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August 23, 1985

Docket No. 50-245
B11662

Director of Nuclear Reactor Regulation
Attn: Mr. Christopher I. Grimes, Chief
Systematic Evaluation Program Branch
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Gentlemen:

Millstone Nuclear Power Station, Unit No. 1
Integrated Safety Assessment Program
Summaries of Public Safety Impact Model Project Analyses

In a letter dated July 31, 1985,(1) Northeast Nuclear Energy Company was requested to provide the Staff with summaries of the public safety risk oriented analyses of a selected number of projects we are evaluating in the Integrated Safety Assessment Program (ISAP).

In response to this request, and in accordance with our understanding of the ISAP process, we are providing the Staff with summaries of the following projects we have evaluated for public safety impacts:

1) ISAP Topic No. 1.04 - "RWCU System Pressure Interlock"

2) ISAP Topic No. 1.06 - "Seismic Qualification of Safety-Related Piping"

3) ISAP Topic No. 1.07 - "Control Room Design Review"

4) ISAP Topic No. 1.18 - "Anticipated Transient Without Scram (ATWS)"

5) ISAP Topic No. 2.01 - "LPCI Remotely Operated Valves 1-LP-50A and B"

It is noted that since we have not completed our analyses of the entire set of ISAP projects, the public safety impact scores are to be considered preliminary at this time. Upon completion of our analyses of the entire ISAP project set, including all five attributes, we will review our analyses and revise our public safety impact results, if necessary, to assure consistency in the ranking of the ISAP projects.

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(1) H. L. Thompson letter to J. F. Opeka, "Integrated Safety Assessment Program," July 31, 1985.

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As further public safety impact analyses are completed, we will promptly forward summaries to the Staff for review.

If you have any questions on this material, please feel free to contact my staff.

Very truly yours,

NORTHEAST NUCLEAR ENERGY COMPANY

J. F. Opeka

J. F. Opeka
Senior Vice President

E J Mroczka

By: E.J. Mroczka
Vice President

cc: J. A. Zwolinski

## ISAP #1.04 RWCU System Pressure Interlock

### Safety Issue

As described in Section 1.1.4 of the Millstone Unit 1 Probabilistic Safety Study (P.S.S.), the Reactor Water Cleanup (R.W.C.U.) System is primarily designed to remove corrosion products from the reactor coolant during all modes of plant operation. The system is also used to let down water from the primary system by diverting it directly to the Main Condenser hotwell.

When the R.W.C.U. system is operating, reactor water flows from the R.P.V. through two sets of heat exchangers (e.g., regenerative and non-regenerative heat exchangers) to a pressure regulating valve, where the water pressure is reduced from over 1000 psig down to 140 psig. The pressure regulating valve is air operated and will fail closed if the air supply is lost. In the event that the pressure regulating valve fails open, the R.W.C.U. system is isolated automatically via two motor operated valves that are located in the high pressure piping between the R.P.V. and the first set of heat exchangers. These valves receive a close signal from a high pressure interlock which is located downstream of the pressure regulating valve, at the interface between the high and low pressure piping.

Because of the fact Millstone Unit 1 has only one pressure interlock, there is a concern that its failure could cause the low pressure side of the R.W.C.U. system to become overpressurized. Assuming an additional failure of the downstream relief valve, system overpressurization could subsequently lead to pipe failure, causing a LOCA outside of containment.

### Proposed Project

The proposed project involves installation of a second, independent pressure interlock to ensure system isolation in the event that the pressure regulating valve fails wide open.

**Analysis of Public Safety Impact**

The public safety impact of this proposed project was evaluated using Method A. Overpressurization of the low pressure piping on the R.W.C.U. without system isolation could lead to a small break LOCA either inside or outside of containment, depending on whether or not the protective relief valve lifted. Although the low pressure piping is 8" stainless steel, the equivalent break area is only 0.018 ft.$^2$ due to pressure drops caused by:

- o the heat exchangers
- o a 3" diameter regulating valve
- o a 3" diameter restricting orifice

all of which are located in series just upstream of the low pressure interface.

In addition to the high pressure interlock, the R.W.C.U. system is presently designed to isolate on indications of:

- o low reactor water level
- o high non-regenerative heat exchanger temperature (140$^{\circ}$F)

The Millstone Unit 1 P.S.S. model was used to determine the sensitivity of the R.W.C.U. system isolation function to the addition of a second pressure interlock. Although the P.S.S. looked at all the ways in which an interfacing system LOCA could initiate, this analysis focuses only on those initiators which could be mitigated by the additional pressure interlock. For example, the P.S.S. examined a LOCA that could occur on the outlet side of R.W.C.U. due to failure of valves on the clean-up pump bypass line. This particular break would not be mitigated by the pressure interlock since the regulating valve would still be controlling pressure on the inlet side of the system. Unless the pressure regulating valve failed coincident with the valves on the bypass line, the pressure interlock would not be challenged by this incident. Accordingly, failure of the pressure regulating valve as an initiating event with subsequent failure of the pressure interlock was examined.

If the pressure regulating valve (CU-10 in Figure #1) fails open while the R.W.C.U. system is operating, there will be an increase of flow in the low pressure piping along with an accompanying pressure rise. At 140 psig the single existing pressure interlock should isolate the high to low pressure

interface by closing the motor operated isolation valves, i.e. either MOV CU-2 or CU-3. In the event that this fails, a second means of isolation is available via the high temperature interlock. The open regulating valve causes an increase in non-regenerative heat exchanger outlet temperature due to increased system flow. When the outlet temperature increases to 140$^{o}$F (normally 120$^{o}$F), the isolation setpoint is reached and the interlock commands both isolation valves to close. Should this fail, a relief valve which discharges to the torus would open to protect the low pressure piping. The result of successful relief valve opening after both interlocks failed would be an isolable SB LOCA with a flow path to the torus. This particular LOCA is roughly 500 times more likely to occur than an interfacing system LOCA where the relief failed to open and the break bypasses containment as discussed below.

The frequency of an unisolable interfacing systems LOCA is calculated using the fault tree model shown in Figure #2 as follows:

$$\lambda = \lambda_{AOVOPENS} \times [(Q_{PRFAIL}) \times (Q_{TEMPFAIL}) + (Q_{MOV2FTC}) \times (Q_{MOV3FTC})] \times Q_{RV68FTO}$$

-where $\lambda_{AOVOPENS}$ is the frequency per year of the pressure regulating valve fails wide open.

Substituting values from the fault tree in Figure 2 gives: $= 3.3 \times 10^{-9}$/yr

The frequency of an isolable SB LOCA with flow return to the torus is:

$$\lambda_{SBLOCA} = \lambda_{AOVOPENS} \times [(Q_{PRFAIL}) \times (Q_{TEMPFAIL}) + (Q_{MOV2FTC}) \times (Q_{MOV3FTC})] \times (1 - Q_{RV68FTO})$$

Substituting values from the fault tree in Figure 3 gives: $= 1.6 \times 10^{-6}$/yr

Based on the higher likelihood of the SB LOCA, a sensitivity study was performed to determine the effect of adding a second pressure interlock in the R.W.C.U. The "new" frequency is calculated by using the fault tree model in Figure 4 as follows:

$$\lambda_{SBLOCA} = \lambda_{AOVOPENS} \times [(Q_{PR1FAIL}) \times (Q_{PR2FAIL}) \times (Q_{TEMPFAIL}) + (Q_{MOV2FTC}) \times (Q_{MOV3FTC})] \times (1 - Q_{RV68FTO})$$

Again, substituting in values from the fault tree in Figure 4 yields:

$$\lambda_{SBLOCA} = 1.4 \times 10^{-7}/yr$$

Adding the second pressure interlock results in lowering the SB LOCA frequency by $2 \times 10^{-7}$/yr. Even if the reactor low water level isolation signal had been considered, the frequency would not be reduced since failure of both isolation valves begins to dominate at the point where 3 isolation interlocks are used.

Given the $2 \times 10^{-7}$/yr. change in SB LOCA frequency, the associated change in core melt frequency is calculated by:

$$\lambda = \lambda_{SBLOCA} \times Q_{AltSDC}$$

Substituting: $\lambda_{SBLOCA} = 2 \times 10^{-7}$/yr. as calculated earlier and $Q_{AltSDC} = .16$ from the Millstone Unit 1 P.S.S. yields:

$$= (2 \times 10^{-7}/yr.) \times (.16)$$
$$= 3.2 \times 10^{-8}/yr.$$

This corresponds to a public risk of:

$$R = 3.2 \times 10^{-8}/yr.\ (1.5)(3. \times 10^{6}\ \text{Man-Rems})(25\ yr.) = 4\ \text{Man-Rem}$$

Where a multiplier of 1.5 is used since a SB LOCA with Alternate Shutdown Cooling failure is a late core melt.

**Results**

The frequency of an R.W.C.U. interfacing system LOCA is already low (i.e. $3.3 \times 10^{-9}$/yr.) and was not considered in calculating public risk since the frequency of SB LOCA for the RWCU is 500 times higher. Based on a maximum 4 Man-Rem reduction for implementing the project, a score of <<0.1 out of 10 was assigned.

