containment has not been performed. However, scoping analyses, as part of the YR SEP effort, to determine containment response to a main steam line rupture have been performed by YAEC. These analyses, based on RELAP4 blowdown analysis and CONTEMPT-LT026 containment analysis, indicate that the design containment pressure of 34.5 psig is not exceeded for the most severe steam line rupture. The blowdown analysis is very conservative since it is based on pure steam blowdown which yields maximum containment pressure and temperature conditions."

3) In Support of SEP topics VI-2.D and VI-3.

III.2 Main Steam Line Rupture Outside Containment

The last complete main steam line rupture outside containment analysis was performed in 1973 in support of the Core XI Reload Submittal and submitted to the NRC via Reference (5). Subsequent cycles, including the present cycle, were licensed via reference to this Core XI analysis because the refueling changes 1) did not impact the thermal hydraulic transient and 2) the core physics parameters were bounded by the Core XI analysis.

IV. REVIEW FINDING

To determine if the concerns specified in Reference (1) impact the analyses supporting Yankee Rowe operation, a four phase review was performed. The four phases consist of the following:

1) Review of Yankee Rowe feedwater system operation including the following:

   a) normal operation - this review provides the boundary conditions (e.g., number of pumps operating at a given power level, feedwater control system and so forth)

   b) normal post trip operation, and

   c) system operation following a main steam line rupture.

2) Review of the containment scoping analysis.

3) Review of the current Steam Line Rupture Analysis performed to determine return-to-power potential.

4) Ability to detect and isolate a damaged steam generator and affects on feedwater pumps.

Sections IV.1 through IV.4 provide a review of each of these four phases. Section IV.5 summarizes the potential problem areas discovered in this review.

IV.1 Review of Yankee Rowe Feedwater System Operation

Feedwater is supplied to the steam generators via the Feedwater System which contains three Boiler Feed Pumps (BFPs), three condensate pumps and two heater drain pumps. The number and combination of pumps operating is dependent on the plant power level. A summary of the review findings of importance to the concerns issued in Reference (1) are provided below.

1) Boiler feed pumps auto trip at power levels >15 MWe.

   Plant Procedure OP-3000, "Emergency Shutdown from Power", requires the operator to close the steam generator feed line valves (backup is air operated feed valve) as an immediate action step. OP-3000 requires affirmation of BFP trip by operators. Based on experience, affirmation of BFP tripping and closure of the feed line valves or air operated feed valves is performed immediately (within approximately 1 minute) for every reactor trip.

2) Boiler feed pumps do not auto trip at power levels <15 MWe.

   However, OP-3000 actions as discussed above ensure rapid feedwater system isolation. Note that only one boiler feed pump operates at this power level.

3) Boiler feed pumps may not trip at any power level for a main steam line rupture outside containment.

   The means of identifying whether the turbine load exceeds 15 MWe is the first stage nozzle pressure. Above 15 MWe, the first stage nozzle pressure closes a contact to a 1.8 second Agastat timer.

This contact remains closed 1.8 seconds after first stage nozzle pressure decreases below setpoint. If a reactor scram occurs during this time, all operating boiler feed pumps auto trip and the condensate pump discharge recirculation valve to the condenser opens.

The concern is that a main steam line rupture could cause a first stage nozzle pressure reduction, but does not cause a reactor scram before 1.8 seconds have expired.

In this case, OP-3000 would apply and feedwater would be terminated by operator action.

4) Condensate pumps do not auto trip on reactor trip at any power level.

   On a normal reactor trip (>15 MWe) the condensate pump recirculation valve will open to supply flow through the air ejector condenser to maintain condenser vacuum.

   The same signal that auto trips the boiler feed pumps causes the condensate pump recirculation valve to open. Therefore, if the BFPs do not receive a trip signal, the condensate pump recirculation valve will probably also not open.

5) Manual feedwater control is used at power levels <15 MWe.

   For power levels below which an auto BFP trip does not occur (<15 MWe), the feedwater control valves can be assumed to remain in position until operator takes action.

6) Heater drain pumps are tripped as a result of low heater drain tank level.

7) Auxiliary feedwater is manually initiated.

IV.2 Review of Containment Scoping Analysis

For a main steam line rupture inside containment, the most important parameters determining peak containment pressure are total mass and energy of the blowdown fluid. The scoping analysis performed assumed a total blowdown of 42,705 pounds of steam. The breakdown of this 42,705 pounds is provided in Table I. The assumption made regarding feedwater flow was as follows:

1) 3 BFPs plus 3 condensate pumps operating,

2) Feedwater delivered only to the ruptured SG,

3) Feedwater control valve was assumed to be wide open,

4) Trip signal occurs on high containment pressure of 5.0 psig at 10.0 seconds after rupture,

5) Reactor trips at 11.0 seconds,

6) All feedwater terminated at 10.0 seconds after trip (including condensate pumps)

A feedwater flow rate of 350 $lbm\text{-}sec^{-1}$ is the average flow rate which would exist for the period 0-21 seconds based on the ruptured steam generator depressurization rate.

