MARK II OWNERS GROUP

ASSUMPTIONS FOR USE IN ANALYZING MARK II BWR SUPPRESSION POOL TEMPERATURE RESPONSE TO PLANT TRANSIENTS INVOLVING SAFETY/RELIEF VALVE DISCHARGE

REVISION 1

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I. INTRODUCTION

In late 1974, the NRC alerted the BWR Owners to the potential for severe vibratory loads on the containment structure due to safety/relief valve (SRV) discharge at elevated suppression pool temperatures (Reference 1). This phenomenon, or condensation instability, was associated with certain SRV discharge device configurations and occurred above given threshold values of pool temperature and steam mass flux. To avoid this phenomenon, pool temperature limits were established by General Electric Company to stay below this threshold combination of pool temperature and mass flux. In Reference 2 and subsequent letters to other Mark I. utilities, NRC requested that the Mark II utilities analyze pool temperature response to various transients involving SRV discharge and provide this information in their DARs. In response to these requests, several utilities provided preliminary pool temperature response information in their DARs. Subsequently, all of the Mark II plants changed their original ramshead-type SRV discharge device design to quenchers. While the condensation instability phenomenon described above has never been exhibited for quencher devices, even in large scale tests where local temperatures approached saturation, the NRC (Reference 3) has taken the position that a local pool temperature limit of 200°F "will provide additional conservatism and will ensure that unstable steam condensation will not occur with a quencher device." As recognized by the NRC (Reference 3), additional information may be submitted by individual applicants to support the position that the limit may be safely raised or eliminated.

The purpose of this document is to present the generic assumptions to be used in the analysis of suppression pool bulk temperature response to plant transients involving SRV discharge. The analysis results submitted by individual applicants will be based on this document and will identify and justify plant unique variations.

II. TRANSIENTS TO BE ANALYZED

A. General

The NRC (Reference 2) has requested an analysis of pool temperature response to the following five events.

a. A stuck-open SRV during power operation, assuming reactor scram at 10 minutes after pool temperature reaches 110°F and all RHR systems operable;

b. Same as event (a) above except that only one RHR train available;

c. A stuck-open SRV during hot standby conditions, assuming 120°F pool temperature initially and only one RHR train available;

d. The Automatic Depressurization System (ADS) activated following a small line break, assuming an initial pool temperature of 120°F and only one RHR train available;

e. The primary system is isolated and depressurized at a rate of 100°F per hour with an initial pool temperature at 120°F and only one RHR train available.

These events fall into three categories: (a) Stuck-Open Relief Valve (SORV) at Power, (2) Isolation/Scram, and (3) Small Break Accident (SBA).

B. SORV at Power Cases

SORV at power cases, which include NRC events (a) and (b) above, are analyzed to demonstrate that the spurious opening of a SRV during normal power operation will not result in high pool temperatures. Since event (b) with half the heat removal capability of event (a) is the more severe of the two, event (a) was not used for the Mark II analysis. It was recognized, however, that the assumption of main steam isolation valve (MSIV) closure rather than loss of one residual heat removal heat exchanger (RHR HX) could be more severe than event (b) due to loss of the main condenser as a heat sink. This case was, therefore, included in the Mark II analysis.

The Mark II analysis assumes that scram occurs at 110°F. This approach is in agreement with the BWR Standard Technical Specifications (Reference 4), which require the mode switch to be placed in "shutdown" (thereby causing an immediate scram) at 110°F. Since Mark II BWR/4 and BWR/5 plants will not isolate on low level, steam flow to the turbine/condenser continues after scram unless MSIV closure is assumed. Following scram, reduced steam flow to the turbine will result in turbine/generator coast-down eventually generating a turbine trip. Closure of the turbine stop valves following the trip will prevent further steam flow to the condenser until bypass valves can be opened. A typical time for stop valve closure to occur is 20 seconds after scram. Since the mode switch is required to be placed in "shutdown", MSIV closure does not occur on low steam line pressure. Steam flow continues to the seal steam evaporator and the steam jet air ejector, maintaining condenser vacuum. At t = 20 minutes, it is assumed that bypass flow to the main condenser is established to maintain the desired depressurization rate.

The spurious opening of a safety/relief valve at power (and remaining open during the entire transient) is an extremely unlikely event for the domestic Mark II plants.

Detailed assumptions for SORV at power are discussed in Subsection IV.B.

C. Isolation/Scram Cases

Isolation/Scram cases, which include NRC events (c) and (e) above, are analyzed to demonstrate that the loss of the main condenser by the sudden closure of the MSIVs and subsequent scram, SRV openings at set pressure, and manual depressurization will not result in high pool temperature. Two single failures are considered separately; one is the loss of a RHR HX and the other is the failure of a SRV to reclose (SORV). While NRC event (c) considered a SORV at hot standby as the initiating event with a single failure of one RHR HX, it is clear that the decay heat load once the reactor is already in hot standby would be small. As a result, peak pool temperature, even with one RHR HX available, is bounded by the isolation/scram cases considered.