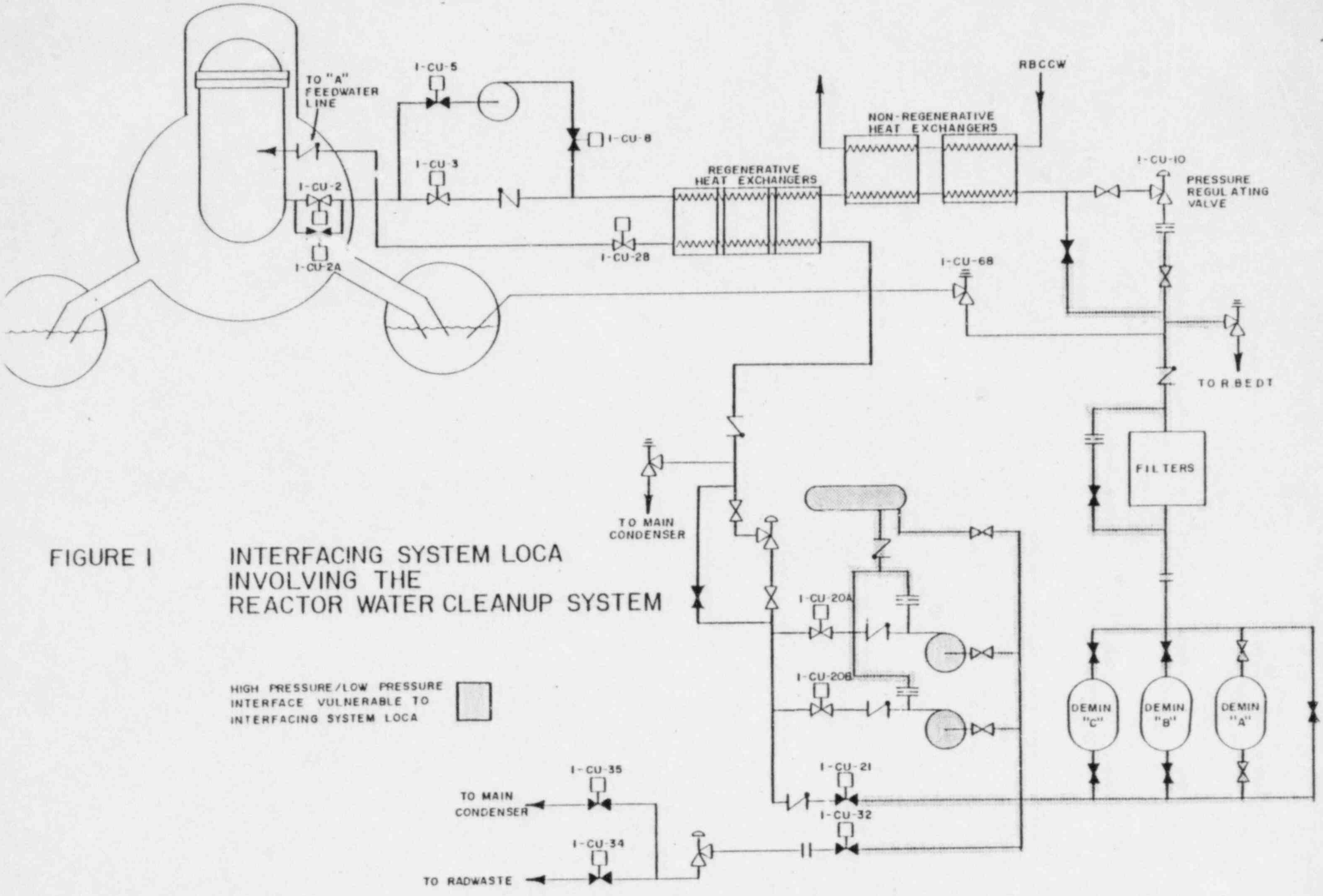



FIGURE I INTERFACING SYSTEM LOCA INVOLVING THE REACTOR WATER CLEANUP SYSTEM

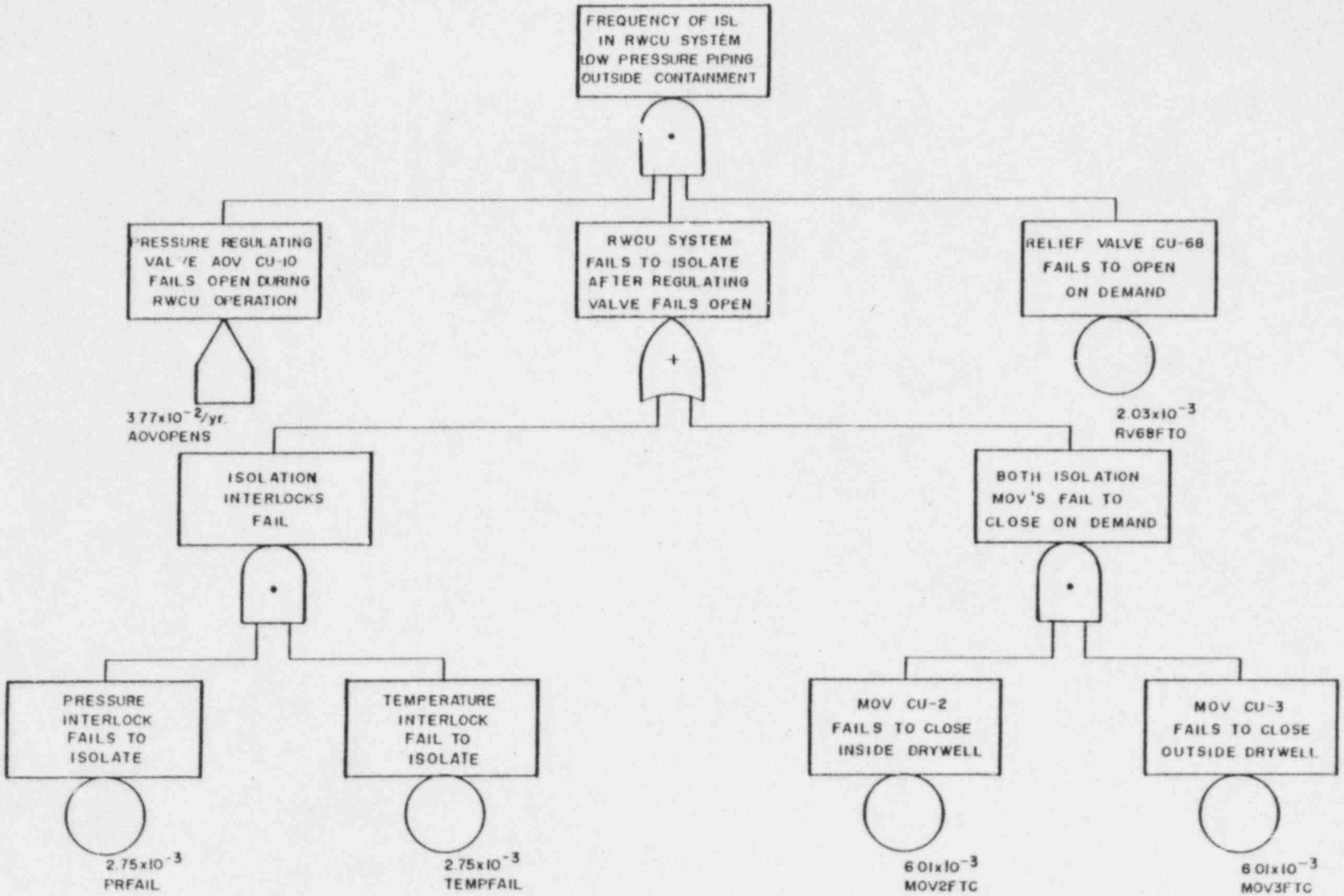



FIGURE 2 ONE PRESSURE INTERLOCK

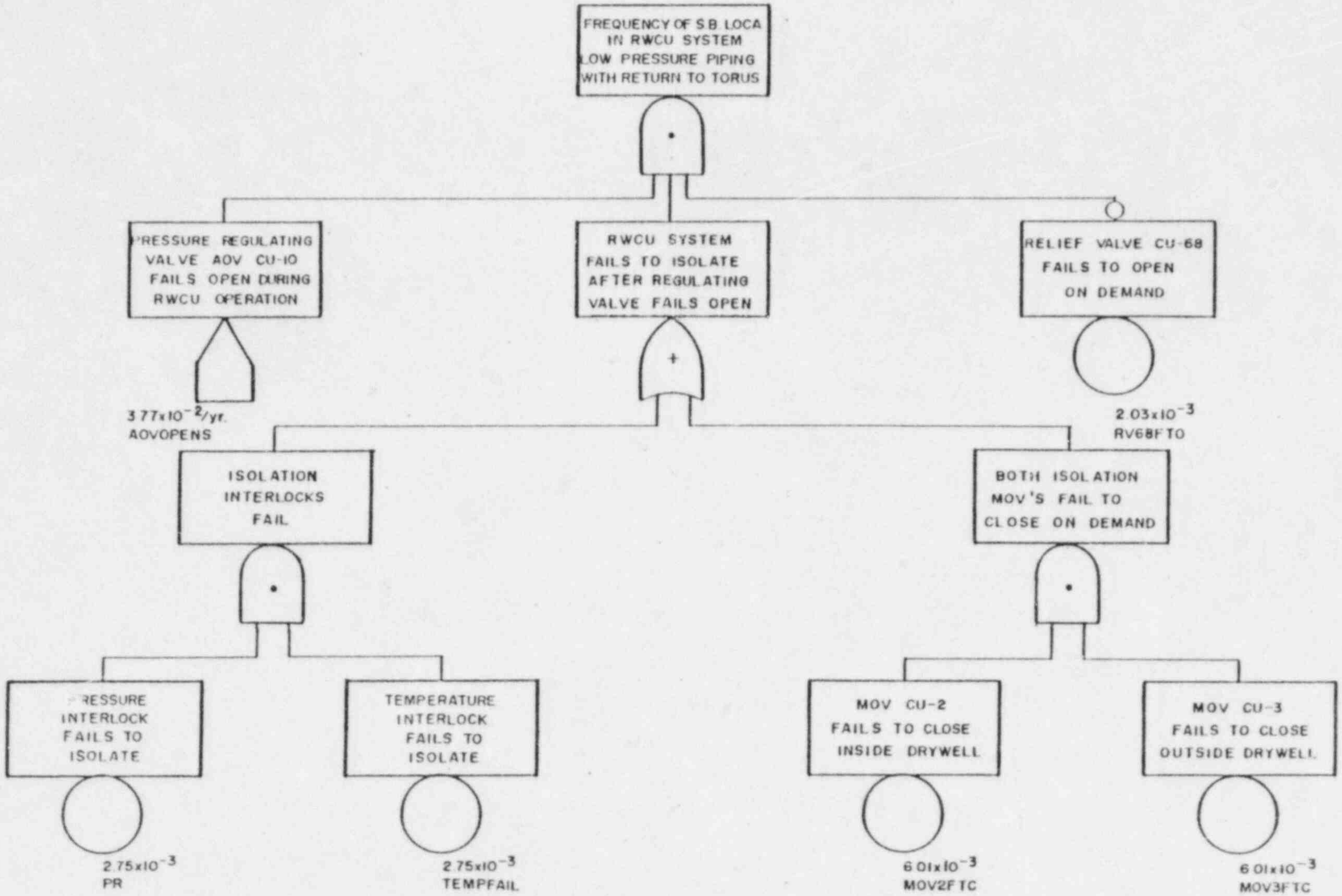



FIGURE 3 ONE PRESSURE INTERLOCK

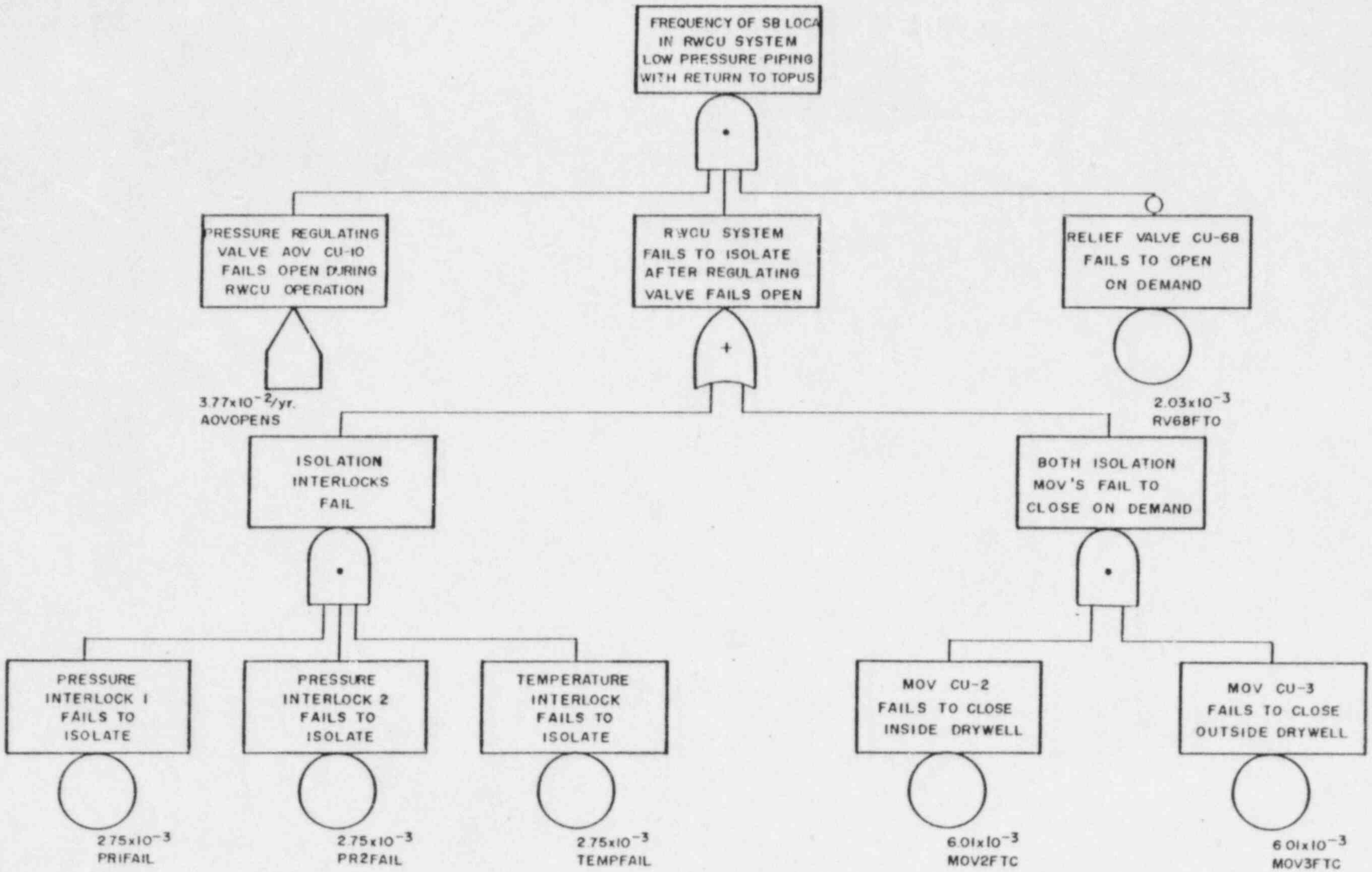



FIGURE 4 TWO PRESSURE INTERLOCKS

## ISAP #1.06 Seismic Qualification of Safety Related Piping

### Safety Issue

In a severe seismic event, it is likely that a reactor trip will occur due to loss of normal power. This disrupts normal operation and places reliance on decay heat removal systems for plant cooldown and long term decay heat removal. An earthquake of sufficient magnitude could fail the decay heat removal systems as well. Failure of all decay heat removal systems will lead to a core melt accident.

All past Seismic PRA analyses have shown that the following components and structures are the most vulnerable to seismic induced failures:

- o Large Water Tanks
- o Structures such as Block Walls and Roof Slabs
- o Anchorage devices for mechanical components such as heat exchangers and pumps
- o Anchorage devices for Instrument Racks and Control Panels
- o Buried piping near buildings
- o Misoperation of Electrical Relays

Although numerous modifications and upgrades have been implemented at Millstone Unit 1 to improve the seismic capacity of the above components, it is expected that the most likely cause of decay heat removal system failure will be similar components and structures. Although unlikely, it is possible that the above ground piping could fail at low ground accelerations when other critical components and structures survive, thus disabling the decay heat removal systems. Further analysis has been suggested to address the risk associated with such an unlikely scenario.

I&E Bulletin 79-14 required field verification of As-Built safety related piping to compare the pipe configuration and support with the design assumed in the analysis performed to show seismic qualification of the plant. The Bulletin also requested any discrepencies found between As-Built and the analysis be reconciled either by reanalysis or by design modifications. The Bulletin applies to all above ground piping with 2-1/2" or larger diameter and

all smaller lines which were analyzed for design of supports etc. For the remaining smaller pipes, only the analysis method used needed to be reviewed to ensure that the method was conservative. I&E Bulletin 79-02 had previously required that the base plates and anchors for pipe supports for the piping which were subsequently covered by I&E Bulletin 79-14 be similarly field verified and analysis methods be reviewed to assure conservatism.

