Maine Yankee

RELIABLE ELECTRICITY FOR MAINE SINCE 1972

EDISON DRIVE • AUGUSTA, MAINE 04336 • (207) 622-4868

October 3, 1990
MN-90-99 SEN-90-278

UNITED STATES NUCLEAR REGULATORY COMMISSION
Attention: Document Control Desk
Washington, DC 20055

References:
(a) License No. DPR-36 (Docket No. 50-309)
(b) MYAPCo Letter to USNRC dated September 22, 1983 (MN-83-199)
(c) USNRC Letter to MYAPCo dated February 14, 1984
(d) MYAPCo Letter to USNRC dated June 15, 1984 (MN-84-109)
(e) USNRC Letter to MYAPCo dated May 23, 1984 - Amendment No. 76 - Maine Yankee Feedwater Trip System Technical Specifications

Subject: Startup Feedwater System - Temporary Waiver of Compliance Request

Gentlemen:

This letter is to request a temporary waiver of compliance to Maine Yankee Technical Specification 3.22, "Feedwater Trip System". This waiver is necessary to permit operation of Maine Yankee's emergency feedwater system in the "startup mode" beginning with the plant startup scheduled for October 3, 1990, and until such time as a clarifying change to the Technical Specifications can be made.

In the startup mode, the emergency feedwater pumps are cross-connected with the main feedwater system. Feedwater flow is directed to the first stage feedwater preheaters, through the main feedwater regulating bypass valves and to each of the three steam generators. Feedwater preheating while in the startup mode is desirable to minimize the thermal stresses imposed on the feedwater ring and piping. While in the startup alignment, the bypass valve in each line would automatically isolate feedwater flow in the event of a low pressure condition (e.g. steam line break) in the associated steam generator. Prior to exceeding two percent reactor power, feedwater flow is realigned to feed directly from the main and emergency feedwater pumps to the generators. In this alignment two means are available to isolate feedwater flow to an affected steam generator.

0.239

SEN90278.LTR

A001
1/1

A waiver of compliance is necessary as it was recently determined that Maine Yankee's existing Technical Specifications were not adequately updated to address operability of the feedwater trip system during startup operations. During Maine Yankee's development of a modified feedwater trip system design, the single failure vulnerability of the system was specifically evaluated and it was concluded that no safety concern existed. However, the Technical Specifications were not revised to specifically address feedwater trip system requirements during startups and shutdowns. A copy of Maine Yankee's evaluation is enclosed. This evaluation concludes that continued feeding of a generator undergoing a steam line break while feeding from the emergency feedwater system, through the first point heaters will not result in a return to criticality or a containment overpressure condition.

It is apparent by review of background documents that we contemplated the single failure vulnerability of the feedwater trip system during startup (and shutdown) evolutions. It also appears that in preparing Technical Specifications changes for the feedwater trip system, we did not believe that any changes were necessary. It is not clear whether this matter had been previously discussed with the NRC staff. Correspondence relating to the feedwater trip system design is enclosed for your convenience. Reference (b) presented a description of the system's design, and mentions alignment during plant startups and shutdowns. Reference (c) approved the feedwater trip system design with open items resolved by Reference (d). The NRC approved the current Technical Specification with Reference (e).

We trust this information is acceptable. We appreciate your prompt attention to this matter.

Very truly yours,

SENichols

S. E. Nichols
Licensing Section Head

SEN:WSD

Enclosures

c: Mr. Thomas T. Martin
Mr. Charles S. Marschall
Mr. E. H. Trottier

SEN90278.LTR

R. Turcotte

MEMORANDUM

YANKEE ATOMIC - FRAMINGHAM

To P. L. Anderson

Date January 23, 1984
Group # YAG-84-16
W.O. # 8100

From K. R. Rousseau/P. J. Guimond

Subject AUXILIARY FEEDWATER PUMP FLOW DURING A SYSTEM LINE BREAK EVENT

I.M.S. #

[illegible]
ENCLOSURE F
REFERENCE (3)
2 pages

REFERENCES

1. Service Request M84-10, "AFW Pump Flow During a SLB Event", dated January 12, 1984
2. MYC-478, "AFW Pump Flow During an SLB Event", dated January 18, 1984
3. MYC-433, "Maine Yankee Cycle 8 MSLB Analysis", dated January 4, 1984
4. SAG 76-190, RESAC.M.IV.C.3, Memo, P. A. Bergeron to A. E. Ladieu, "Maine Yankee Steam Line Break Results", dated August 19, 1976

DISCUSSION

Reference (1) requested NSD to evaluate a single active failure of a main feedwater regulating bypass valve in the open position during a steam line rupture event occurring during plant startup/shutdown when the auxiliary feedwater pumps are aligned through the first-point heaters. Such a failure could potentially result in the auxiliary feedwater pumps continuing to feed the faulted steam generator. The major concerns for the evaluation were to assure that continued auxiliary feedwater pump flow would not create a return to power or containment overpressure event. Reference (1) also requested an estimate of the time to dry out the intact (isolated) steam generators, assuming all auxiliary feedwater is being pumped to the depressurized steam generator.

RECEIVED
JAN 27 1984
MAINE YANKEE PROJECT

The Reference (2) documents our evaluation of these concerns. Results are discussed below.

Return to power is not a concern for this transient. This event has previously been addressed in Reference (3). The results of that calculation concluded that the cooldown from SLB with continuous auxiliary feedwater addition to the depressurized steam generator is bounded by the cooldown from an SLB with a failed main feedwater regulating valve. No return to power will occur in either case.

Containment overpressure for this event will not exceed the design pressure. Reference (4) documented several containment pressure sensitivities performed by Combustion Engineering. Evaluation of the sensitivities indicates that the peak pressures are proportional to the total amount of feedwater delivered to the faulted steam generator. A main feedwater regulating valve failed in the open position will deliver more feedwater to the depressurized SG (31% full rated flow[3]) than a failed open bypass valve with 1330 gpm auxiliary feedwater (18% full rated flow). CE has analyzed the former case which results in peak pressures on the order of 45 psig. Thus, this transient is not a concern from a containment overpressure standpoint.

The time to dry out the two intact steam generators has conservatively been estimated (i.e., no credit taken for any heat removal via the AFW flow to faulted SG) to be a minimum of approximately 1.2 hours[2] for a transient initiated from 2% power. This should allow ample time for the operator to manually realign AFW flow to the intact steam generators.

SAFETY EVALUATION

This memo is safety-related. It confirms that the concerns raised in Service Request M84-10, resulting from the potential for continuous feedwater to the depressurized steam generator following a SLR with the single failure of a MFWRV bypass valve to close when AFW pumps are aligned in the startup mode, will not result in either a return to power or containment pressurization above the design pressure.