Detailed assumptions for isolation/scram are discussed in Subsection IV.C.

D. SBA Cases

SBA cases are analyzed to demonstrate that SRV discharge required to depressurize the reactor coolant system following a small break will not result in high pool temperatures. As a result of continued flow through the break, peak pool temperature is not reached until after SRV discharge has terminated.

The NRC event (d), SBA with actuation of Automatic Depressurization System (ADS), is less severe from the standpoint of pool temperature than SBA without ADS. Actuation of ADS permits a more rapid transfer to shutdown cooling which is a more efficient mode of heat removal from the reactor/containment system and, thus, results in a lower pool temperature. The more severe event (without ADS), therefore, will be included in the Mark II analysis.

Two cases of SBA are analyzed: one with single failure of a RHR HX and one with loss of shutdown cooling.

Detailed assumptions for SBA are discussed in Section IV.D.

Initial conditions and general assumptions for all cases are given in Section III and Subsection IV.A, respectively.

III. INITIAL CONDITIONS

The following are the initial conditions for all transients:

a. Power level, decay heat standard, and service water temperature to be the same as that used in the FSAR for containment analysis.

b. Suppression pool to be at minimum technical specification water level.

c. Suppression pool to be at maximum technical specification temperature for continuous power operation without pool cooling in operation.

d. Design fouling factors will be considered in determining RHR HX effectiveness.

IV. ASSUMPTIONS

This section provides general assumptions common to all transients and unique assumptions for each transient. Assumptions requiring operator action will be reflected in the operating procedures for each plant.

Technical Specification pool temperatures are defined as follows: TS1 - maximum for continuous power operation without pool cooling in operation; TS2 - maximum for continued testing at power; TS3 - maximum for maintaining reactor critical; and TS4 - maximum for maintaining reactor pressurized. Transients described below are consistent with the above Technical Specifications.

The bases for utilizing certain assumptions (indicated by an asterisk) are provided in Appendix C.

A. General

*1. Feedwater addition to the reactor pressure vessel (RPV).

a. Plants utilizing turbine-driven feedpumps.

i) Upon main steam isolation valve (MSIV) closure, the turbine-driven feedpumps supply feedwater until the discharge head is below the reactor pressure.

ii) For cases where main steam isolation has occurred, the condensate (booster) pump(s) supplies feedwater to the RPV when the reactor pressure is below the condensate (booster) pump discharge head.

b. Plants utilizing motor-driven feedpumps.

i) Feedpumps supply feedwater to the RPV until the feedpumps trip on an automatic signal (e.g., vessel high water level trip).

ii) After the feedpumps have tripped, the condensate (booster) pump(s) supplies feedwater to the RPV when the reactor pressure is below the condensate (booster) pump discharge head.

c. Feedwater will be supplied to the RPV, as described above, until the enthalpy of the feedwater is less than or equal to the enthalpy of the suppression pool water.

*2. Offsite power is assumed available for all cases.

3. HPCI/HPCS injection of suppression pool water into the RPV is determined by the automatic start and stop signals generated (e.g., low-low RPV level starts the HPCI/HPCS injection, high RPV level stops HPCI/HPCS injection).

*Assumption basis is provided in Appendix C.

*4. MSIV closure is complete 3.5 seconds after isolation signal at t = 0 for transients where isolation occurs.

5. All transients involving one RHR HX operation assume a minimum controlled depressurization rate and will employ a rapid transfer (1F minutes) from pool cooling to shutdown cooling using th .vailable RHR HX when the reactor pressure reaches the permissive value. Shutdown cooling will not be utilized in the analyses for those transients having both RHR trains available.

*6. Pool cooling mode is established 10 minutes after TS1 is exceeded.

*7. In accordance with the plant Technical Sp .ifications, manual depressurization begins at TS4 unless the depressurization rate for the event itself (e.g., SORV, HPCI/HPCS cooldown) exceeds the required rate at that time. Manual depressurization is terminated upon initiation of shutdown cooling.

8. SRV flow rate = 122.5 percent of ASME rated.

*9. For plants with suppression pool water contained within the reactor vessel pedestal, the mass of this water is neglected when determining the heat capacity of the pool.

B. <u>SORV at Power</u>

1. The SORV is the initiating event and two cases are considered separately.

    a. Loss of one RHR HX.

    b. MSIV closure signal at t = 0.