For Millstone Unit 1, all pipes covered by I&E Bulletin 79-14 have been field verified. The differences found between As-Built and the design assumed in the analysis were analyzed with the static method utilizing standard FSAR criteria of acceptable stress level. Because of low acceptable stress level, the FSAR method provided conservative results and showed that several of the piping supports needed modifications. These modifications were grouped in two categories, as discussed below:

- Priority Modifications

  These modifications which were needed to qualify the piping for the Operating Basis Earthquake or O.B.E. (ground acceleration $\leq$ 0.07g) were classified as priority modifications. All these modifications have been completed.

- Upgrading Modifications

  Those modifications which are needed to qualify the piping for the Safe Shutdown Earthquake or S.S.E. (ground acceleration $\leq$ 0.17g) were classified as the upgrading modification. Many of these modifications have been completed. The analysis presented here estimates a reduction in public risk if the remaining modifications are implemented.

**Proposed Project**

As discussed above, an appreciable number of pipe supports have already been modified to qualify the piping for an S.S.E. The proposed project is to modify supports for the remaining safety related piping to qualify them for the S.S.E.

## Analysis of Public Safety Impact

The public safety impact of this proposed project was evaluated using Method B. The piping supports are identified for modifications as a result of the piping analysis which may show that piping stresses exceeded the acceptable values of the FSAR. However, exceeding this conservative value of stress level does not necessarily imply that the pressure boundary will be breached or that the piping will lose its operability following the earthquake. Although not demonstrated by a Millstone Unit 1 plant specific analysis, the piping identified for support modification is expected to serve its function following a safe shutdown level earthquake.

Seismic experience data collected by the Seismic Qualification Utility Group (S.Q.U.G.) and others, and high level seismic tests on piping conducted in foreign countries and in the United States show that piping is not susceptible to failure due to seismic inertia loads (See Enclosure 1 of Reference 1). The data was collected as a part of the analysis performed for USI A-46, "Seismic Qualification of Equipment in Operating Plants."