Kenneth R. Rousseau

Kenneth R. Rousseau, Engineer
Transient Analysis Group

Philip J. Guimond

Philip J. Guimond, Senior Engineer
Transient Analysis Group

P. A. Bergeron

Paul A. Bergeron, Manager
Transient Analysis Group

KRR/ds
cc:
M. P. LeFrancois
G. E. Jarka
M. W. Scott
J. T. McCumber
R. P. Sanson
R. T. Turcotte
R. P. Shone

Ref.(s)
Page 2/2

7.7 Description of Operations

With a low water level in any steam generator, both P-25A and P-25C pumps will automatically start. Auxiliary feedwater flow is manually controlled from the Control Room using the AFW flow control valves. The air required for modulation of the control valves is supplied from the normal plant instrument air system. (As always, Air Accumulator TK-111 is only used for isolation air.) The new isolation valves are maintained normally open and do not actuate on a low level signal. AFW flow to each steam generator is indicated on the MCB.

With a low steam generator pressure event (SLB), the Feedwater Train Trip signal (generated at 400 psig SG pressure) will close the AFW control valve and new isolation valve in line to the depressurized steam generator(s). The feedwater train trip signal is annunciated at window F9-7W, Auto Feedwater Valve/Train Trip. The valves will remain closed as long as SG pressure is less than 400 psig. The air required to isolate the AFW valves comes from their respective train accumulators. An assumed single active failure of one valve leaves the second valve available to perform the isolation function.

Depending on the actual location of the postulated SLB, low pressure could be experienced in all steam generators. This would cause momentary closure of the AFW valves to all generators until a pressure greater than 400 psig is restored. At this point, the AFW valves will automatically re-open. No manual reset is required to get the valves open. An active failure of any single component in the isolation scheme, (a relay, power supply, pressure transmitter, solenoid, or control/isolation valve), occurring after all the valves are closed, but before 400 psig (or greater) is restored to two intact SGs, will result in a minimum of one AFW flow path available to feed an intact SG.

The AFW pumps will not be tripped by the low pressure event and will continue to feed the pressurized steam generators if the pumps are already running or when a low water level signal is sensed. AFW feedwater flow to each steam generator is indicated on the MCB along with SG level and pressure indications.

During normal operation between zero and 2% power, the AFW pumps may crossfeed through the first point heaters using the main feedwater regulating valve bypass valves (FW-A-112, 212, and 312) for flow control. The bypass valves will also close on a Feedwater Train Trip signal. With the removal of the AFW automatic pump trip, an assumed single failure of a bypass valve to close to the affected steam generator, will result in the AFW pumps continuing to feed that generator. It has been confirmed by Reference (s) that continued feeding of a broken steam generator with extreme AFW pump runout flow (2 pumps operating), between zero and 2% power, <u>will not</u> create a containment overpressure or a return to power due to RCS cooldown. Therefore, the AFW automatic isolation scheme need not be redundant when operating between zero and 2% power and the single isolation valve scheme when crossfeeding is acceptable. Furthermore, operators will have a minimum of 1.2 hours to the intact steam generators before dry-out occurs. This is more than adequate time to stop the AFW pumps, align the direct flow path to the intact SGs and reinitiate AFW flow.

The existing key-locked switches on the MCB used to block the low steam generator pressure trips during startups and shutdowns are unaffected by this change. Additionally, closure of the main feedwater regulation valve and bypass valve to the affected SG on a feedwater train trip signal and tripping of the main feedwater pumps, condensate pumps and heater drain pumps on this signal coincident with a SIAS is unaffected by this change.

7.8 <u>Seismic Supports</u>

The new valves and piping modifications are classified Seismic Category 1, consistent with the AFW System. Cygna Energy Services has performed the seismic, deadweight, and thermal piping analysis associated with this change. New support design drawings are enclosed with this design package. Support Bill of Materials are indicated on the drawings.

The scope of the support work to be done is as follows:

1. Anchors H-371, H-372, and H-379 will be removed and three new anchors (H-101, 102, and 103) will be installed just downstream (AFW Pump Room).

2. Two new lateral (N-S, E-W) (box) supports (H-104 and H-105) will be installed on two of the risers from the 6" diameter header (AFW Pump Room).

3. Lateral Supports H-371, H-372, and H-379 will be removed (AFW Pump Room).

4. Rod Hanger H-382 RH will require replacement of the rod with a plate (AFW Pump Room).

5. Rod Hanger PS-1 (RH) will be removed (lower PAB penetration area).

No rebar shall be cut during the support installation without first obtaining approval from the Mechanical Engineering Group.

12.3 The plant is responsible for revising the plant operating procedures to reflect this design change.

12.4 NSD will prepare a recommended instruction for the valve and accumulator test with appropriate acceptance criteria. This will be forwarded by ECN.

13.0 FSAR PAGES AFFECTED

FSAR Section 10.2.3, Page 10-5 and Figure 10.2-3 are affected by this EDCR and are included in Enclosure (E) of this design package.

14.0 LICENSING

This EDCR affects the Maine Yankee Technical Specifications by:

1. Deleting the automatic AFW pump trip and 5-minute time delayed restart on a Feedwater Train Trip signal.

2. Adding a redundant isolation valve in line to each steam generator that will close on a Feedwater Train Trip signal (low pressure in its associated steam generator).

The required word changes associated with the AFW modifications have been incorporated into the proposed Technical Specifications for Cycle 8 operation. The specific pages affected by this change are 2.1-3, 3.22-1, 4.6-4, 4.6-5, and 4.6-6. These proposed changes have been submitted to the USNRC in Reference (v).

15.0 SAFETY EVALUATION

15.1 Safety Classification

The Auxiliary Feedwater System is classified as Safety Class 3 upstream of the AFW control valves and Safety Class 2 downstream of the control valves. Modifications to piping and supports will meet the appropriate safety class requirements. The new control switches and cables have been purchased as being Class 1E. Control power and logic to the new isolation valves is also classified 1E.

15.2 Summary

The new AFW automatic isolation scheme enhances the reliability of the Auxiliary Feedwater System in response to a postulated steam line break by closing the associated AFW valves instead of tripping the pumps. As a result, the AFW pumps will always be available for feeding the intact steam generators upon receipt of a low steam generator level signal.

The feedwater train trip signal will initiate closure of the redundant valves in line to the affected steam generator. Therefore, operator intervention to manually isolate the affected steam generator during power operation (assuming a valve fails to close) to direct feed flow to the intact generators will not be required.

If flow pressure is sensed in all three SGs causing closure of all AFW automatic valves, a single active failure preventing a valve to reopen after pressure is restored to two SGs, allows as a minimum, one path available for feeding an intact SG without relying on operator action. This is acceptable since one SG is capable of removing post-shutdown decay heat.

At zero to 2% power, crossfeeding through the main feedwater bypass valves (single valve per steam generator) is acceptable, since assumed failure of the valve to close, resulting in continued run-out flow of both AFW pumps to the affected steam generator, will not cause a return to power incident or containment overpressurization. Additionally, operators will have approximately 1.2 hours to re-establish flow to the intact steam generators before dry-out occurred.

The seismically installed air system will provide a reliable source of isolation air to the new valves should the normal instrument air system fail. The new valves are seismically qualified to operate during a seismic event and the new piping arrangement will be analyzed for seismic, deadweight, and thermal stress to ensure the pressure boundary integrity of the system.

This EDCR has been addressed in the Maine Yankee Steam Line Break analysis (Calculation No. 4 and 5, Section 11.0 of this EDCR package) and is considered acceptable.

