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Figure 7.1

JCO APPROVAL COVER SHEET

JCO# 2-91-1

Implementation Date 10/21/91 *
Expiration Date End of Cycle 11

Subject Main Steam Line Break Inside Containment

Unit 2

Requested By (NUPOC Director) J.S. Keenan

Date 10/18/91

Concurrence: [signature] 10/21/91
Manager, Responsible Organization — Date

Concurrence: [signature] 10/21/91
Manager, Responsible Organization — Date

Concurrence: [signature] 10/21/91
Manager, Responsible Organization — Date

Concurrence: [signature] 10-21-91
Manager, Generation Facilities Licensing — Date

Concurrence: _____ _____
PORC Meeting # — Date

Approval: _____ _____
NUPOC Director — Date

*JCO is implemented upon final approval of the NUPOC Director.

Lower Portion To Be Used For Closure Of JCO

Concurrence: _____ _____
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Millstone Unit No. 2

Justification for Continued Operation #2-91-1

Main Steam Line Break Inside Containment

TABLE OF CONTENTS

INTRODUCTION

This document provides the justification for Millstone Unit No. 2 to continue to operate following the identification that the containment structure is outside of its design basis for certain postulated main steam line breaks (MSLBs) inside of containment. This hypothetical condition is based upon strict licensing requirements which include, but are not limited to, hypothesizing a guillotine MSLB from 100% power with the feedwater control valve failing in the open position and no operator intervention for the first 10 minutes of the transient. This condition was analyzed and resulted in a calculated peak containment pressure and temperature in excess of 90 psig and 400°F respectively. This is in contrast to the Millstone Unit No. 2 technical specification Section 5.2.2 which states that the reactor containment is designed and shall be maintained for a maximum internal pressure of 54 psig and a temperature of 289°F.

PLANT CONDITION DISCOVERED

As a result of the Steam Generator Replacement Project, new MSLB analyses were being performed to assess the impact of the new steam generators on the containment response. In performing these analyses, it was discovered that the conditions assumed for the current design basis containment response for a MSLB are not limiting. In particular, it was identified that a full power case with a smaller break size and off-site power available will result in a higher peak containment pressure and temperature.

This condition was evaluated under REF 91-48 (MP2) and was determined to be reportable as a condition outside the design basis of the plant. REF 91-48 has been attached to this JCO (Attachment 1) and should be referenced for a detailed discussion of the background surrounding previous analysis of MSLBs inside containment.

REF 91-48 raises the possibility that the analyzed MSLB location and assumed single failure are nonconservative. Additional analyses have confirmed that the current design basis analysis assumptions with respect to power level, break size and single failure are not limiting. For the case of a 1.0 ft^2 MSLB at full power with a single failure of the feedwater control valve closure signal upon turbine trip, the peak containment pressure is predicted to exceed the containment design pressure of 54 psig. Further, a peak containment pressure in excess of 90 psig was predicted for a double-ended (6 ft^2) MSLB at full power with the single failure of the feedwater control valve to close.

Zero power and other single failures were evaluated and shown to give an acceptable containment response.

IDENTIFICATION OF POTENTIAL ADVERSE EFFECT ON SAFETY

In REF 91-48, the impact of the more limiting conditions for MSLBs were evaluated. In summary, it was found that a MSLB at hot full power was more limiting than zero power for containment response. In addition, it was identified that the limiting break size and the limiting single failure were

For the single failure in which the feedwater control valve was operable but a failure in the turbine trip closure signal was assumed, the limiting break size was 1.0 ft^2. For these conditions peak containment pressure was predicted to be 68 psig with a peak containment air temperature of 419°F. Since the design pressure is 54 psig, the predicted peak containment pressure exceeded the design pressure. While the air temperature exceeded the containment design temperature, it was concluded that because of condensation effects and thermal lag, the containment structure did not exceed the containment design basis temperature. Further, based upon engineering judgment that the short-term temperature peak will be offset by condensation and thermal lag, it was concluded that the equipment on the Electrical Environmental Qualification (EEQ) Master List would continue to be operable.