Recent Seismic PRA Analysis of Millstone Unit 1 vintage and newer plants have also consistently shown that above ground piping is generally not the dominant contributor to the seismic risk (Reference 2). All past analyses have shown that usually the failures of large tanks, structures, mechanical and electrical equipment, and buried piping contribute most to the seismic risks. Therefore, an improvement in the seismic capacity of the remaining above ground piping is not expected to increase the overall seismic capacity of Millstone Unit 1.

Based on data collected by S.Q.U.G. and reviewed by the Senior Seismic Review Advisory Panel (SSRAP), the N.R.C. in Reference 1 has also concluded that mechanical and electrical equipment of types commonly used in nuclear power plants are unlikely to fail at earthquake levels typical of the S.S.E. for U.S. nuclear power plants on the East Coast. Reference 1 also states that there is strong evidence that accident mitigating systems would function as designed in the unlikely event they are required following an S.S.E. The analyses performed by the N.R.C. for Millstone Unit 1 under S.E.P. Topics # III-6 and III-11 came to similar conclusions (Reference 3). Considering the fact that the past PRAs

have shown piping to typically have higher fragilities than the rest of the systems, no significant improvement in public risk (incremental improvement) is expected if the piping supports are modified. Based on engineering judgement the project was thus assigned a score of 0.1 out of 10.

**References**


1. Letter from Harold Denton (N.R.C.) to Victor Stello,Jr. (N.R.C./C.R.G.R.), "Proposed Requirements Resulting From Resolution of USI A-46, Seismic Qualification of Equipment in Operating Plants," June 7, 1985.

2. "Seismic Risk: Sensitivities and Contributors", by M. K. Ravindra, R. P. Kennedy and D. H. Worledge. Presented at the 8th International Conference of Structural Mechanics in Reactor Technology, Brussels, Belgium, August, 19-23, 1985.

3. Letter from Dennis M. Crutchfield (N.R.C.) to W. G. Counsil (N.U.), "S.E.P. Safety Topics III-6, Seismic Design Considerations and III-11, Component Integrity - Millstone Nuclear Power Station Unit 1", July 6, 1982.

## ISAP #1.07 Control Room Design Review

### Safety Issue

The safety issue which led to the desire to perform systematic control room design reviews was the recognition that the control rooms in many nuclear power plants contain significant human engineering deficiencies. Such human engineering deficiencies have been identified as the root cause behind:

- o unintentional plant shutdowns and transients caused by operation of the wrong device by a control room operator
- o unintentional disabling of decay heat removal and engineered safeguards systems due to operator errors while manipulating controls
- o premature termination of engineered safeguards systems due to cognitive errors arising from incorrect interpretation of control board instruments.

Each of these types of problems is discussed below.

Inadvertant transients and plant shutdowns have occurred in some plants where control switches which must be periodically manipulated are located adjacent to switches (of similar size, shape, or color) whose change of state will result in a plant transient necessitating reactor trip. Identifying such switches and making improvements will eliminate both a safety and a plant reliability problem. By eliminating a potential source of a future transient, a potential initiating event which could possibly be the first system failure of a core melt accident sequence is eliminated, and public risk is reduced.

Human engineering deficiencies in some nuclear power plants have led to the unintentional disabling of decay heat removal and engineered safeguards systems during transient events. In many cases problems related to control board layout resulted in operators selecting and manipulating the wrong switches. This problem is exacerbated during periods of increased stress (during response to a transient), when multiple manual actions are required in a relatively short period of time, or when the operating crew is relatively inexperienced. Another type of human engineering deficiency is the violation of a populational stereotype in rotary switches or thumbwheel controllers.

The final category of control board human engineering deficiencies are those design features (indicating devices in particular) which could result in the misdiagnosis of plant status by an operator. A good example is a valve position indicator light which is based on actuator status and not actual valve position.

To identify such potential human engineering deficiencies a study of the Millstone Unit 1 control room design including review of the panel layouts has been proposed. The study will be performed in accordance with the recommendations of NUREG-0578 and NUREG-0737, Supl. 1. The outcome of the study will be the identification of any human engineering deficiencies and potential modifications of control room configuration that would help the operator in preventing accidents or in coping with accidents if they occur.

**Proposed Project**

The proposed project involves a systematic review of control room design. The outcome of the review will be the identification of recommendations for possible control room design changes. The scope of the project evaluated does not include implementation of these recommendations.

The review will include the following items:

- Identification of control room operator tasks, and information and control requirements during emergency operations. This will be achieved by walking through the emergency and off-normal operating procedures.

- A comparison of the display and control requirements with a control room inventory to identify missing displays and controls.

- A control room survey to identify deviations from accepted human factors principles.

## Analysis of Public Safety Impact

The public safety impact of this project was evaluated using engineering judgement (Method B) based on the engineering insights obtained from the Millstone Unit 1 Probabilistic Safety Study. Three potential sources of benefits were each considered, and these are:

- o Benefits from avoiding human error caused transients
- o Benefits from avoiding human error caused system unavailability
- o Benefits from avoiding cognitive errors due to instrument deficiencies.

## A. Benefits From Avoiding Transients Caused By Human Error

Millstone Unit 1 has operated since October 1970 and the experience data base from all operating events was systematically considered when the Millstone Unit 1 P.S.S. was performed. A review of this operating experience indicated that in most cases where transients were caused by human error (attributable to a human engineering deficiency) the errors occurred outside of the control room. A good example are plant trips initiated during instrument surveillance testing. Such trips have resulted in revisions to procedures, improvements in operator and I&C personnel training, and in some cases modifications to the control board mimic's, displays, color coding, and control switches. Throughout the process, operations and I&C personnel have provided inputs and suggestions. These types of improvements were driven by the need to assure high plant reliability. While not a part of a full scope control room design review effort, they have already accomplished the equivalent desired result by concentrating on areas where actual experience (from Millstone Unit 1) indicated deficiencies existed. As a measure of the effectiveness of this long term approach it should be noted that the frequency of plant trips caused by human error has dropped substantially over the past fifteen years of operation along with all other sources of reactor trips.

In terms of public risk perspective, the Millstone Unit 1 P.S.S. assumed an average of ~5.5 plant transients per year due to all causes. This value is dominated by the statistics from the earlier years of plant operation. The likelihood that a full scale control room design review could identify an

improvement that could result in a substantial reduction in the frequency of transient initiators is judged to be highly unlikely. Hence there is an insignificant benefit in risk reduction due to avoiding human caused plant trips and transients.

### B. Benefits From Avoiding Human Error Caused System Unavailability

The second area considered for possible benefits was in avoiding human errors which could result in the unavailability of systems needed to remove decay heat following any plant trip or mitigate an accident. As a part of the human reliability analysis documented in Section 4.0 of the Millstone Unit 1 P.S.S., the control board panel layouts, instrumentation, and annunciators available to the operator were systematically reviewed. A screening criterion was established to identify potential risk sensitive areas which should be examined for potential human engineering deficiencies. This criterion involves the identification of those potential accident sequences in which multiple operator actions must be taken in less than ten minutes in order to prevent the onset of severe core damage. Having identified those sequences and quantified their likelihood, the potential benefits of the control room design review could then be assessed. The following accident sequences were considered:

- Reactor Trip/Failure to Fast Transfer 4160V Buses to the R.S.S.T.

  The operator actions necessary to recover A.C. power would require multiple manipulations of breaker switches on the control board. The layout of such breaker switches is mimicked and color coded, however, the breakers are located in close proximity to each other and this could possibly result in an operator error. The onset of severe core damage would take longer than ten minutes and would additionally require the failure of the Isolation Condenser (both automatically and due to local initiation). The likelihood of such a scenario would be exceedingly small due to the unlikelihood of being in Support State 7 (Station AC Blackout) following a non-LNP event. Because of this there is only minimal benefit to be obtained from additional improvements.

- Loss of Feedwater/Failure of Isolation Condenser or S/R Valve Open.

  The operator actions necessary for recovery from this event

involves starting any two low pressure pumps and initiating Emergency Vessel Depressurization (EOP 577). The frequency of a similar sequence was quantified (See Table 5.3-5, Millstone Unit 1 P.S.S.) and found to be important. (This sequence, however, was dominated by cognitive error in failing to recognize the need to control level.) The controls necessary to perform this action were then reviewed in detail. The control board layout for the switches involved was found to involve no human engineering deficiencies which could be improved.

- Small LOCA/Failure of Automatic E.C.C.S. Actuation

  The operator action necessary for recovery from this event involves maintaining water level using any available pump. Failure to take action within ten minutes will result in the onset of severe fuel damage. The likelihood of this sequence is exceedingly small; hence there is minimal benefit to be obtained from improvements to the control boards as a result of a control room design review.

- Transient with Main Condenser Unavailable/Failure to Scram

  The operator actions necessary to mitigate such a scenario involve the manipulation of the feedwater control system in order to reduce RPV water level. The key control board area involved in this action is the feedwater regulating GEMAC controllers. This was reviewed for possible human error deficiencies and it was concluded that no significant problems exist which could be corrected as a result of a control room design review.

In general the issues which dominate public risk at Millstone Unit 1 are related to equipment capacity (long term cooling), specific hardware design configurations and single failures (Shutdown Cooling, Isolation Condenser, and Alternate Shutdown Cooling), and cognitive errors in recognizing the need to take manual control of water level. These types of issues are not resolvable by improvements to the control board layout.