This EDCR does not impair the safety to the general public in that it does not:

1. increase the probability of occurrence of a previously evaluated accident;

2. create the possibility of an unreviewed accident; or

3. reduce the margin of safety as defined in the Technical Specifications.

15.3 Conclusion

The proposed modification has been analyzed to assure that it does not create any unreviewed safety questions as defined in 10CFR50.59(a)(2).

1/MY-NRC
2/MY 25.1
II.E.1.1

MAINE YANKEE ATOMIC POWER COMPANY

EDISON DRIVE
AUGUSTA, MAINE 04336
(207) 623-3521

September 22, 1983
MN-83-199

GDW-83-71

* Hebert
YNSD

RESPONSIBILITY Prouty

RESPOND BY NONE

NRC DUE DATE NONE

Director of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. James R. Miller, Chief
Operating Reactors Branch No. 3
Division of Licensing

References:
(a) License No. DPR-36 (Docket No. 50-309)
(b) USNRC Letter to MYAPCo, dated March 24, 1983
(c) USNRC Meeting with MYAPCo on June 28, 1983, Re: Auxiliary Feedwater System Evaluation
(d) MYAPCo Telephone Conference with USNRC on July 13, 1983
(e) MYAPCo Letter to USNRC, dated July 21, 1983, MN-83-154
(f) MYAPCo Letter to USNRC, dated July 14, 1983, MN-83-147, Re: Proposed Change No. 97

Subject: Maine Yankee Auxiliary Feedwater System - TMI Item II.E.1.1

Dear Sir:

In accordance with Reference (e), this letter provides a brief description of conceptual design modifications planned for our AFW System (Enclosure). We plan on installing these design changes during the next scheduled refueling. In order to maintain this schedule, we request that you provide us any comments that you may have by October 28, 1983.

Also provided in Enclosure 2 to this letter is our formal response to your AFW evaluation submitted to us in Reference (b).

We trust this information is satisfactory; however, should you have any questions, please contact us.

Very truly yours,

MAINE YANKEE ATOMIC POWER COMPANY

G D Whittier

G. D. Whittier
Licensing Section Head

GDW/bjp
Enclosures (6 Pages)

cc: Dr. Thomas E. Murley
Mr. Cornelius F. Holden

1783L-GDW U.S. Mail 9/23/83 ~~8309280167~~ (7P)

ENCLOSURE NO. 1

CONCEPTUAL DESIGN MODIFICATIONS FOR AFW SYSTEM SIMPLIFICATION

Present System Description (Refer to attached sketch)

The present AFW System consists of two (2) motor-driven pumps, which also serve as the emergency feed pumps, and one steam-driven pump all connected in parallel. Each of the pumps is capable of supplying the necessary feedwater flow during abnormal transient conditions.

Each of the emergency feed pumps is powered from a separate emergency bus. The pumps are normally aligned to take suction from the demineralized water storage tank (DWST); however, the primary water storage tank (PWST) can also be used.

During power operation, the emergency feed pumps are aligned to take suction from the DWST and discharge to a common header which supplies three separate discharge lines, one to each steam generator. Each discharge line is equipped with a single air-operated flow control valve to regulate feedwater flow. The valves are normally opened and controlled from either the MCB or the Steam Generator Emergency Panel. The valves are also equipped with trip solenoid valves which will close the control valve on receipt of a Feedwater Train Trip signal (low steam generator pressure).

During plant startups and shutdowns an alternate discharge flow path is normally used which directs auxiliary feed flow through feedwater heaters. This allows preheating of the feedwater and also permits automatic flow control using the main feedwater bypass valves.

Present System Operation Description

During power operation the emergency feed pumps P25-A and P25-C are aligned to discharge directly into the steam generators as discussed above. Should an abnormal condition occur causing low steam generator water level (35% in any generator) both pumps will automatically start to supply emergency feedwater makeup.

Should a steam generator depressurization event occur (steam line break incident) a Feedwater Train Trip signal is generated on low pressure of approximately 400 psig. This signal will automatically trip both emergency feed pumps and automatically close the faulted steam generator associated AFW control valve. The pumps remain secured for a period of five (5) minutes by means of a timer. After the five (5) minute sequence, both pumps will automatically start if a low level signal still exists and provide feed flow to the intact steam generators.

Together, the pump trip (with delayed restart) and control valve closure provide the redundant means for isolating AFW flow to a faulted steam generator. The steam driven feed pump is maintained in functional readiness for operation in the unlikely event of a total loss of AC power. The steam driven pump P25-B is not required to perform any plant design basis safety function. Its use is strictly limited to a station blackout event or Appendix R fire considerations.

Modified System Description

Several modifications are planned for the AFW System. These include installing an additional air-operated isolation valve in each discharge line to the steam generators and removal of the automatic pump trip and time delayed restart.

The new isolation valves will be installed in series with the existing AFW control valves. Each valve will be equipped with a single nuclear qualified solenoid valve (similar to those which exist on the present control valves), which will also receive the Feedwater Train Trip signal.

The new valves will be normally open and will only be used for isolation purposes. The valves will be spring-to-open, air-to-close and will receive supply air from the existing seismic air accumulator. The accumulator will maintain sufficient air pressure to the actuator diaphragms for closing the valves when normal plant instrument air is not available. The AFW automatic pump trip and timer sequence on Feedwater Train Trip signal (SG low pressure) will be removed. This will no longer be necessary since a redundant isolation scheme of the faulted steam generator will be provided with the additional valve. Pump auto-start on low steam generator level will not change however, the pumps will not trip on low pressure.

Modified System Operation Description

As before, with a low water level in any steam generator, both P25-A and P25-C pumps will automatically start. Feedwater flow is manually controlled with the AFW flow control valves.

With a low steam generator pressure event, the Feedwater Train Trip signal will close the flow control valve and the new isolation valve to the depressurized steam generator(s). An assumed single active failure of one valve leaves the second valve available to perform the isolation function. The pumps will not be tripped by the low pressure event and will continue to feed the pressurized steam generators.

Additional Feedwater Modifications

Further conceptual modifications for the feedwater system include the installation of a new condenser surge tank. This tank will take the place of the existing DWST during power operation, supplying and receiving condensate water to maintain appropriate condenser levels. This action will allow keeping the present DWST isolated from the condenser and maintaining the tank contents (100,000 gallons minimum) available for emergency feedwater use during power operation when the emergency feed alignment is required. The new tank, and connecting piping will be classified as non-nuclear safety class, consistent with existing secondary plant equipment.

ENCLOSURE NO. 2

MAINE YANKEE RESPONSE TO NRC
AFW SYSTEM EVALUATION LETTER, MARCH 24, 1983
(as modified by References c and d)

1. Additional Short-Term Recommendation 1

Summary of NRC Positions

1a) The DWST must have redundant level indication and redundant alarms. Two redundant channels each having level indication and an alarm would be acceptable.

1b) Provide a Technical Specification which requires that 100,000 gallons of water be stored in the DWST.

Response

1a) The Annunciator System at Maine Yankee is not Class 1E; therefore, any level alarms on the DWST will not be Class 1E.