For the failure of the feedwater control valve to close on demand, the limiting break size is a double ended break. For this case, it was estimated that the peak containment pressure would be on the order of 70 to 80 psig. An additional parametric study (Attachment 2) performed by ASEA-Brown Boveri (ABB) shows that with credit for operator action at ten minutes to terminate feedwater, the predicted peak containment pressure is 93 psig and the peak containment air temperature was 427°F. In this case, the long-term temperature could also exceed the current qualification temperature of EEQ equipment and thermal lag could not be credited to offset the high temperature.

In spite of exceeding the containment design pressure for a design basis MSLB, it is expected that there will be no significant radioactivity releases. It is concluded that the potential off-site doses would be a small fraction of the limits provided in 10CFR100.

JUSTIFICATION FOR CONTINUED OPERATION

I. Procedural Changes and Operator Training

A dedicated licensed operator shall be stationed at CO-5 and will perform the following actions after a reactor trip:

a. OBSERVE a valid reactor trip.

b. CLOSE both main feedwater block valves, 2-FW-42A and 2-FW-42B.

Note - The following step is a contingency action to prevent containment overpressurization, although due to time limitations, no credit has been taken for this action in the main steam line break evaluation.

c. If either main feedwater block valve fails to close and the associated feedwater control valve fails to close, then SECURE all condensate pumps.

The dedicated operator shall have no concurrent duties with the main feedwater control valve isolation following a reactor trip.

The above operator actions are reasonable and can be performed in the time frame evaluated in the main steam line break analysis:

a. The dedicated operator will be stationed at the feedwater control station, and

b. The required operator actions are simple and require no diagnosis.

The above operator actions are consistent with the existing Standard Post Trip Action procedure (EOP 2525). The existing contingency for excessive feedwater flow is to manually close the main feedwater block valves.

Accomplishing this existing contingency action early does not effect any other preceding actions since these actions are completely independent of the main feedwater action step.

Training will be provided to each shift via a training guide and a briefing by the Shift Supervisor concerning the dedicated operator duties and actions.

II. <u>Analysis of Operator Action</u>

Immediate corrective action (as detailed above) involves positioning an operator at the main control board dedica closing the feedwater block valve (also titled the Feedwater Regul Valve Block Valve) upon a reactor trip. This will be incorporated into procedures and the dedicated operators will be provided training for this action. Based upon demonstrations on the Millstone Unit No. 2 specific simulator, it is concluded that initiation of feedwater block valve closure can be reliably accomplished in less than fifteen seconds. Since the stroke time for the block valve is less than 10 seconds, closure of the block valve within 25 seconds after a reactor trip can be credited as a back up to a failure of the closure of the feedwater control valve.

Parametric study #5 of Attachment 2 includes a case in which feedwater isolation was assumed within 25 seconds after a reactor trip. This case is a full power case with a break size of 6.0 ft^2. In addition, this case assumes that the initial containment pressure is 2.1 psig and 5% feedwater flow through the bypass valve is maintained throughout the accident. For this case, a peak containment pressure of 54 psig is predicted. Since this is the limiting case for the failure of the feedwater control valve, credit for this operator action as back up to the feedwater control valve failure will maintain containment pressure below the containment design pressure.

Credit for this operator action will also provide adequate mitigation for the failure of the turbine trip signal to close the feedwater control valve. For this single failure, the limiting condition is a small MSLB. Assuming a failure of the turbine trip signal to close the feedwater control valve, the Main Steam Isolation Signal will not be generated for 140 seconds, thus feedwater isolation (closure of the feedwater control valve) will not occur until 150 seconds. However, since a reactor trip is predicted to occur at 10 seconds, crediting operator action to close within 25 seconds would mean

that for this case feedwater would be isolated within 35 seconds of the MSLB event. This would result in the peak containment pressure remaining below the design pressure.

III. Additional Compensating Factors

A. Evaluation of Probability of Occurrence

To evaluate the likelihood of the event sequence, a probabilistic risk assessment was performed. The overall probability of a main steam line break inside containment with a failure of the feedwater control valve is conservatively estimated to be 10^{-5} per year. The short-term corrective measures (remote manual closure of the feedwater block valves by a dedicated operator) reduces the probability of this event (main steam line break concurrent with failure to isolate feedwater) to about 10^{-8} per year. Alternative corrective measures involving new automatic functions (as discussed in Section C) reduces this probability to about 10^{-7} per year.