**C. Benefits From Avoiding Cognitive Errors Due to Instrument Deficiencies**

The final area considered in which a comprehensive control room design review

might provide benefits was: the existence of instrument deficiencies which could result in cognitive errors. The new sympton-based emergency operating procedures (EOPs) were thoroughly walked through prior to their implementation in June 1983. The instruments available in the control room are sufficient to carry out all actions listed in the EOPs. A review of the Millstone Unit 1 P.S.S. indicates that the most dominant cognitive errors are:

- o Failure to ignore indicated water level and flood the RPV when containment temperatures are likely to cause reference leg flashing.
- o Failure to recognize the need to manually control water level.

The first issue is related to the location of the reference leg in the drywell and can not be improved upon by anything done in the control board layout. The second issue is a pure cognitive error which occurs despite the availability of eight different water level indicators. Again for this error there is no remedy which can be provided for by control board layouts.

**Results**

Over the last 15 years of plant operation, a number of control room layouts have been improved. The current design (re-layout) provides clear and unambigious information to the operator. Any further improvements to control room layout is not expected to provide any significant improvement in human reliability of the operator during an emergency. Based on engineering judgment a score of 0.1 out of 10 is assigned to this project.

ISAP #1.18 **A.T.W.S.: Upgrading of the Standby Liquid Control System**

**Safety Issue**

To mitigate Anticipated Transients Without Scram (A.T.W.S.), each boiling water reactor uses the three following systems:

- o Alternate Rod Insertion (A.R.I.), diverse from the Reactor Protection System to provide an alternate path for actuating the scram pilot valves.
- o Recirculation Pump Trip (R.P.T.), to automatically trip the Recirculation Pumps.
- o Standby Liquid Control System (S.L.C.S.), capable of injecting sodium pentaborate solution into the core.

The Millstone Unit 1 design currently incorporates all of these features.

Because of the severe nature of A.T.W.S. transients, prompt operator actions are taken to address two basic concerns:

- o Reduce heat input to the torus while the S.L.C.S. is being injected.
- o To maintain stable steam condensation loads on the torus to maintain torus (containment) integrity.

The operator is required to initiate S.L.C.S. promptly and to lower the R.P.V. water level. Lowering R.P.V. water level reduces the core power and thus heat load on the torus. Following this, the operator needs to depressurize the R.P.V. when the torus heats up to the heat capacity temperature limit (H.C.T.L.). Lower R.P.V. pressure eliminates unstable steam condensaion as the torus heats up. With the current 43 g.p.m. S.L.C.S., any error in operator judgement or even a significant delay in action will lead to severe core damage.

The current S.L.C.S. design provides a flow rate of 43 g.p.m. at 13 weight percent sodium pentaborate solution. Upgrading the S.L.C.S. by increasing the flow rate to 86 g.p.m. (or boron equivalent in shutdown worth) will reduce the time it takes to inject the necessary amount of boron required to achieve hot shutdown. Upgrading of the S.L.C.S. to 86 g.p.m. injection rate is thus

expected to improve the ability of Millstone Unit 1 to cope with A.T.W.S. sequences of the types already considered in the Millstone Unit 1 Probabilistic Safety Study.

**Proposed Project**

The current S.L.C.S. at Millstone Unit 1 (discussed in Section 3.2.22 of the Millstone Unit 1 P.S.S.) provides a flow rate of 43 g.p.m. At this rate it takes about 35 minutes to inject the amount of boron necessary (270 lbs.) to achieve hot shutdown.

The proposed project is to obtain a flow rate of approximately 86 g.p.m. of 13 weight percent of sodium pentaborate or the equivalent. (The actual flow rate will be calculated for Millstone Unit 1 based on R.P.V. diameter, core power, etc.) The options to achieve this being considered are:

- o Increase the flow rate to 86 g.p.m. via providing the capability for 2 S.L.C. pump operation.
- o Increase the concentration of sodium pentaborate to achieve the equivalent of 86 g.p.m. at 13 weight percent
- o Use enriched boron ($B^{10}$) in the sodium pentaborate solution to achieve the equivalent of 86 g.p.m. at 13 weight percent.

**Analysis of Public Safety Impact**

The public safety impact of this proposed project was assessed using Method A. All three options reduce the time it takes to inject the required amount of sodium pentaborate and achieve hot shutdown by 50%. The proposed system, like the current one, will be manually initiated. The purpose of the analysis presented here is to evaluate the relative merit of 86 g.p.m. capability and not which is the best way to achieve it.

All A.T.W.S. events can be classified into one of two categories:

1. A.T.W.S. events where the Main Condenser remains available for heat removal.
2. A.T.W.S. events where the Main Condenser is isolated and the core heat is discharged into the torus. The most severe transient in

this category is initiated by the M.S.I.V.s closing. Therefore, this category is discussed assuming the initiator is the M.S.I.V. closure.

During A.T.W.S. events where the Main Condenser is available, core power begins to increase as the feedwater subcooling increases following turbine trip. The operator is instructed to run back and then trip the recirculation pumps (See EOP 572, Appendix 2B of the Millstone Unit 1 P.S.S.). The recirculation pumps need to be tripped before the M.S.I.V.s close on high steam flow or Turbine Bypass Valves (T.B.V.s) close on low condenser vacuum. In such an A.T.W.S. event, the core power increases above 105%, which is the capacity of the Millstone Unit 1 Main Condenser. Therefore, the Main Condenser will slowly begin to loose vacuum, which eventually causes the T.B.V.s to close. Following recirculation pump trip, core power drops to about 50 - 55%. At this point, the operator can initiate the S.L.C.S. However, the plant is in a quasi-steady state where the feedwater system and Main Condenser maintain the R.P.V. water level and pressure. In this scenario, the rate at which the boron is injected is not significant. Hence, increasing flow rate of S.L.C.S. from 43 g.p.m. to 86 g.p.m. does not offer any advantage.

If the operator fails to trip the recirculation pumps before isolation of the R.P.V. occurs (due to either M.S.I.V.s or T.B.V.s closing), it becomes an A.T.W.S. event with the Main Condenser isolated, which is discussed next.

In an A.T.W.S. event initiated by M.S.I.V. closure, the recirculation pumps trip automatically on high dome pressure. As shown in Figure 1, the core power decreases to 30 - 35%, but slowly returns to 50 - 55% as the feedwater maintains the R.P.V. level. The results presented in Figure 1 represent a generic plant response, obtained from Reference 1. A similar response would be expected for Millstone Unit 1. Millstone Unit 1 emergency procedure 575 (See Appendix 2B, Millstone Unit 1 P.S.S.) directs the operator to lower the R.P.V. water level to reduce heat input to the torus. As shown in Figure 2, when the torus heats up to 164°F, EOP 580 also requires the operator to depressurize the R.P.V. to maintain the heat capacity temperature limit (H.C.T.L.) of the torus.

With the present system, 270 lbs of boron needs to be injected to achieve hot shutdown. It takes about 34 minutes to inject the boron. During injection, the boron stratifies in the lower plenum. After the required amount of boron is injected, the operator raises the R.P.V. water level to its normal value. The resulting natural circulation flow set up in the core readily mixes the boron (Figure 3, Reference 1).

The public safety benefit of 86 g.p.m. can be evaluated in two ways. If the operator follows the emergency procedures, the peak torus temperature will be lower. With 86 g.p.m. S.L.C.S., the A.T.W.S. event becomes more manageable via operator actions. With increased time available not all operator errors result in severe core damage. These two points are separately considered.

Analysis performed for the Browns Ferry Nuclear Plant (Reference 2) shows that increasing the S.L.C.S. flow rate from 50 to 86 g.p.m. reduces the peak torus temperature from 195°F to 173°F (See Figure 4). If analyzed, Millstone Unit 1 would show a similar reduction in the peak torus temperature. The benefits of this effect, however, cannot be quantified using PRA techniques.

Emergency procedure EOP 575 (See Appendix 2B, Millsone Unit 1 P.S.S.) directs the operator to lower the R.P.V. level when the torus heats up to 110°F to reduce the core power and thus heat load to the torus. With the existing system (43 g.p.m. S.L.C.S.), this step is necessary to prevent containment overpressure (and possible failure) caused by torus boiling. With 86 g.p.m. S.L.C.S., since the time needed to inject the required amount of boron is reduced by one half, such an operator action, although highly desirable, is not required to ensure containment integrity. Therefore, the benefit of 86 g.p.m. S.L.C.S. can be determined by eliminating those core melt sequences associated with this operator action.

To determine the torus temperature, if the operator fails to lower the R.P.V. water level, a simplified torus heat-up calculation was performed. At a rate of 86 g.p.m. and 13 weight percent, the time needed to inject 270 lbs of boron is about 17.5 minutes. Since the operator has not lowered the water level, the natural circulation flow in the core will readily mix the boron and the core power will begin to drop. A core power transient shown in Figure 5 was assumed

in the torus heat-up calculation. This power transient is based on the following assumptions:

- Initial core power, at the time of S.L.C.S. initiation is 55%
- A one minute time delay exists between the time boron is injected and it becomes fully active in the core.
- Core power drops linearly as boron concentration in the core increases (linearly).
- Core power drops to 7% when hot shutdown is achieved (core decay heat)
- Operator initiates S.L.C.S. at a torus temperature of $110^{\circ}F$.

Making these assumptions, the torus begins to boil at about 17 minutes. The containment pressure by the time hot shutdown is achieved is expected to be less than 4 psig. In other words, with 86 g.p.m. S.L.C.S., failure of the operator to lower R.P.V. water level does not result in a containment failure. In the Millstone Unit 1 P.S.S., it was assumed that a core melt is inevitable following containment failure.