The existing tank level indicator on the MCB provides the "primary" means for level monitoring of the DWST. The low-level alarm provides backup to the monitoring of this indicator. The sensor for the low-level alarm is independent of the sensor for the indication channel. The level indication channel is powered separately from the Annunciator System. In addition, a separate review of instrumentation requirements is being conducted as part of the Regulatory Guide 1.97 study, scheduled to be completed by the end of 1983.

1b) Proposed Change No. 97 (Reference f) has been submitted for NRC review and approval. This proposed change provides a Technical Specification which requires 100,000 gallons of water be stored in the DWST.

2. Short-Term Recommendation GS-5 and Long-Term Recommendation GL-3

Summary of NRC Positions

2a) The steam driven AFW pump should automatically start and initiate feed flow in the event of loss of all ac. Accordingly, the fail-safe function of the CIS valve in the steam admission line to the turbine driven pump should be reviewed.

2b) Provide assurance that the steam generator levels at 30 minutes into the station blackout includes the steam generator blowdown line if this line remains open after loss of ac.

Response

2a) The steam-driven AFW pump is not required to mitigate any design basis accident at Maine Yankee. Accordingly, the steam supply valves, air system, and controls associated with the pump need not be safety class, seismic, nor redundant.

Use of the steam-driven AFW pump is required following a station blackout (a nondesign basis event). Presently, the steam driven feed pump can be quickly initiated following a loss of ac event, by manual action of a single control switch located in the control room. Proposed Appendix R modifications will provide further enhancement to the manual control capability of this pump. Included in the modifications will be supply air to appropriate valves from the EFCV air accumulator, manual valve operators where feasible, and new controls for the valves to provide an additional manual remote control station.

A review of the CIS valve (MS-T-163) located in the steam admission line to the steam-driven AFW pump indicates that the valve is normally opened and is designed for air-to-open, spring-to-close. The actuator is equipped with a single 3-way, dc solenoid valve which is normally de-energized. This valve is subject to Appendix R modifications and, therefore, will receive its control air from the EFCV accumulator (reliable source). Once this modification is complete, this valve will not fail closed following a loss of ac event.

2b) Within approximately 30 minutes following a station blackout, the operator must initiate feedflow with the steam driven pump. This can easily be accomplished from the control room as indicated above. The steam generator levels at the 30 minute mark include the normal steam generator blowdown rate.

It is Maine Yankee's position that with the remote-manual control scheme described, 30 minutes provides more than sufficient time for operators to manually initiate the steam-drive AFW pump following a complete loss of ac. Automatic start of this pump adds complexity without providing significant benefit (if any) to the safety of the plant.

3. <u>Long-Term Recommendation GL-2</u>

<u>Summary of NRC Positions</u>

3a) Provide a modification to guard against a mechanistic failure of or inadvertent closure of the single manual valve located in the single suction line to the auxiliary feedwater pumps.

<u>Response</u>

3a)1. The internals of valve AFW-1 will be removed during the next refueling outage.

4. <u>Long-Term Recommendation 4 (Part b)</u>

<u>Summary of NRC Position</u>

4a) With respect to the potential for an AFW line break, assure that such a break can be isolated in time to prevent steam generator dryout.

Response

4a) The failure of passive emergency systems, i.e., tanks, pipes, structures, is not part of Maine Yankee's design basis.

Never-the-less, the auxiliary feed system arrangement is such that should the common suction line or storage tank fail, alternative suction line and supply is available from the PWST. Should the common discharge line fail an alternative flow path is available via the crossconnect to the normal feed system. In either case, the alternative paths or supply can be made available by operting two manual valves.

Since the overall usage factor of the system is low and it is not subject to harsh service or environment, pipe breaks in the system need not be postulated.

5. Basis for AFW Flow Requirements

Summary of NRC Position

5a) The licensee must provide more assurance that the operator has sufficient time to close a failed AFW control valve to direct flow to the intact generators before dryout occurs (following a SLB).

Response

5a) It must be recognized that the added valves and complexity pose an additional incremental risk that flow may be impeded. First the postulated failed valve must be in the line to the steam generator with a failed steam line, reducing the improbable. Second, assuming no flow to the viable steam generators, the operator has 30 minutes to act, more than enough time, to close a manual valve adjacent to the failed open flow control valve. None-the-less to satisfy staff concerns, as described in Enclosure 1, the AFW modifications will include an additional automatic isolation valve (air-operated) in series with each control valve to provide redundant valve isolation capability to each steam generator.

6. Short-Term Recommendation 1

Summary of NRC Position

6a) Provide a Technical Specification change to require monthly AFW System operability testing in lieu of testing on a quarterly basis.

Response

6a)2. Maine yankee will submit a proposed change to Technical Specifications which will require emergency feed System operability testing every month rather than every three months. It should be noted that the proposed change submittal will be made with the understanding that emergency feed pump testing at minimum recirculation flow and then separate cycle testing of key valves will be acceptable to the NRC.

7. Limiting Condition for Operation (LCO) for Steam-Driven AFW Pump

Summary of NRC Position

7a) Provide a Technical Specification (which resembles the current CE Standard Technical Specification, Revision 2) to include a limiting condition for operation "LCO" for the steam-driven auxiliary feed pump.

Response

7a) At Maine Yankee, the steam driven feed pump is not required to provide emergency feed for design basis events. Redundant emergency feed capability is provided by the two 100% capacity motor driven pumps, each powered from a separate emergency power bus. Therefore, application of Standard Technical Specifications to Maine Yankee is inappropriate.

The steam driven feed pump would be required to function only in the event of a complete station blackout or an Appendix R worst case fire, both of which are of extremely low probability and not plant design basis.

Consistent with the pump's intended function, yet comparable to the limiting condition for the electric-driven AFW pumps, a more realistic allowable outage time (LCO) for the steam driven pump would be seven (7) days.

The specification should include a remedial action statement that would permit continual operation after the seven-day period, provided a continuous fire watch is posted to ensure no fire damage would occur that could result in a total loss of station power. These remedial actions would remain until the steam driven pump is again capable of performing its function or a suitable substitute pump(s) is installed.

11 my - NRC
2 my 5.5

Excerpt

MAINE YANKEE ATOMIC POWER COMPANY

EDISON DRIVE
AUGUSTA, MAINE 04336
(207) 623-3521

June 15, 1984
MN-84-109

GDW-84-143

RESPONSIBILITY ______

RESPOND BY ______

NRC DUE DATE ______

Director of Nuclear Reactor Regulation
United States Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Mr. James R. Miller, Chief
Operating Reactors Branch No. 3
Division of Licensing

References: (a) License No. DPR-36 (Docket No. 50-309)
(b) USNRC Letter to MYAPCo dated March 13, 1980 - I.E. Bulletin No. 80-06
(c) MYAPCo Letter to USNRC dated August 30, 1983 (MN-83-178)
(d) MYAPCo Letter to USNRC dated September 12, 1983 (MN-83-175)
(e) MYAPCo Letter to USNRC dated September 22, 1983 (MN-83-199)
(f) USNRC Letter to MYAPCo dated February 14, 1984

Subject: Maine Yankee - Feedwater Trip System

Gentlemen:

The staff has completed their post-implementation review and issued a Safety Evaluation, Reference (e), for the Maine Yankee feedwater trip system. One item, regarding the system's conformance to the single failure criterion, however, remains open. So that the staff may complete their review of the feedwater trip system, written confirmation in response to the NRC open item is provided in the enclosure to this letter.