It should be noted that the overall probability discussed above applies only to a scenario that could lead to exceeding the containment design pressure and does not consider the magnitude of the pressure response, nor does it include the likelihood of a subsequent loss of containment integrity. These types of events would be of an even smaller probability.

B. Feedwater Block Valve Capabilities

The operator action discussed above results in closure of both feedwater block valves (MOV 2-FW-42A/B). These valves are non-QA, are not powered from a vital power source, and are not currently in the Northeast Utilities (NU) GL 89-10 program. As a result of this concern, the NU GL 89-10 MOV Test Program ap, ich has been applied to these valves and the results indicate that an analytical approach backed by field testing provides sufficient confidence that the actuator/valve will operate successfully and can thus be credited to minimize the consequences of a MSLB. The analytical and testing efforts undertaken to address the capabilities of the MOVs (2-FW-42A and B) is described below.

1. MOV Fluid Conditions

The new MSLB analysis indicates that two MSL Break cases provide bounding fluid conditions for the MOV's (2-FW-42 A and B). Specifically; i) large MSLBs resulting in a feed pump trip prior to MOV actuation and ii) a smaller MSLB resulting in no feed pump trip. Fluid conditions resulting from these cases are discussed below.

The large MSLB case results in a feedwater pump trip prior to MOV-2-FW-42A and B closure, due to a main steam line isolation signal. Thus, MOV upstream conditions at valve closure will be 538 psig (max condensate pump pressure; Reference 1). MOV

was determined to be 277 psig. Thus, MOV delta-P = 538 psig - 277 psig = 261 psid for the large MSL break case.

The small break case does not result in a feed pump trip, however, steam generator backpressure is significantly higher. MOV upstream conditions at MOV closure considers both condensate pump and feedpump pressures; 538 psig and 1022 psig respectively (Reference 2). MOV downstream pressure at MOV closure is conservatively assumed to be a minimum of 465 psig (low SG pressure, MSI setpoint). Thus, MOV Delta-P = 538 psig + 1022 psig - 465 psig = 1095 psid (for the small MSL break case).

The worst case fluid conditions resulting from a MSLB for MOVs 2-FW-42 A/B is 1095 psid differential pressure.

2. MOV Thrust Calculations

Valve actuator thrust evaluations undertaken to address this JCO show that MOV's 2-FW-42A and B are capable of closure against worst case main steam line break fluid conditions (1095 psid as discussed above). The original design requirements were used to determine the required stem thrust. This will be documented in NUSCO QA calculation No. 89-078-073-EM Rev. 0. In addition, valve structural integrity was evaluated with respect to the thrust requirements. The analytical approach discussed above is consistent with the NU GL 89-10 MOV test program. It is noted that the original specification for 2-FW-42A and B required a valve closure at 1600 psid (Reference 3).

3. MOV Stroke Time

MOV's 2-FW-42A and B exhibit acceptable stroke speed characteristics to satisfy the 10-second stroke time required by this JCO. Specifically, i) the original specification (Reference 3) required a maximum stroke time of 10 seconds, ii) the motor operator vendor data (from Limitorque) determined stroke time to be 9.5 seconds based on stem travel speed and valve stroke and iii) Millstone Unit No. 2 plant start-up tests measured the stroke time to be 9 seconds.

4. MOV Field Testing

Prior to mode 1 operation, valve diagnostic testing (VOTES) in accordance with established procedures developed under the NU GL 89-10 MOV Test Program will be performed to confirm the analytical approach described above. Portions of this testing include direct measurement of actuator thrust via strain gages and stroke time measurements.

Analytical evaluations backed up by the field testing described above provides sufficient confidence that MOV's 2-FW-42A and B

will close in the event of feed control valve failure during the spectrum of main steam line breaks considered by this JCO.