The reduction in core melt frequency can be calculated as shown below. With the current S.L.C.S., the frequency of A.T.W.S. core melt sequences are:

$$\lambda = \lambda_1 Q_{RPS} Q_{HEP1} Q_{HEP2} + (\lambda_2 + \lambda_3) Q_{RPS} Q_{HEP3}$$

With the upgraded S.L.C.S. system (86 g.p.m. or equivalent), the frequency of these core melt sequences becomes:

$$\lambda = \lambda_1 Q_{RPS} Q_{HEP1} Q_{HEP2} Q_{SLCS} + (\lambda_2 + \lambda_3) Q_{RPS} Q_{HEP3} Q_{SLCS}$$

The reduction in core melt frequency can then be computed as:

$$\lambda = \lambda_1 Q_{RPS} Q_{HEP1} Q_{HEP2} (1-Q_{SLCS}) + (\lambda_2 + \lambda_3) Q_{RPS} Q_{HEP3} (1-Q_{SLCS})$$

These expressions are evaluated using the following values:

$\lambda_1$= frequency of reactor transients with Main Condenser available, 3.[illegible]09/yr.

$\lambda_2$= frequency of reactor transients with Main Condenser unavailable,

0.435/yr.

$\lambda_3$= frequency of loss of normal power, 0.124/yr.

$Q_{RPS}$= unavailability of the Reactor Protection System, $5.4 \times 10^{-5}$

$Q_{HEP1}$= human error probability for failing to trip the recirculation pumps before Main Condenser is isolated due to M.S.I.V. closure or T.B.V. closure, $1.3 \times 10^{-2}$

$Q_{HEP2}$= conditional human error probability for failing to lower R.P.V. water level given an error in failing to trip recirculation pumps, 1.0

$Q_{HEP3}$= human error probability for the operator failing to lower R.P.V. water level, $1.3 \times 10^{-1}$.

**Results**

Upgrading the S.L.C.S. to 86 g.p.m. (or the equivalent) will result in a drop of $6 \times 10^{-6}$/yr in predicted core melt frequency (0.75% decrease). All of this reduction comes from plant damage state TE2 (See Section 2.2 of the Millstone Unit 1 P.S.S. for the definition of the plant damage states). The resulting risk reduction is about 450 Man-Rems over the remaining life of the plant. A score of 1 out of 10 is thus assessed for this project.

**References**


1. "Power Suppression and Boron Remixing Mechanism for General Electric Boiling Water Reactor Emergency Procedures Guidelines," NEDC-22166, August 1983.

2. "Severe Accident Sequence Analysis Program Anticipated Transients Without Scram Simulations for Browns Ferry Nuclear Plant Unit 1," NUREG/CR-4155, EGG-2379, February 1985.

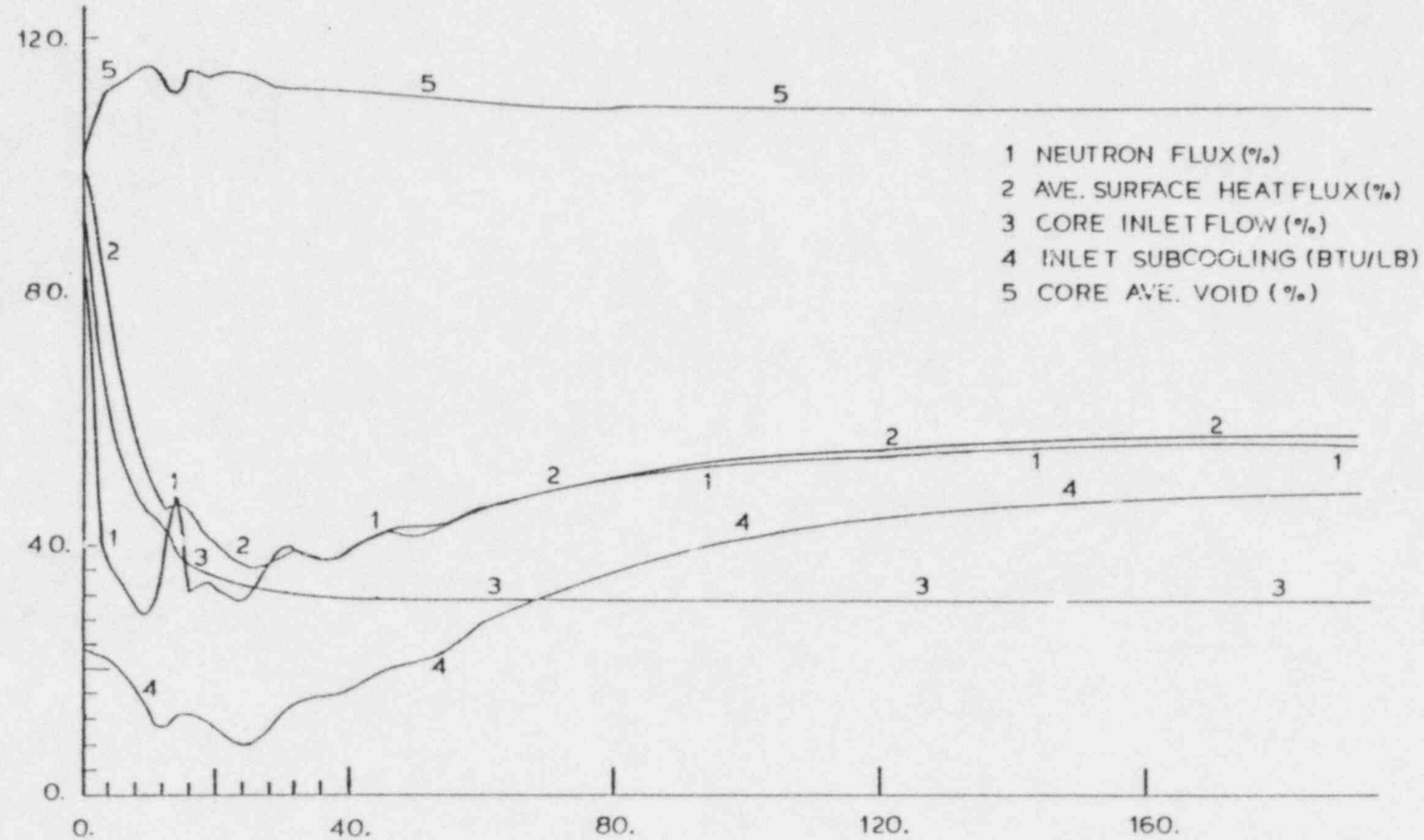



FIGURE 1: CORE POWER FOLLOWING RECIRCULATION PUMP TRIP (Ref. 1)