Very truly yours,

MAINE YANKEE ATOMIC POWER COMPANY

G D Whittier

G. D. Whittier
Licensing Section Head

GDW/bjp

Enclosure: (2 Pages)

cc: Dr. Thomas E. Murley
Mr. Cornelius F. Holden

~~8406190354~~ (3p)

3062L-SEN

ENCLOSURE

NRC Open Item, Part (1)

Written confirmation that the changes made to correct the deficiencies described in Maine Yankee Reportable Occurrence No. 83-028/01T-1 do not violate the Maine Yankee physical separation criterion, as defined in Sections 8.3.7.5 and 8.3.7.7 of your FSAR. Schematics should be provided for the final system to confirm that the system satisfies the single failure criterion.

Maine Yankee Confirmation

The plant modification implemented to correct the design deficiencies described in Reference (c) were accomplished in accordance with the Maine Yankee physical separation criteria. Schematics of the final system design are attached for your review, (ESK-7G and 7GA - EDCR 80-38, ECN #3).

NRC Open Item, Part (2)

Written confirmation that the changes related to deletion of the auxiliary feedwater pump trip and addition of auxiliary feedwater isolation valve closure do not violate the Maine Yankee single failure criterion. This confirmation should include schematics of the final system design.

Maine Yankee Confirmation

As described in Reference (e), Maine Yankee removed the auxiliary feedwater pump trip circuitry and restored its function by installing auxiliary feedwater isolation valves in each discharge line to the steam generators. The isolation valves combined with the auxiliary feedwater control valves provide independent and redundant means for isolating auxiliary feedwater. Removal of the auxiliary pump trip circuitry and installation of feedwater isolation valves does not alter the system's conformance to the single failure cirterion. The attached schematics (ESK-5AR, 5AX, 7GA, and 11AG - EDCR 83-29) support this position.

NRC Open Item, Part (3)

Written confirmation that the proposed additional annunciation for manual switches, which can inhibit main steam line, main feedwater, and auxiliary feedwater isolation, has been implemented. This confirmation should include final schematics reflecting these changes.

Maine Yankee Confirmation

As committed in Reference (d), all manual switches which could be left in positions to inhibit main steam line, main feedwater and auxiliary feedwater isolation shall be annunciated in the main control room when they are in a position to inhibit isolation. Installation of the additional required annunciation was completed during the cycle 7/8 refueling outage. The attached schematics (ESK-3A, 7G, 7GA, 10AV and 10H - EDCR 84-16) reflect this installation.

ENCLOSURE (Continued)

NRC Open Item, Part (4)

Written confirmation that the Feedwater Trip System will be modified to meet the requirements of IE Bulletin 80-06, "Engineered Safety Feature (ESF) Reset Controls", or provide an evaluation to justify not modifying the system.

Maine Yankee Confirmation

The Feedwater Train Trip System has been reviewed with regard to IE Bulletin No. 80-06. As designed, the system need not comply with the requirements of 80-06, as discussed below.

A decrease in steam generator pressure to less than approximately 400 psig due to a steam line break (SLB), will cause the main feedwater regulating valve and bypass valve feeding the depressurized generator to close. A coincident steam generator low pressure (SGLP) and safety injection actuation signal (SIAS) causes the pumps in the main feedwater train to trip resulting in termination of main feedwater flow to all steam generators. Tripping of the main feedwater train together with closure of the associated feed regulating and bypass valves, provide redundant feed flow isolation to the depressurized steam generator(s). Therefore, automatic restoration of main feedwater flow to the depressurized steam generator should not occur and blocking of the signal for reopening of the associated feed valves is not necessary.

If pressure in any steam generator is above 400 psig or returns to 400 psig following excess flow check valve closure, the associated feedwater regulating and bypass valves will continue to be operated by the normal plant feedwater control logic. Assuming the main feedwater train does not trip on the coincident SGLP and SIAS (possible single active failure), flow to these steam generators would be through the bypass valves. During a plant trip, the turbine trip logic overrides the steam generator water level control system (SGWLCS) and causes the feedwater regulating valves to close and the bypass valves to open to a preset stroke for decay heat removal. This continued feed flow through the bypass valves to the pressurized steam generators does not adversely affect the steam line break analysis and is, therefore, acceptable. Assuming conservative conditions (with AFW flow included), the Control Room operators would have approximately 15 to 24 minutes to take manual control of the bypass valves and prevent eventual overfilling of the associated steam generators. We feel this is more than adequate time for operator response. Furthermore, the SLB procedure requires Control Room operators to check that the main feed train has tripped and the associated feed valves have closed following receipt of the initiating signals. Therefore, plant operators will be well aware of the system operating status.

Failure of the turbine trip override logic to the main feed and bypass valves would result in the SGWLCS continuing to control the main feedwater regulating valves and the bypass valves would remain closed. This control system, which is powered from two independent sources, would automatically close the feedwater regulating valves on increasing water level in the associated steam generators.

MAY 2 9 1984

1/NRC-MY
2/MY 1.6.4
RC#104

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

MAY 2 3 1984

Docket No. 50-309

* Whittier
YNSD

RESPONSIBILITY Brinkler

RESPOND BY None

NRC DUE DATE None

Mr. J. B. Randazza
Executive Vice President
Maine Yankee Atomic Power Company
83 Edison Drive
Augusta, Maine 04336

Dear Mr. Randazza:

The Commission has issued the enclosed Amendment No. 76 to Facility Operating License No. DPR-36 for the Maine Yankee Atomic Power Station. This amendment consists of changes to the Technical Specifications (TS) in partial response to your application dated January 27, 1984 as supplemented by your letter dated May 1, 1984.

This amendment reflects changes made to the auxiliary feedwater isolation system.

A copy of the Safety Evaluation is also enclosed. The notice of issuance will be included in the Commission's next regular monthly Federal Register notice.