*2. In accordance with the Technical Specifications, manual scram occurs at TS3. Manual scram is accomplished in a single manipulation by transferring the mode switch from "run" to "shutdown".

*3. For 1 (a) above, the main condenser remains available.

4. For 1 (b) above, two RHR HX are available.

C. <u>Isolation/Scram</u>

1. Isolation/scram is the initiating event and two cases are considered separately:

    a. Loss of one RHR HX.

    b. Spur f re of a safety/relief valve in the open posit.

*Assumption basis is provic Appendix C.

2. For 1 (b) above, the SORV is assumed to occur at t = 0.

3. For 1 (b) above, two RHR HX are available.

D. Small Break Accident (SBA)

1. The small break is the initiating event and two single failures are considered separately:

   a. Loss of one RHR HX.

   b. Loss of shutdown cooling mode.

*2. Scram on high drywell pressure and MSIV closure signal assumed at t = 0.

3. For 1 (b) above, two RHR HX are available.

V. SUMMARY

In order to facilitate discussion of the transients, each will be identified alphanumerically as follows:

	Transients	Section	Designation
1.	SORV at power - loss of RHR HX	IV.B.1.a	1(a)
2.	SORV at power - spurious isolation	IV.B.1.b	1(b)
3.	Isolation/scram - loss of RHR HX	IV.C.1.a	2(a)
4.	Isolation/scram - SORV	IV.C.1.b	2(b)
5.	SBA - loss of RHR HX	IV.D.1.a	3(a)
6.	SBA - loss of shutdown cooling	IV.D.1.b	3(b)

APPENDIX A

References

1. RO Bulletin 74-14, "BWR Relief Valve Discharge to Suppression Pool", November 15, 1974.

2. Letter, Mr. J. F. Stoltz (NRC) to Mr. E. A. Borgmann (CG&E), September 6, 1977.

3. NUREG-0487 "Mark II Containment Lead Plant Program - Load Evaluation and Acceptance Criteria," October 1978.

4. NUREG-0123, Revision 2, "Standard Technical Specifications - General Electric Boiling Water Reactors (GE-STS)", August 1979.

APPENDIX B

Information from Appendix B is deleted in Revision 1 of this document.

APPENDIX C

ASSUMPTION BASES

This appendix presents the bases for the assumptions listed in Section IV.

1. <u>MSIV Closure</u>

ASSUMPTION: MSIV closure is complete 3.5 seconds after isolation signal at t = 0 for transients where isolation occurs.

BASIS: The 3.5 second closure time includes a 0.5 second instrumentation delay before valve closure begins and a 3.0 second linear closure time. The 0.5 second delay is a typical instrumentation delay and consistent with plant FSARs. The 3.0 second closure is conservatively fast based on the tested nominal values of 5 to 10 seconds. Fast closure of MSIVs and isolation signal at t = 0 maximizes the heat addition to the suppression pool.

2. <u>RHR Pool Cooling</u>

ASSUMPTION: Pool cooling mode is established 10 minutes after TS1 is exceeded.

BASIS: Plant operating experience has shown that RHR pool cooling can be initiated in less than 5 minutes. An alarm will sound at TS1. The plant Technical Specifications and normal and emergency operating procedures state that the operator is to initiate RHR pool cooling when TS1 is exceeded.

3. <u>Manual Depressurization</u>

ASSUMPTION: In accordance with the plant Technical Specifications, manual depressurization begins at TS4 unless the depressurization rate for the event itself (e.g., SORV, HPCI/HPCS cooldown) exceeds the required rate at that time. Manual depressurization is terminated upon initiation of shutdown cooling.

BASIS: The operator will act in accordance with the plant Technical Specifications and normal and emergency procedures for these transient events. There will be a sufficient delay from the first alarm until TS4 is reached so that the operator will be well prepared to begin the required depressurization.

When RHR shutdown cooling is initiated, the decay heat generated is small enough so that the shutdown cooling mode of the RHR is sufficient to cool down and depressurize the RPV. Therefore, manual depressurization is terminated when RHR shutdown cooling is established.

4. Suppression Pool Water

ASSUMPTION: For plants with suppression pool water contained within the reactor vessel pedestal, the mass of this water is neglected when determining the heat capacity of the pool.

BASIS: The mixing between the water in the pedestal and the main portion of the pool is neglected. Taking no credit for the water volume inside the pedestal is conservative.