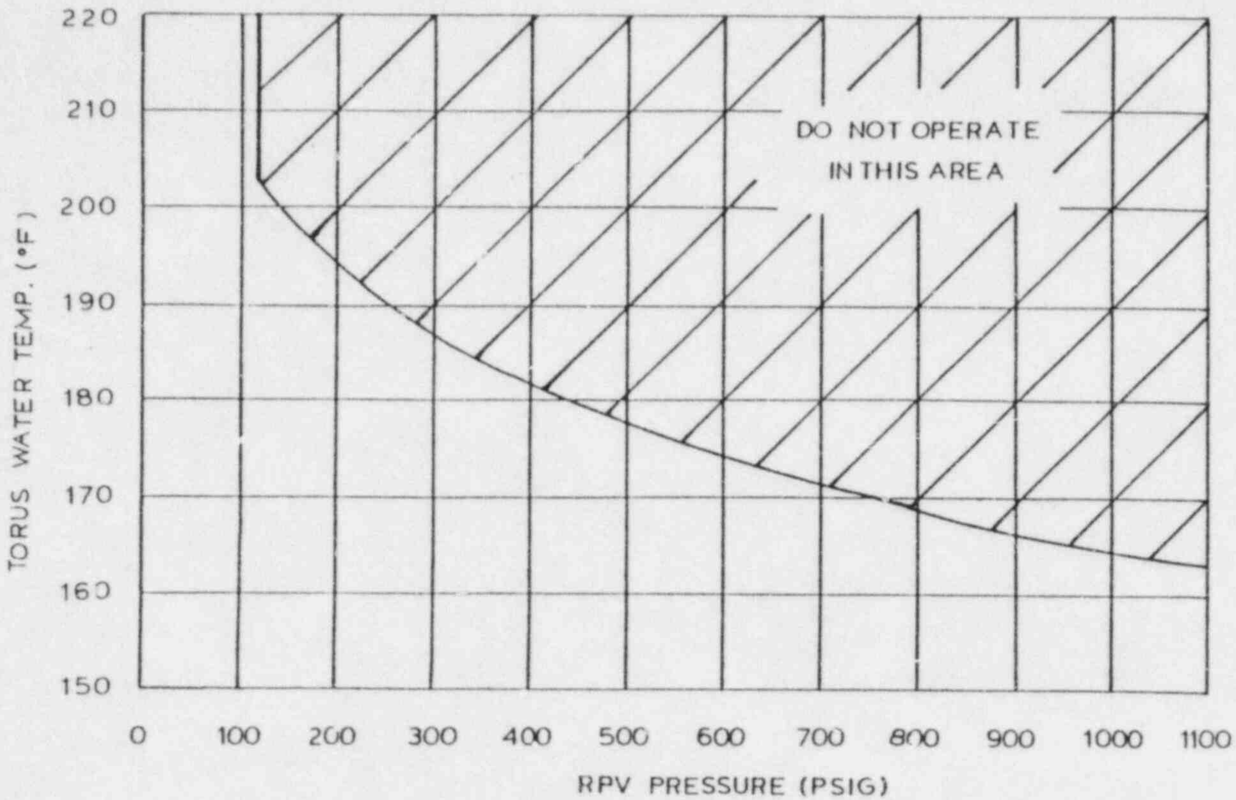



FIGURE 2: MILLSTONE UNIT 1 TORUS HEAT CAPACITY TEMPERATURE LIMIT CURVE

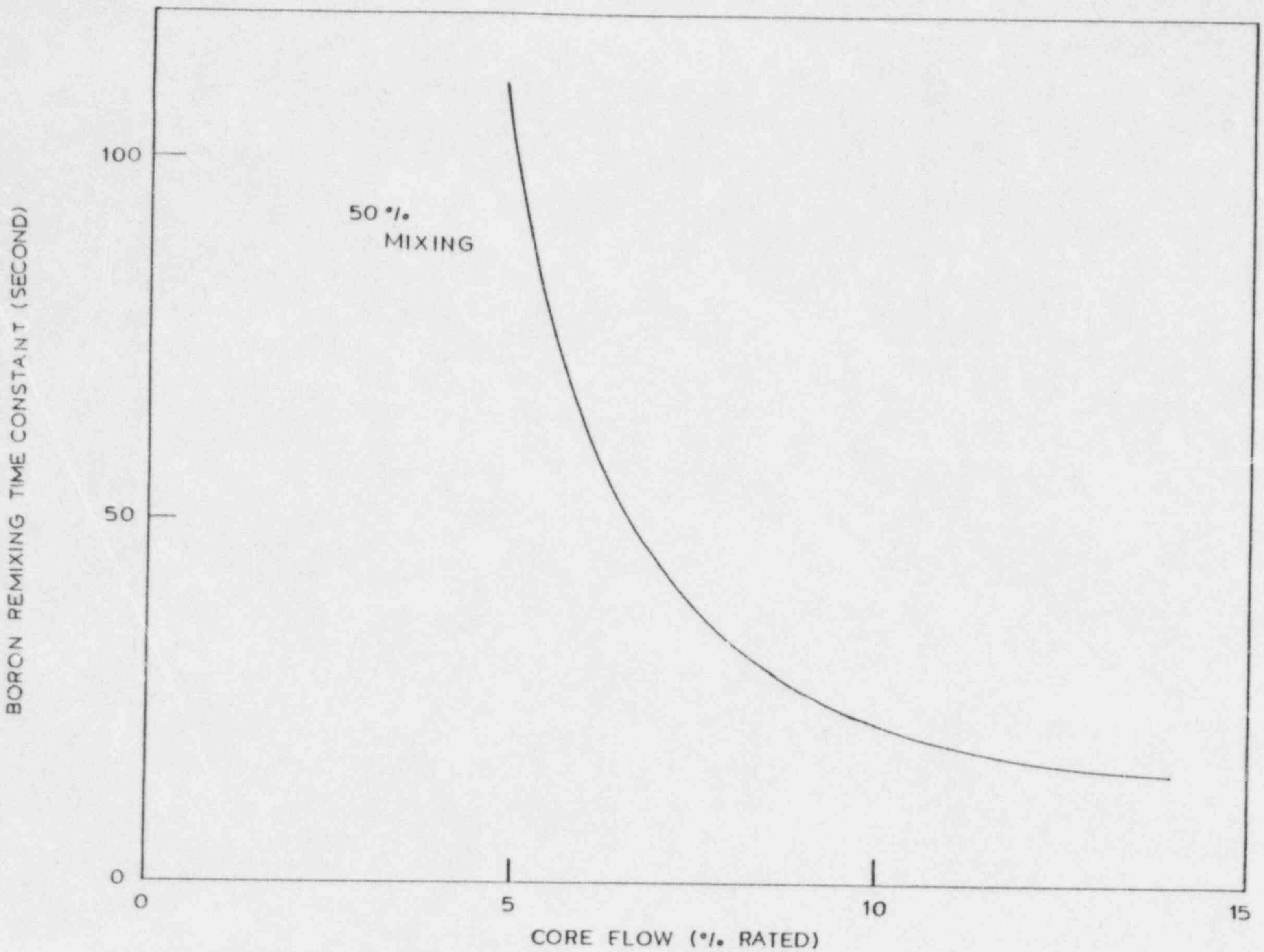



FIGURE 3. NATURAL CIRCULATION BORON REMIXING TIME CONSTANT vs CORE FLOW

NOTE: BORON REMIXING TIME CONSTANT IS DEFINED AS THE TIME REQUIRED TO RAISE IN-CORE AVERAGE BORON CONCENTRATION TO 50% OF TOTAL VESSEL AVERAGE BORON CONCENTRATION

FIGURE 4: TORUS WATER TEMPERATURE SENSITIVITY STUDY FOR BROWNS FERRY, REF. 2

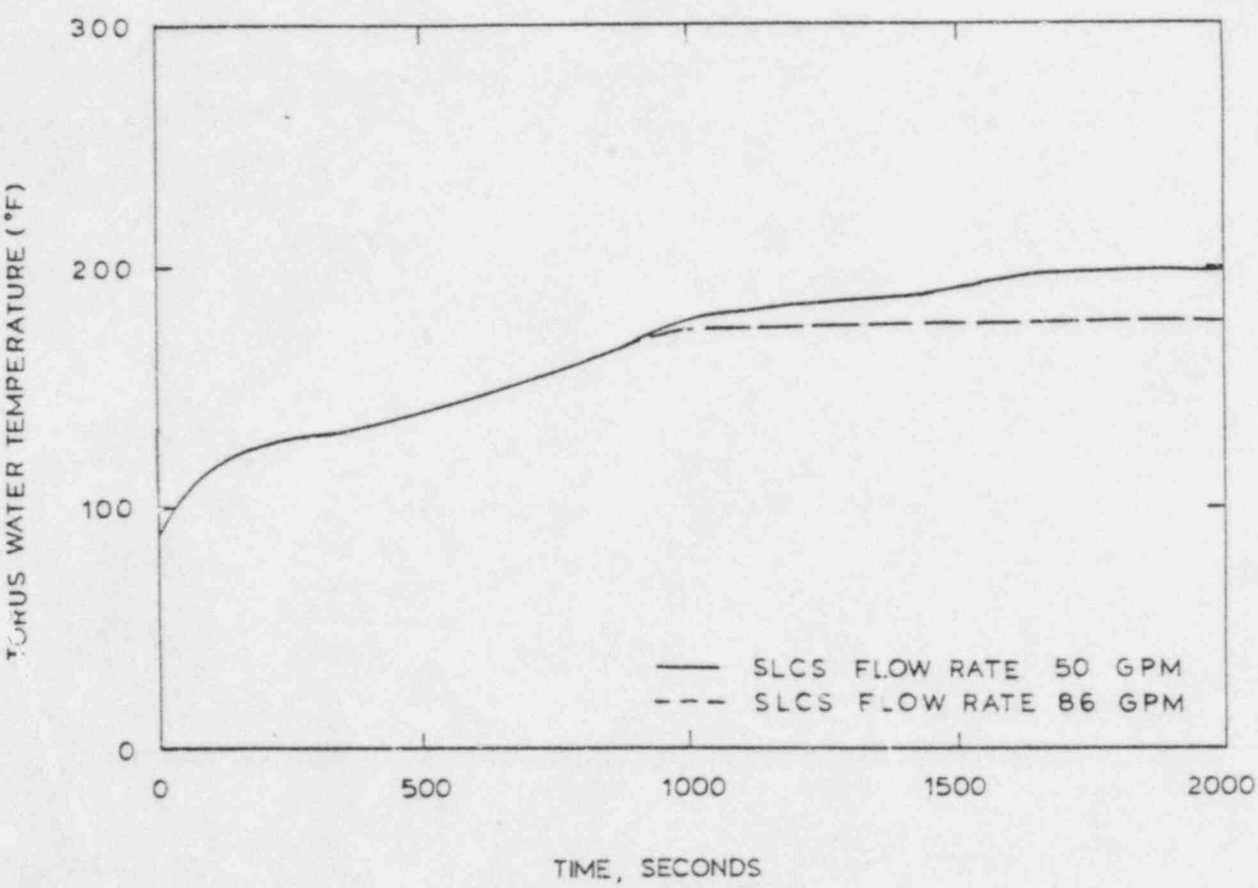
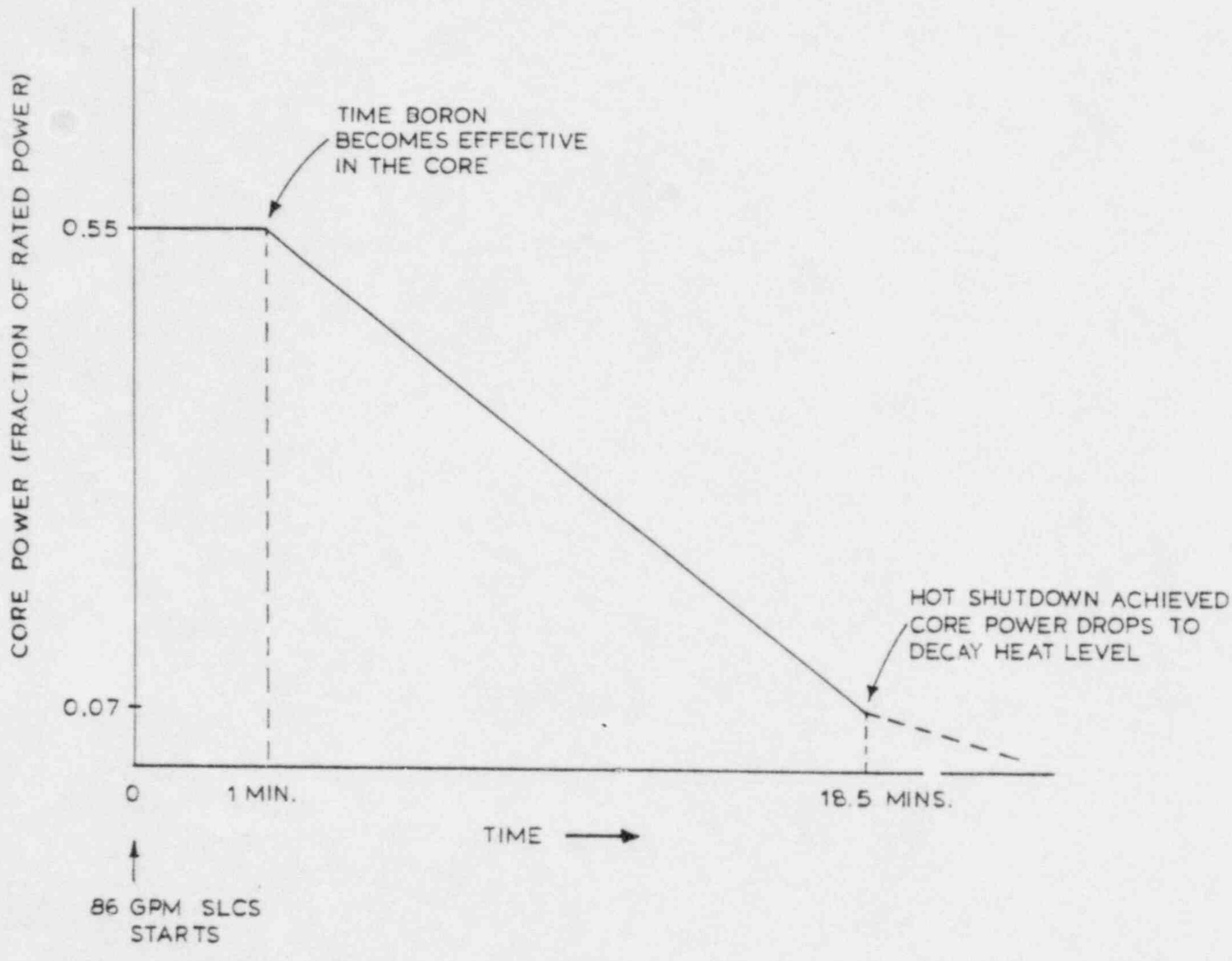