Sincerely,

Kenneth L. Heitner

Kenneth L. Heitner, Project Manager
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 76 to DPR-36
2. Safety Evaluation

cc w/enclosures:
See next page

~~8406070068~~ (2p)

Maine Yankee Atomic Power Company

cc:
Charles E. Monty, President
Maine Yankee Atomic Power Company
Edison Drive
Augusta, Maine 04336

Mr. Charles B. Brinkman
Manager - Washington Nuclear Operations
Combustion Engineering, Inc.
7910 Woodmont Avenue
Bethesda, Maryland 20014

John A. Ritsher, Esquire
Ropes & Gray
225 Franklin Street
Boston, Massachusetts 02110

State Planning Officer
Executive Department
189 State Street
Augusta, Maine 04330

Mr. E. C. Wood, Plant Manager
Maine Yankee Atomic Power Company
P. O. Box 3270
Wiscasset, Maine 04578

Regional Administrator
U.S. Nuclear Regulatory Commission
Region I
631 Park Avenue
King of Prussia, Pennsylvania 19406

First Selectman of Wiscasset
Municipal Building
U.S. Route 1
Wiscasset, Maine 04578

Mr. Cornelius F. Holden
Resident Inspector
c/o U.S. Nuclear Regulatory Commission
P. O. Box E
Wiscasset, Maine 04578

Mr. P. L. Anderson
Project Manager
Yankee Atomic Electric Company
1671 Worcester Road
Framingham, Massachusetts 01701

U.S. Environmental Protection Agency
Region I Office
ATTN: Reg. Radiation Representative
JFK Federal Building
Boston, Massachusetts 02203

Mr. Rufus E. Brown
Deputy Attorney General
State of Maine
Augusta, Maine 04330

David Santee Miller, Esquire
Perkins Road
Boothbay Harbor, Maine 04538

Mr. John H. Garrity, Senior Director
Nuclear Engineering and Licensing
Maine Yankee Atomic Power Company
83 Edison Drive
Augusta, Maine 04336

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

MAINE YANKEE ATOMIC POWER COMPANY

DOCKET NO. 50-309

MAINE YANKEE ATOMIC POWER STATION

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 76
License No. DPR-36

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Maine Yankee Atomic Power Company, (the licensee) dated January 27, 1984 complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

~~8406070073~~ (7/)

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.B(6)(b) of Facility Operating License No. DPR-36 is hereby amended to read as follows:

   (b) <u>Technical Specifications</u>

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 76 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

James R. Miller

James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: May 23, 1984

ATTACHMENT TO LICENSE AMENDMENT NO. 76

FACILITY OPERATING LICENSE NO. DPR-36

DOCKET NO. 50-309

Revise Appendix A as follows:

Remove Pages	Insert Pages
3.22-1	3.22-1
3.22-2	3.22-2
4.6-4	4.6-4
4.6-6	4.6-6

3.22 FEEDWATER TRIP SYSTEM

Applicability:

Applies to the operating status of the feedwater trip system.

Objective:

To specify conditions of the feedwater trip system necessary to ensure steam generator cooldown potential remains acceptable in the event of a main steam line break.

Specification:

The feedwater trip system shall be operable to perform the following functions whenever the reactor coolant boron concentration is less than that required for hot shutdown.

1. Automatic shutdown of all main feedwater, condensate and heater drain pumps which are operating or set for automatic start.

2. Automatic closure of all main feedwater regulating valves and main feedwater regulating bypass valves which are open or set to open automatically in the line to the low pressure SG(s).

3. Automatic closure of the auxiliary feedwater control and isolation valves in the AFW lines which are open or set to open automatically to the low pressure Steam Generator(s).

   Exception: Specifications 1 and 2 do not apply when the main feedwater lines are isolated from the steam generators.

   Remedial Action: If the feedwater trip system is found to be inoperable, it must be restored to an operable status within the next two hours, or else the reactor must be shut down within the next six hours and the reactor coolant system borated to hot shutdown concentration within an additional six hours.

Basis:

The feedwater trip system limits the cooldown of the reactor coolant system in the event of a main steam line break by limiting the flow of cold feedwater into the steam generators. Limiting the reactor coolant system cooldown limits reactivity insertion associated with a negative reactivity temperature coefficient during a cooldown.

The feedwater trip system is actuated by signals generated by safety related circuitry associated with the reactor protective system and safety injection system. This safety related circuitry is not itself part of the feedwater trip system. The system provides signals to the controls of feedwater system pumps (main feedwater pumps, condensate pumps, and heater drain pumps), and to the controls of the feedwater regulating valves, feedwater regulating valve bypass valves, and auxiliary feedwater control and isolation valves.

The system valves are aligned to provide flow to each steam generator following system actuation upon low steam generator water level signal from any one of the three steam generators. However, for a steam generator depressurization event, such as a steam line break, receipt of a low steam generator pressure signal initiates closure of the control and isolation valve(s) feeding the depressurized steam generator(s). This limits excessive reactor coolant system cooldown and the resultant reactivity insertion produced by excessive feedwater flow to a depressurized steam generator. Flow will continue to steam generators remaining pressurized. Flow to a depressurized steam generator will be reestablished by reopening the control and isolation valves after repressurization e.g., by isolation from the steam line break.

Operability of the system assures that the reactivity attributable to reactor coolant system cooldown due to feedwater addition to steam generators after a main steam line break is within the limits established in the steam line break safety analysis.

If the feedwater trip system is discovered to be inoperable, the best course of action is to restore its operability promptly, thus avoiding challenges to plant systems that result from perturbing steady state operation. A two-hour time period presents low risk of a main steam line break yet allows enough time for deliberate restoration of system operability through maintenance actions.

If operability cannot be restored the reactor must be shut down. Six hours provides ample time for an orderly controlled shutdown. If operability cannot be restored by that time, the reactor coolant system must be borated to hot shutdown concentration within an additional six hours. Twelve hours permits an orderly shutdown while assuring that the risk of a main steam line break during the period is very low.

The intended function of the feedwater trip system can be accomplished under conditions of partial system inoperability provided all feedwater system pumps and valves tripped by the system which are operating can be tripped by the operable portions of the trip system. Pumps which cannot be tripped by the trip system due to partial trip system inoperability can be shut down to assure functional capability.

When the reactor coolant system is at hot shutdown boron concentration, the steam line break cooldown cannot cause sufficient reactivity insertion to cause a return to critical, so the feed trip system is not required to function.

Monthly inspections shall be performed to verify that all manual valves in the AFW system necessary to assure flow from the primary water source to the steam generators are locked in the proper position.

During normal plant operation, each auxiliary feed pump shall be tested at quarterly intervals to demonstrate operability of pumps, system valves and instrumentation.

C. MAIN STEAM EXCESS FLOW CHECK VALVES

The main steam excess flow check valves shall be tested once every 6 weeks for movement of the valve disc through a distance of approximately one and one-half inches. These valves will be tested through full travel distance during each refueling interval.

D. FEEDWATER TRIP SYSTEM

1. The following tests will be performed at each refueling interval:

   a. Main Feedwater Pumps

      Each main feedwater pump, condensate pump, and heater drain pump trip system shall be tested by tripping the actuation circuitry with a safety injection signal coincident with steam generator low pressure signal.

   b. Feedwater Valves

      Each main feedwater regulating valve, main feedwater regulating bypass valve, and auxiliary feedwater control and isolation valve trip system shall be tested by tripping the valves with a low pressure signal from their respective steam generators.

Basis:

The safety injection system and the containment spray system are principal plant safeguards systems that are normally operable during reactor operation.

Complete system tests cannot be performed when the reactor is operating because of their inter-relation with operating systems. The method of assuring operability of these systems is a combination of complete system tests performed during refueling shutdowns and monthly tests of active system components (pumps and valves) which can be performed during reactor operation. The test interval is based on the judgment that more frequent testing would not significantly increase the reliability (i.e., the probability that the component would operate when required), yet more frequent tests would result in increased wear over a long period of time.

Prior to plant startup following an extended cold shutdown, a flow test is performed on the Auxiliary Feedwater System to functionally verify the system alignment from the demineralized water storage tank to the steam generators.