C. Alternative Corrective Action

Modifications are contemplated during the current cycle that would preclude the need to credit operator action. These modifications may include the following:

1. Installation of a safety grade trip on the feedwater block valves 2-FW-42A and 2-FW-42B which will automatically close the valves on receipt of a high containment pressure signal. The isolation signal shall originate from the Engineered Safety Actuation System (ESAS). The proposed safety grade isolation signal for each motor operated valve shall not be common to the safety grade isolation signal for its in-series control valve.

2) Installing a safety grade trip to the feedwater pumps, condensate pumps and heater drains tank pump on receipt of a high containment pressure signal. This option may not be as desirable as Option 1.

3) For the small MSLB scenario where failure of the nonsafety grade turbine trip signal will not result in the automatic closure of the feedwater control valves, the addition of another safety grade trip may remove the need for operator action given the failure of the nonsafety grade turbine trip signal. Due to the length of time required to generate a main steam isolation signal for the small MSLB scenario, reliance is placed on a nonsafety grade closure signal. To remove the reliance on this nonsafety grade trip signal, it may be desirable to add a safety grade trip signal on receipt of a high containment pressure signal.

Longer-term, additional corrective actions are also being considered. These actions may include replacement of the existing feedwater block valves with QA Category 1E Safety Related valves and controls which will automatically close upon receipt of, most likely, a high containment pressure signal. Also, the valves must be properly sized to close against the expected differential pressures as well as fast acting to limit the amount of feedwater to the steam generator.

ADDITIONAL CONSIDERATIONS

The above demonstrates that the containment and associated equipment is operable. The actions specified ensure that during the most limiting MSLB that containment pressure does not exceed 54 psig. However, as an added measure of confidence for the protection of the public health and safety and thus the implementation of the JCO, the following analysis was conducted which provides reasonable assurance that even if containment pressure were to exceed its design basis of 54 psig, the containment and associated equipment would still be capable of performing their intended safety functions.

I. Containment Structural Evaluation for > 54 psig

The Millstone Unit No. 2 containment consists of a prestressed, reinforced concrete cylinder and dome connected to and supported by a massive reinforced concrete foundation slab. The containment was designed for an internal pressure of 54 psig, and was tested to 62 psig during the structural integrity test. The working stress design method was used to design the containment structure for various load cases, including the case of a design pressure of 54 psig. The containment structure was checked for factored loads and load combinations, including the case with a 1.5 load factor on the design pressure, which corresponds to 81 psig. The code requires that "strength be adequate to support the factored loads and that serviceability of the structure at the service load level be assured," (ACI-318-71 Commentary Section 9.1.1).

The ultimate capacity of containments has been studied and documented by many sources recently. In general, the anticipated ultimate capacity of a containment structure has been found to be 2 to 2.5 times design pressure. NUREG-1150 entitled "Severe Accident Risks: An Assessment for Five U.S. Nuclear Power Plants" studies the ultimate capacity of typical containment structures. Included in this study was Zion, which is a prestressed containment, with a design pressure of 47 psig. The evaluation determined that a lower bound on the ultimate capacity was around 100 psig (a factor of 2). These detailed studies have taken into account material strengths being higher than assumed, code allowables being conservative, as well as a detailed evaluation of structural behavior during beyond design basis events. A similar detailed study has not been performed for Millstone Unit No. 2, but the same factors which contribute to a lower bound ultimate capacity of 2 - 2.5 times design exist in the Millstone Unit No. 2 containment structure. This discussion on studies relative to ultimate capacity further substantiates that the containment can support the factored load case and beyond. These postulated load cases are beyond the design basis of the containment structure, but within the overall load carrying capability of the structure.

II. EEQ Evaluation for > 54 psig and > 289°F

The new postulated main steam line break includes an initial 70-second duration peak of 412°F at 54 psig. The corresponding saturated temperature during this superheated temperature peak is 302°F. The EEQ Master List equipment are determined to be qualifiable under these conditions by analysis which shows that equipment qualification tests are more severe than the temperature that the components will experience during the postulated steam line break.