The scoping analysis combined the maximum initial steam generator inventory (zero power conditions) with the limiting feedwater operational mode. Therefore, the only concern is whether the loss of feedwater 10 seconds after trip is appropriate. This analysis is significantly conservative. If feedwater flow is not terminated within 10 seconds, operating experience and OP-3000 actions indicate that the results of the containment pressure response analysis would remain valid by not allowing more than 42,705 lbm of steam to be released into the containment atmosphere.

TABLE I

Main Steam Line Rupture Inside Containment

(Total Blowdown Sources, Pounds of Steam)

Total Zero Power Liquid Inventory	31,500
Total Zero Power Vapor Inventory	760
Feedwater that enters SG following forced feedwater flow termination due to elevation differences in feedwater piping height to feedwater nozzle location	860
Feedwater that enters SG following feedwater termination due to flashing of feedwater when the SG pressure decreases below the feedwater saturation pressure	1,060
21 seconds of feedwater flow at 350 $lbm\text{-}sec^{-1}$	7,350
Mass of steam between SG nozzle and NRV	425
Reverse flow through NRV prior to closure	790
Total fluid available to enter containment	42,745
Minus vapor remaining in SG	-40
Resulting total fluid entering containment	42,705

IV.3 Review of Steam Line Rupture Analysis Performed to Determine Return-to-Power Potential

For the steam line break transient, the potential for returning to power exists when the reactor coolant system is rapidly cooled by secondary blowdown, thereby producing a positive reactivity insertion through the various reactivity feedback parameters. This effect is further enhanced by the amount of secondary inventory available for blowdown and continued feedwater operation following the break or reactor trip. The details of the differences between breaks inside and outside the containment were discussed in Section II. From a return-to-power aspect, the steam line break outside containment is the worse case.

As discussed in Section III.2, the safety analysis of the Yankee Rowe reactor for steam line break was performed for Core XI and rechecked for each subsequent refueling.

In the course of the review of these analyses for the response to Reference (1), the feedwater assumption used in the Core XI analysis was found suspect. That assumption is the following:

"The feedwater flow at the start of the transient corresponds to the steady state value. It is assumed to decrease to zero when the secondary pressure reaches the saturation pressure corresponding to the temperature of the feedwater."

The validity of this assumption was tested in Section IV.1, where the operation of the feedwater system was thoroughly discussed. The review of this system revealed that continued feedwater operation outside the limit of the assumption previously stated was possible. In particular, items 3) and 4) in Section IV.1 indicate that for a steam line break outside containment, at power, the feedwater system may continue to feed the steam generators until manually tripped. This, of course, can lead to an excessive cooldown transient and potentially a return-to-power.

The assessment of this potential is discussed in Section V.2.

IV.4 Ability to Detect and Isolate a Damaged Steam Generator and Affects on Feedwater Pumps

Yankee has reviewed the capability to detect and isolate a damaged steam generator from various energy sources and reviewed the capability of pumps to remain operable after extended operation at runout flow.

The following instrumentation can be used to identify the affected steam generators:

a) steam generator pressure (3 channels/generator),

b) steam generator level,

c) feedwater flow,

d) steam flow, and

e) cold leg temperature.

Section IV.1 provides a description on the operation of the feedwater system. Above 15 MWe, the boiler feed pumps trip automatically on a reactor scram, and the heater drain pumps trip on low level in the heater drain tank. Presently, the condensate pumps are not tripped; however, a modification will be installed during the next refueling outage which will trip the condensate pumps on a coincident containment isolation signal and low steam line pressure signal. This will isolate the affected steam generator from forced flow of the feedwater/condensate system. Section VI discusses this design modification.

In addition, forced flow can be terminated by remotely closing the feedwater regulating valves or the motor operated feedwater isolation valves.

Because of the long steam generator dryout times (approximately 1 hour) at Yankee Rowe, automatic initiation of the emergency feedwater system is not required. The operators would leave the affected steam generator isolated by closed valves as the system is put into operation.

A key advantage to a manually initiated emergency feedwater system is that the chance of a pump being subjected to runout flow conditions and possible cavitation affects is significantly reduced. Rowe has a positive displacement pump which is essentially unaffected by reduced discharge pressure.