Monthly inspections are performed to verify that all manual valves in the Auxiliary Feedwater System from the primary water source to the steam generators are locked in the proper position.

Proper functioning of the steam turbine admission valve and starting of the auxiliary feed pump will demonstrate the operability of the steam driven pump. Verification of correct operation will be made both from instrumentation in the Main Control Room and direct visual observation of the pumps.

The main steam, excess flow check valves serve to limit an excessive reactor coolant system cooldown rate and resultant reactivity insertion following a main steam break incident. Their freedom to move will be verified periodically.

The feedwater trip system acts to limit excessive reactor coolant system cooldown and the resultant reactivity insertion produced by excessive feedwater flow to the steam generators in the event of a main steam line break. The system acts to trip feedwater pumps, condensate pumps, and heater drain pumps, and close the main feedwater regulating valve, feedwater regulating valve bypass valve, and emergency feedwater control and isolation valves to the affected steam generator. Signals activating the system are developed by instrumentation, logic, and relaying associated with the safety injection actuation system and the excess flow check valve actuation system. The circuitry which develops these signals is subject to surveillance requirements of Tables 4.1-1 and 4.1-2 which assure their reliability.

The main feedwater pumps, condensate pumps, and heater drain pumps trip upon coincidence of SIAS and a low steam generator pressure. The valves close on the low steam generator pressure in the associated steam generator. The reliability of the coincidence logic is assured by testing in accordance with #20 of Table 4.1-2.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 76 TO LICENSE NO. DPR-36

MAINE YANKEE ATOMIC POWER COMPANY

MAINE YANKEE ATOMIC POWER STATION

DOCKET NO. 50-309

Introduction

By letter dated January 27, 1984 the Maine Yankee Atomic Power Company requested certain changes to the technical specifications (TS) for the Maine Yankee Atomic Power Station. These changes included TS changes to support a modification to the AFW trip system. The proposed changes to the AFW trip system are described in a letter dated September 22, 1983. The proposed modification concerns the manner in which AFW flow is isolated from a faulted steam generator. In the current design, depressurization of a steam generator produces a trip signal which closed the AFW flow control valve to that steam generator and tripped both electrically driven AFW pumps. The pumps remained tripped for five minutes before flow was restored to the intact steam generators. During this interval, the intact steam generators were not fed by the AFW system.

The modification proposed by Maine Yankee involves adding a second isolation valve to each AFW line to the steam generators. This second isolation valve will provide the necessary redundant action to assure isolation of a faulted steam generator. The trip of the AFW pumps will be removed and they will be allowed to feed the intact steam generators.

Maine Yankee has proposed revisions to sections 3.22 and 4.6 of its TS to reflect these plant modifications. Maine Yankee has also proposed a revision to Section 4.6 of the TS changing the testing interval of the AFW pumps to monthly in accordance with the commitment made by letter dated September 22, 1983. However, by letter dated May 1, 1984, Maine Yankee withdrew the portion of the application concerning the revised pump interval.

2.0 Evaluation

By letter dated February 14, 1984, the staff transmitted to Maine Yankee its evaluation of the entire Feedwater Trip System. In this evaluation, the staff concluded that the design of the feedwater trip system, including the proposed modification, was acceptable subject to certain confirmatory items. Our acceptance of the licensee's proposed changes to the plant technical specifications to reflect this modification is based on the conclusions of the above referenced evaluation.

3.0 Technical Specification Changes

3.1 Section 3.22 Feedwater Trip System

This specification has been rewritten to reflect the proposed modification. Reference to the AFW pump trip has been deleted. The specification has been clarified to indicate that the main feedwater regulating and regulating bypass valves and AFW control and isolation valves to the faulted (low pressure) steam generator are tripped. These changes reflect the approved modification to the system and are therefore acceptable.

3.2 Section 4.6 - Periodic Testing

Section 4.6.D on feedwater valves has been modified to delete reference to the AFW pump trip. Reference has been added to the new AFW isolation valves. These changes reflect the approved modification to the system and are therefore acceptable.

4.0 Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

5.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: May 23, 1984

Principal Contributors:
K. L. Heitner
T. Dunning, ICSB
N. Wagner, ASB

FEB 16 1984

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

FEB 14 1984

Docket No. 50-309

RESPONSIBILITY

RESPOND BY 5/7/84

NRC DUE DATE 6/15/84

Mr. John H. Garrity, Senior Director
Nuclear Engineering and Licensing
Maine Yankee Atomic Power Company
83 Edison Drive
Augusta, Maine 04336

Dear Mr. Garrity:

SUBJECT: MAINE YANKEE - FEEDWATER TRIP SYSTEM

We have reviewed your responses dated September 12 and 22 and November 2, 1983 concerning the feedwater trip system at Maine Yankee. This letter and the enclosed Safety Evaluation (SE) document our post-implementation review of your response to IE Bulletin 80-04, "Analysis of a PWR Main Steam Line Break with Continued Feedwater Addition". Your response provided for the implementation of a feedwater trip system.

Based on our review of your submittals and the results of discussions held with your staff, we conclude that the feedwater trip system design satisfies the single failure requirements consistent with the original design requirements for Maine Yankee. Our conclusion is subject to the satisfactory completion of the following confirmatory items:

(1) Written confirmation that the changes made to correct the deficiencies described in Maine Yankee Reportable Occurrence #83-028/01T-1 do not violate the Maine Yankee physical separation criterion as defined in Sections 8.3.7.5 and 8.3.7.7 of your FSAR. Schematics should be provided for the final system to confirm that the system satisfies the single failure criterion.

(2) Written confirmation that the changes related to deletion of the auxiliary feedwater pump trip and addition of auxiliary feedwater isolation valve closure do not violate the Maine Yankee single failure criterion. This confirmation should include schematics of the final system design.

(3) Written confirmation that the proposed additional annunciation for manual switches which can inhibit main steam line, main feedwater and auxiliary feedwater isolation has been implemented. This confirmation should include final schematics reflecting these changes.

(4) Written confirmation that the feedwater trip system will be modified to meet the requirements of IE Bulletin 80-06, "Engineered Safety Feature (ESF) Reset Controls," or provide an evaluation to justify not modifying the system.

~~8403020283~~ (10pp)

We request that you submit these confirmatory items within 120 days of your receipt of this letter. If you have any questions, please contact the NRC Project Manager, Kenneth L. Heitner.

The information requested in this letter affects fewer than 10 respondents; therefore OMB clearance is not required under P.L. 96-511.