The 412°F superheated steam will not heat equipment beyond their qualified temperature because of the short-term period at the superheated conditions. The temperature a component will reach is dependent upon the rate of heat transfer from the environment to the surface of the component. The rate of heat transfer will be dependent upon the saturated temperature of the environment for the short time that the conditions remain superheated. A condensate layer will form on all component surfaces during the initial phase of the accident. The temperature of this condensate layer will be less than or equal to the saturated

temperature, therefore the temperature of the component will remain less than the saturated temperature while this condensate layer exists.

The CAR fans are important for containment heat removal. The motors associated with these fans are enclosed, and have an internal cooling system. These motors were tested up to 80 psig with an associated saturated temperature of 324°F. With an internal cooling system they would not be directly exposed to a postulated higher temperature. Another item of particular interest for this event are the containment penetrations. These have been tested to 70 psig which corresponds to a 316°F saturation temperature.

A review of EEQ equipment in containment shows that all qualification testing was performed at a peak pressure of at least 69 psig with durations in the order of several minutes. Since this pressure and its' associated saturated temperature of 315°F is more severe than the postulated peak saturated temperature of the MSLB, we concluded that all EEQ Master List equipment in containment could be qualified for containment pressures up to 69 psig, and that this equipment is operable for the postulated transient, and the ensuing post accident operating time.

III. Independence of Feedwater Control Valve and Feedwater Block Valve

To provide additional assurance on the reliability of using either the feedwater block valve or the feedwater control valve for each respective steam generator, an evaluation was conducted to determine the degree of electrical separation between the valves. The following describes the electrical separation between the valves.

Each steam generator has a feedwater block valve and feedwater control valve in series on the secondary side of the steam generator. For steam generator "1", feedwater block valve FW-42A is fed from MCC B12 which is fed from 4kv bus 24A via a load center transformer. Feedwater control valve FW-51A is powered from AC Panel IAC-1 which in turn is fed from MCC bus 24C. Bus 24C is connected to bus 24A and this bus is connected to the NSST through breaker 52-A102. The common electrical power source is bus 24A for these two valves.

For steam generator "2", the feedwater block valve is FW-42B and the feedwater control valve is FW-51B. Feedwater block valve FW-42B is fed from MCC 21 which is connected to 4kv bus 24B via a load center transformer. Feedwater control valve FW-51B is powered from AC panel IAC-2 which in turn is fed from MCC 61 which is then powered via a load center transformer from the 4kv bus 24D. Bus 24D is connected to bus 24B and this bus is connected to the NSST through breaker 52-A206.

The control power for both feedwater block valves is from the control transformer located in each MCC for the individual valve. A loss of power for an individual feedwater block valve does not affect the redundant feedwater control valve. In addition, the control power for the turbine trip scheme was reviewed. This scheme is powered from a separate 24 vdc battery for the turbine generator EHC control system.

The routing of the electrical cables for FW-42A and FW-51A are in separate trays and conduits except a single tray located in the cable vault room. The electrical cables for FW-42B and FW-51B share three trays located in the turbine building.

The probability of a common mode failure of the cables located in the same cable tray during the short time interval of concern with the hypothetical main steam line break is approximately 10^{-7} while the probability of the failure of a 4kv bus is less than 10^{-6}.

The separation of the electrical power for control and operation of the valves to their respective 4kv buses and the probability of an event rendering either 4kv bus unavailable is low. Although the routing of the cables for the redundant valves do share the same cable trays in some locations, the probability of a single event affecting all the cables in the tray is also sufficiently low.

CONCLUSION

Based upon the information provided above, there is a reasonable assurance that with the actions of a dedicated operator or implementation of one or more of the alternative corrective actions, the containment pressure will remain below the design basis value during a main steam line break event. If one or both of the first two options identified on page 6 is implemented, there would be no further need for the dedicated operator. Therefore, it is our determination that the continued operation of Millstone Unit No. 2 for the remainder of cycle 11 will not involve any undue risk to the health and safety of the public.

REFERENCES

1. NUSCO QA calculation 90-RPS-736 EM Rev. 5

2. NUSCO QA calculation 89-078-073 EM Rev. 0

3. Bechtel specification 7604-M-2706, Rev. 5.