IV.5 Potential Problem Areas Identified

Based on the review provided in the preceding sections the following three items which impact the consequences of a main steam line rupture were determined to be potential problem areas:

1) Boiler feed pumps and condensate pumps do not trip at plant power levels less than 15 MWe.

This situation potentially impacts both the core reactivity transient and the containment response resulting from a main steam line rupture.

2) Boiler feed pumps may not trip at any power level for a main steam line rupture outside containment.

   This situation has the potential for worsening the reactivity transient resulting from a main steam line rupture.

3) Condensate pumps do not trip on reactor trip at any power level.

   This situation has the potential to impact both the core reactivity transient and the containment response following a main steam line rupture.

These .hree potential problem areas can be restated relative to the concerns pecified in Reference (1), containment pressure response and potential for core return-to-power, as follows:

1) Containment Pressure Response Potential problem areas

   a) boiler feed pumps do not trip at power levels less than 15 MWe, and

   b) condensate feed pumps do not trip at any power level.

2) Potential for Core Return-to-Power problem areas

   a) boiler feed pumps do not trip at power levels less than 15 MWe,

b) boiler feed pumps may not trip at any power level for a severe main steam line rupture outside containment, and

c) condensate feed pumps do not trip at any power level.

Section V discusses the actual impact of each of these potential problem areas.

V. IMPACT OF REVIEW FINDINGS

The review findings discussed in Section IV.5 point out potential problem areas with both the containment pressure response and the core return-to-power response analyses currently supporting Yankee Rowe operation. The purpose of this section is to determine the actual impact of these review findings to determine if either plant changes and/or operating procedural changes are warranted.

Section V.1 discusses the impact of the review findings on the containment scoping analysis. Section V.2 discusses the impact of the review findings on the return-to-power analysis currently supporting plant operation.

V.1 Impact of Review Findings on Containment Scoping Analysis

The review finding specified in Section IV.5 identified two potential problem areas. These potential problem areas are the following:

1) Boiler feed pumps do not auto trip at power levels less than 15 MWe, and

2) Condensate feed pumps do not auto trip at any power level.

These two items are concerns because the scoping containment analysis assumed complete termination of all feedwater flow within 10.0 seconds after reactor trip.

The situation can be divided into two categories. These two categories are the following:

1) Operation less than 15 MWe,

2) Operation greater than 15 MWe.

For operation at less than 15 MWe, feedwater flow is low, ranging from that required to remove main coolant pump heat (10 GPM per steam generator) to 100 GPM per steam generator. The scoping analysis assumed full feedwater system operation combined with the maximum possible initial steam generator inventory. Based on operating experience and the minimal feedwater flow at power levels less than 15 MWe, adequate time is available for the operator to trip the single boiler feed pump operating at these conditions and close the feed line valve and air operated valve to terminate flow to a ruptured steam generator to ensure the applicability of the scoping analysis.

For operation at power levels greater than 15 Mwe, boiler feed pump auto trip will occur. However, presently the condensate pumps will continue to run. In actuality, the operators have adequate time to respond to ensure the conservatism of the scoping analysis. This is due to the significant conservatism built into the scoping analysis. The scoping analysis combined the limiting plant operating conditions from zero power to full power. This approach inherently adds significant conservatism. For example, the initial steam generator mass was assumed to be 31,500 pounds, whereas at full power, actual inventory is approximately 20,000 pounds. However, it is prudent to terminate condensate pumped flow as soon as possible to minimize the potential containment response. Therefore a design change will be made to ensure a condensate pump trip. The trip will be an automatic coincidence trip on high containment pressure and low steam line pressure in one of the main steam lines. This modification will ensure that the containment scoping analysis remains significantly conservative and bounding. Section VI provides the details of this design modification.

Additionally, operating procedures will be modified to ensure that on every reactor trip both the feed line valves and the air operated feed valves are closed immediately. Currently, either of the valves (there are four sets, two for each steam generator) is required to be immediately closed. The procedural modification will add additional assurance of feedwater system isolation to a damaged steam generator.

V.2 Impact of Review Findings on Potential for Return-to-Power

In Section IV.3, the potential impact of continued feedwater flow on a steam line break was discussed. Since there was a question as to validity of the assumption used in the Core XI analysis, it was necessary to evaluate the present cycle for a steam line break outside containment with continued feedwater flow in consideration. The results of this assessment and impact on the potential for a return-to-power are now discussed.