Sincerely,

James R. Miller, Chief
Operating Reactors Branch #3
Division of Licensing

Enclosure:
Safety Evaluation

cc: See next page

Maine Yankee Atomic Power Company

cc: Charles E. Monty, President
Maine Yankee Atomic Power Company
Edison Drive
Augusta, Maine 04336

Mr. Charles B. Brinkman
Manager - Washington Nuclear Operations
Combustion Engineering, Inc.
7910 Woodmont Avenue
Bethesda, Maryland 20814

John A. Ritsher, Esq.
Ropes & Gray
225 Franklin Street
Boston, Massachusetts 02110

State Planning Officer
Executive Department
189 State Street
Augusta, Maine 04330

Mr. E. C. Wood, Plant Manager
Maine Yankee Atomic Power Company
P. O. Box 3270
Wiscasset, Maine 04578

Regional Administrator
U. S. Nuclear Regulatory Commission
Region I
631 Park Avenue
King of Prussia, Pennsylvania 19406

First Selectman of Wiscasset
Municipal Building
U. S. Route 1
Wiscasset, Maine 04578

Mr. Cornelius F. Holden
Resident Inspector
c/o U. S. Nuclear Regulatory
Commission
P. O. Box E
Wiscasset, Maine 04578

Mr. P. L. Anderson
Project Manager
Yankee Atomic Electric Company
1671 Worcester Road
Framingham, Massachusetts 01701

U. S. Environmental Protection Agency
Region I Office
ATTN: Reg. Radiation Representative
JFK Federal Building
Boston, Massachusetts 02203

Mr. Rufus E. Brown
Deputy Attorney General
State of Maine
Augusta, Maine 04330

David Santee Miller, Esq.
Perkins Road
Boothbay Harbor, ME 04538

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

SAFETY EVALUATION

FEEDWATER TRIP SYSTEM

MAINE YANKEE

DOCKET NO. 50-309

I. INTRODUCTION

IE Bulletin 80-04, "Analysis of a PWR Main Steam Line Break with Continued Feedwater Addition," was issued to identify deficiencies in safety analyses of main steam line breaks, in part, as a result of incorrect analysis assumptions identified by Maine Yankee. The concerns centered on the consequences of continued main feedwater flow and subsequent initiation of the auxiliary feedwater system and their impact on both containment pressure and return to criticality caused by rapid cooldown of the reactor coolant system (RCS).

In response to IE Bulletin 80-04, the licensee, in a letter dated May 7, 1980, committed to implement changes in the Maine Yankee protection system using signals from the existing main steam line break protection circuits to effect safety grade termination of main feedwater by closure of the main feedwater regulating valves and their bypass valves and in combination with a safety injection signal to trip the main feedwater motor-driven, condensate and heater drain pumps. Also on low steam generator pressure, the auxiliary feedwater control valves would be closed and the pumps tripped for a period of five minutes. The staff's SE for this issue, dated July 10, 1981, conditioned its conclusions on conformance to IE Bulletin 80-04 on a subsequent post-implementation review of the instrumentation and control details of the system modifications. This report provides the staff's evaluation of those modifications described in the licensee's informal submittal of schematics and piping diagrams provided for staff review.

II. EVALUATION

Prior to implementation of the feedwater trip system, the plant design included protection circuits which terminated main steam flow to all steam generators on low pressure sensed in any steam generator. The protection circuits consisted of four redundant low pressure trip channels and one 2-out-of-4 logic circuit for each steam generator. In order to satisfy the single failure criterion, the logic for each steam generator was supplied power from a bus which was independent of that used for logic of the other steam generators. Each logic provided a closure signal to two separate solenoid valves for each main steam line isolation valve.

To provide for safety grade termination of feedwater under steam line break conditions, the licensee utilized the original low pressure trip channels (four per steam generator) for inputs into an additional 2-out-of-4 logic circuit for each steam generator. The new logic circuit for each steam generator is supplied power from a bus which is independent of that used for the existing logic circuit. Each 2-out-of-4 logic (one original and one new) provides a separate closure signal to a separate solenoid valve such that on low steam generator pressure either logic will close that steam generator's feedwater regulating and regulating bypass valves and auxiliary feedwater control valve.

A new logic is provided for each auxiliary feedwater pump which trips the pump on low steam generator pressure and allows the pump to be restarted either manually or automatically after a five minute time delay.

This new logic in coincidence with a safety injection signal provides a trip signal to the main feedwater, condensate, and heater drain pumps.

As a result of the staff's concerns on the design, the licensee conducted a detailed review of the system and found several instances where the single failure criterion was not met. These deficiencies, described in Maine Yankee Reportable Occurrence #83-028/01T-1, were a result of the use of three independent power sources where the logic and valve actuation circuits or different logic circuits were supplied power such that the loss of a single power source violated the single failure criterion. By letter dated November 2, 1983, the licensee described the modifications to correct these deficiencies. The staff has reviewed the proposed modifications and is concerned that these changes may compromise the independence of redundant systems due to lack of adequate physical separation of the circuits. Since the plant separation criterion is on a power source basis, the staff's concern arose from the licensee's implying that power sources to selected circuits would be merely switched with no regard for separation and cable routing. The staff therefore requires that the licensee provide written confirmation that the modifications, as proposed, will not violate the physical separation criterion for redundant circuits and that schematics for the final system design be provided to confirm that the system satisfies the single failure criterion.

Enclosure 1 to the licensee's letter dated September 22, 1983 states that tripping of the auxiliary feedwater pumps on low steam generator pressure will be deleted and replaced by closure of additional auxiliary feedwater isolation valves. Since the original design relied on both the tripping of the auxiliary feedwater pumps and the closure of the auxiliary feedwater control valves to meet the single failure criterion, the staff requires that the licensee provide written confirmation that the deletion of the pump trip and addition of isolation valves will not violate the single failure criterion which is the design basis for this system. Additionally, the licensee should provide schematics for the final system design.

Enclosure 1 to the licensee's letter dated September 12, 1983 states that a design will be implemented to provide annunciation of incorrect positioning of any manual switch which can be left in a position which can inhibit main steam line, main feedwater and auxiliary feedwater isolation if annunciation is not currently provided. The staff requires written confirmation that these changes have been implemented and submittal of the final schematics reflecting those changes.

During the staff's review of this feedwater trip system, it was noted that components, such as the main feedwater regulating and bypass valves and the auxiliary feedwater control valves, reopen automatically following the removal of their closure signal due to the self-resetting (no seal-in) of the system logic circuitry. This system, as designed, does not therefore comply with the requirements of IE Bulletin 80-06, "Engineered Safety Feature (ESF) Reset Controls". With regard to the isolation of the auxiliary feedwater system, this self-reset feature does assure that flow will be restored to intact steam generators if isolation was initiated by a line break or single failure initially resulting in low pressure in multiple steam generators. However, with regard to isolation of the main feedwater regulating and bypass valves, it does not appear to be appropriate to permit these valves to reopen automatically. While it is recognized that this may not pose a problem if a safety injection occurs and the main feedwater pumps are subsequently tripped and do not automatically restart, the staff questions whether this aspect of the design has been adequately evaluated. Therefore, the staff requests confirmation that the system will be modified to satisfy the requirements of IE Bulletin 80-06 or that an evaluation will be provided which demonstrates that the consequences related to high energy line break analyses and automatic restoration of main feedwater flow are acceptable.

III. CONCLUSION

Based on the staff's review of the Maine Yankee feedwater trip system, implemented in response to IE Bulletin 80-04, the staff concludes that the design satisfies the single failure requirements consistent with the original plant design basis subject to satisfactory completion of the confirmatory items noted herein.

Principal Contributor:

F. Burrows, DSI