FIGURE 5: CORE POWER TRANSIENT ASSUMED IN MILLSTONE 1 TORUS HEAT-UP CALCULATION



ISAP #2.01 LPCI Remotely Operated Valves 1-LP-50A-8B

**Safety Issue**

The Millstone Unit 1 Technical Specification Section 3.7 (Reference 1) requires that torus water level be maintained within allowable limits. Therefore, the operator needs to periodically drain the torus as its water inventory increases due to normal leakage. The torus is drained by manually opening two normally closed valves 1-LP-50A and 50B. Following an accident, the operator may also need to drain the torus. Such an operation may be needed for example to cool the torus by removing hot water from it and replenshing it with cool water (feed & bleed operation) or during clean up. If the accident resulted in significant fuel damage, the area where these valves are located may become inaccessible due to high radiation level. The project proposes a remote operation capability of these valves, which will ensure capability of draining torus water even when high radiation levels are present.

**Proposed Project**

As shown in Figure 1, the valves 1-LP-50A & B are located on the "B" side of the L.P.C.I. system cross-tie. Flow from these lines is discharged to the Radwaste system. Two options for the remote operation capability of valves 1-LP-50A and B are being investigated, which are:

- Add motor operators to the valves
- Add reach rods to the valves to allow manual operation from remote location (about 40 ft. above the valves).

**Analysis of Public Safety Impact and Results**

The public safety impact of this proposed project was evaluated using Method B. Addition of remote operation capability to the valves 1-LP-50A and 50B does not affect the course of a transient. However, following a severe accident with significant fuel damage, this capability will be helpful in clean-up operation and this may limit any further release to the public.

It was envisioned that the remote operation capability of the valves 1-LP-50A and B may allow the operator to perform a feed and bleed operation on the torus. Such an operation could provide a method of removing long term decay heat. The sequence envisioned was as follows:

- Discharge core decay heat into the torus via opened a safety/relief (S/R) valve.

- Remove hot water from the torus via the valves 1-LP-50A & B

- Replenish the torus water with cold water from feedwater system (via the opened S/R valve.

However, success of feed and bleed operation required a torus draining rate of about 1000 gpm. The actual draining rate of the torus (which can be achieved through a 3" line) and the capacity of the liquid rad waste system are too small to provide any appreciable cooling to the torus.

The negative impact of this proposed modification has also been found to be insignificant as discussed next. Addition of remote operation capability of valves 1-LP-50A and B may slightly increase the probability of inadvertently opening the valves during an accident. Even if the valves are left open, its effect on the transient is negligible. The only concern will be diversion of the LPCI pump flow to the rad waste system. The amount of flow diversion will be small (few hundered gpm) and will not change the success criteria for the LPCI system. In any case, the operator error of inadvertently opening the valves is restorable by closing the valves.

Based on above considerations and the engineering judgment, a score of 0.1 out of 10 is assigned to this project.

**Reference**

1. "Millstone Unit 1 Technical Specifications", Docket No. 50-245

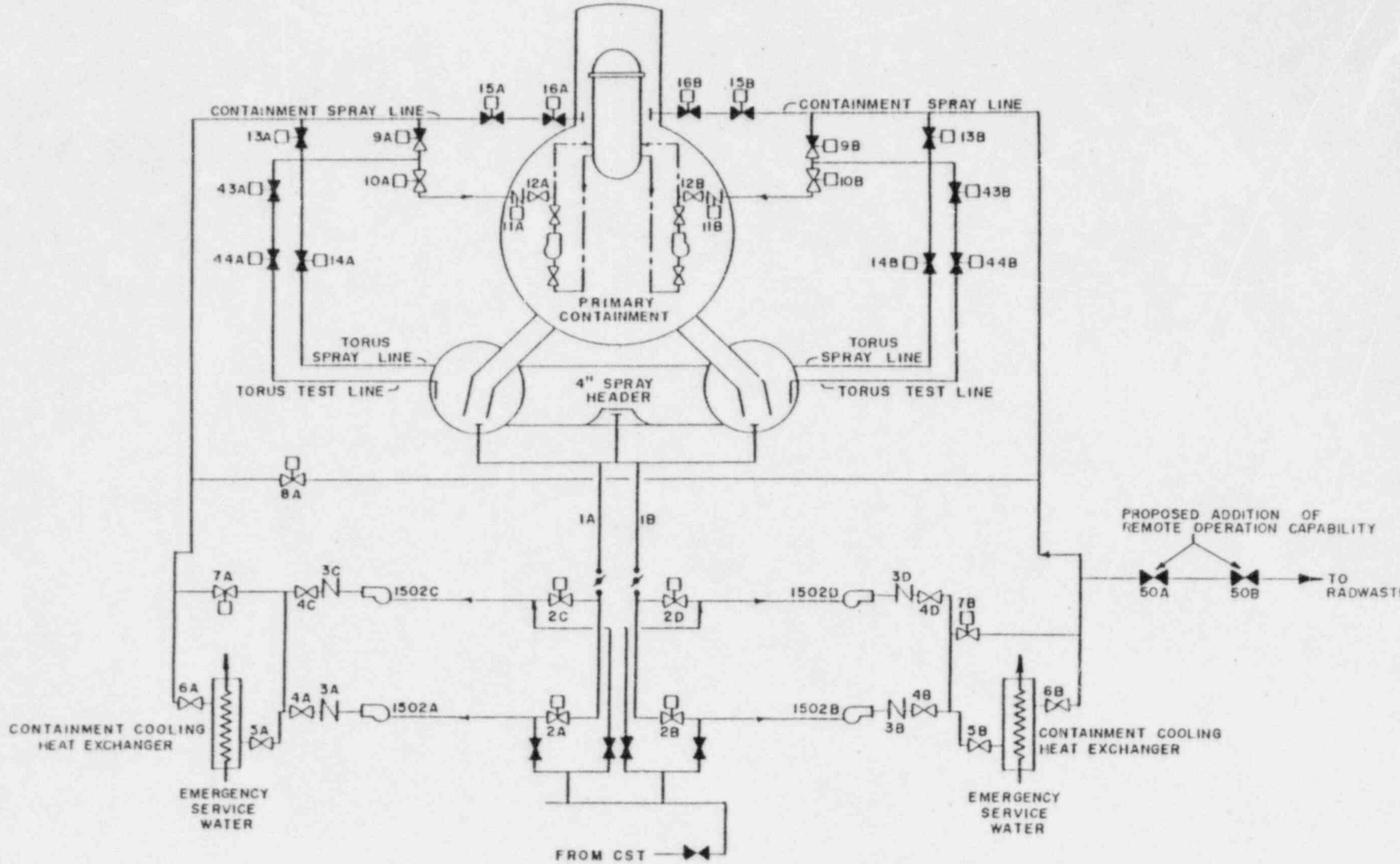



FIGURE I LPCI SYSTEM DIAGRAM

NOTE:

ALL VALVE NUMBERS ARE PRECEEDED BY LP