Using the Core XI assumption for feedwater operation, it was determined that the secondary inventory was more than adequate to cool the primary reactor coolant system to 212°F or atmospheric saturated conditions. This calculation assumed that only the primary coolant was allowed to lose its stored energy and that any additional heat sources to the primary, such as reactor decay heat or stored energy in the metal mass, are neglected, thereby maximizing the RCS cooldown. Also, safety injection water (cold) was homogeneously mixed with the RCS to further cool the RCS. The impact from continued feedwater operation is to cool the RCS even further by feeding the steam generators with relatively cold feedwater. Since there is a limit as to how cold the feedwater could get, for consistency, 70°F was assumed to be the lowest temperature the RCS could obtain. The problem then becomes whether or not the reactor returns critical.

A detailed reactivity calculation for the current cycle, Cycle XIV, under the most conservative reactivity conditions was performed. This analysis essentially shows that for both the full power and zero power cases, the reactivity level in the reactor remains subcritical for RCS cooldown to 70°F. This calculation included moderator temperature defect, fuel temperature Doppler defect, the most reactive control rod stuck out, boron insertion with safety injection, and conservative uncertainties on all the physics data.

In addition to this calculation, it was determined that even without boron, the reactor would remain subcritical down to 70°F when the uncertainties were not included. With reactivity uncertainties, boron was required. However, the amount of boron required to maintain a subcritical condition was quite small. In fact, the time required to inject enough boron with the safety injection system to maintain subcriticality is substantially shorter than the time required to cool the reactor coolant system to 70°F, thereby precluding the return-to-power.

However, to lessen the severity of a main steam line rupture outside containment, the Agastat timer will be replaced with a new timer which will have a sufficient delay to insure that the required pump trip occurs. This replacement will be completed before the plant goes back on line.

VI. SUMMARY OF DESIGN AND PROCEDURAL MODIFICATIONS

Section V briefly describes some design and procedural modifications that will take place as a result of our findings.

VI.1 Design Modifications

Section IV.1 identifies the potential for a loss of automatic tripping of the boiler feed pumps when power is >15 MWe. To resolve this concern, Yankee plans to replace the existing Agastat timer with a new timer which will have a sufficient delay to insure that the required pump trip occurs. This replacement will be completed before the plant goes back on line.

Section IV.1 also identifies a potential problem in that the condensate pumps are not tripped at any power level. This may present a problem with respect to V.C. overpressurization. To resolve this concern, a modification will be installed during the next refueling outage which will trip the condensate pumps on a coincident containment isolation signal (5 psig V.C. pressure) and low steam line pressure in one main steam line. This will terminate forced flow of the feedwater/condensate system to the affected steam generator for breaks inside containment.

VI.2 Procedural Modifications

Emergency operating procedure OP-3000, "Emergency Shutdown from Power", will be modified to require the operator to immediately close both the steam generator feed line valve and the air operated feed valve in each of the four feedwater lines. Currently, emergency operating procedure OP-3000 requires only that either of these two valves per steam generator be immediately closed on every reactor trip. This procedural modification will add additional assurance of feedwater system isolation to a damaged steam generator. This modification will take place before the plant goes back on line.

VII. CONCLUSIONS

The preceding sections of this report have addressed each of the review items requested in Reference (1). Although two of the concerns specified in Reference (1) apply to Yankee Rowe, at this time the overall conservatism of the Yankee Rowe design to main steam line ruptures has not been lessened. However, to add additional conservatism to the overall design and lessen the impact of a main steam line rupture, two minor system design changes and a procedural change will be implemented at Yankee Rowe. These changes as discussed in Section VI will lessen the impact of a postulated main steam line rupture either inside or outside containment. At this time, it is not clear that these changes are required to ensure acceptable consequences for these events. However, Yankee feels these modifications are prudent and hence, will implement these modifications.

Once again, we point out that a second level review of the supporting analyses is in progress and will be completed shortly. If we discover any information that alters our conclusions, we will inform you of our findings. However, Yankee feels the information provided is conservative.

VIII. REFERENCES

1) USNRC/IE letter to Yankee Atomic Electric Company, February 8, 1980, "IE Bulletin 80-04, Analysis of a PWR Main Steam Line Break with Continued Feedwater Addition".

2) Proposed Change No. 96, Yankee Atomic Electric Company to US AEC, August 6, 1971.

3) Letter R. H. Groce to USNRC, November 27, 1978, WYR 78-103, "Systematic Evaluation Program (SEP)".

4) Letter D. E. Vandenburgh to USNRC, October 9, 1979, WYR 79-145, "Potential Unreviewed Safety Question Regarding Safety Function Interations with Non-Safety Grade Systems".

5) YAEC letter to USNRC dated March 29, 1974, Proposed Change No. 115.