5. Feedwater

ASSUMPTION: Feedwater addition to the reactor pressure vessel (RPV).

a. Plants utilizing turbine-driven feedpumps.

i) Upon main steam isolation valve (MSIV) closure, the turbine-driven feedpumps supply feedwater until the discharge head is below the reactor pressure.

ii) For cases where main steam isolation has occurred, the condensate (booster) pump(s) supplies feedwater to the RPV when the reactor pressure is below the condensate (booster) pump discharge head.

b. Plants utilizing motor-driven feedpumps.

i) Feedpumps supply feedwater to the RPV until the feedpumps trip on an automatic signal (e.g., vessel high water level trip).

ii) After the feedpumps have tripped, the condensate (booster) pump(s) supplies feedwater to the RPV when the reactor pressure is below the condensate (booster) pump discharge head.

c. Feedwater will be supplied to the RPV, as described above, until the enthalpy of the feedwater is less than or equal to the enthalpy of the suppression pool water.

BASIS: Makeup to the RPV via the feedwater system is conservative for long term containment response. This is true when the enthalpy of the feedwater is greater than the enthalpy of water supplied by the ECCS.

The assumptions have been developed to take the above into account and to provide an outline for a conservative mechanistic treatment of the feedwater system. The assumptions do this by recognizing that a pump is capable of supplying feedwater to the RPV when the discharge head is above the reactor vessel pressure and that it is conservative to add feedwater when the feedwater enthalpy is greater than the enthalpy of the water in the alternate source (suppression pool).

6. <u>Offsite Power</u>

ASSUMPTION: Offsite power is assumed available for all cases.

BASIS: Assuming the availability of offsite power is required to maintain a consistent set of assumptions regarding feedwater addition to the RPV. The motor-driven feedwater pumps and condensate (booster) pumps cannot operate unless offsite power is available. Since feedwater addition is conservative, offsite power being available is conservative.

7. <u>Manual Scram</u>

ASSUMPTION: In accordance with the Technical Specifications, manual scram occurs at TS3, for SORV at power. Manual scram is accomplished in a single manipulation by transferring the mode switch from "run" to "shutdown."

BASIS: A SORV at power is a transient with no direct automatic scram signal generated. The operator will manually scram the reactor in keeping with the plant Technical Specifications.

The operator will have positive indication of an open relief valve (e.g., acoustic monitor, position switch, etc.). In addition, the plant pool temperature monitoring system will show a continuous pool temperature increase. Alarms will be activated at TS1 and TS3.

The operator will be trained for a SORV and know how to respond to it. The operator will scram the reactor at or before TS3 in accordance with the plant Technical Specifications and normal and emergency procedures. Therefore, pool temperature analysis is conservative by assuming the reactor will continue to operate at full power until TS3 is reached.

8. Scram On High Drywell Pressure

ASSUMPTION: Scram on high drywell pressure and MSIV closure signal assumed at t = 0, for SBA.

BASIS: One of the plant safety features is that scram occurs automatically on high drywell pressure. The high drywell pressure signal is generated before the vents are cleared and the pool temperature increase begins. The analysis method conservatively ignores holdup of steam or liquid in the drywell and drywell heat sinks for the pool temperature calculations. The break flow is vented directly to the pool starting at the time the break occurs. Therefore, it is conservative to assume that scram occurs at the same time as the pool temperature incre begins.

A spurious MSIV signal at t = 0 conservatively maximizes the heat retained in the RPV which must be transferred to the pool.

9. Main Condenser

ASSUMPTION: For the case with the initiating event of a SORV with the loss of one RHR HX, the main condenser remains available.

BASIS: The main condenser is a known preferential heat sink and its use is consistent with the emergency and standard operating procedures. It is a normal piece of equipment that the operator is familiar with and it will be used. During a SORV, normal plant systems such as the main condenser will remain available.

A simple schematic of the reactor, turbine, main condenser and associated valving and piping is shown in Figure C-1. It can be referred to for the following event description.

The plant is initially operating at full power with the suppression pool temperature just below TS1. A safety relief valve then spuriously opens and sticks open.

The operator will immediately have positive indication of a SORV and the pool temperature alarm at TS1 will sound.

The automatic controls of the plant respond to the reduced flow and adjust to a new operating condition. The reactor system will very quickly be in equilibrium again, except there will be a turbine/feedwater flow mismatch as approximately 7% of the reactor steam is flowing to the suppression pool.

The operator will take immediate actions to attempt to close the stuck open valve and scram the reactor. The operator scrams the reactor by taking the mode switch out of the "run" mode which bypasses the MSIV isolation signal. Subsequent action is taken to place the available RHR systems in pool cooling.

Following scram, the reactor steam generation will decrease so that the turbine control valves will close as the RPV pressure drops. Since the SORV depressurizes the reactor, the turbine bypass valves

will not open automatically. No MSIV closure signals* are mechanistically generated. Therefore, the feedwater system will continue to maintain RPV water level.

The operator begins to manually depressurize the RPV at TS4. The operator will open the turbine bypass valves and discharge reactor steam to the main condenser. The operator will place the RHR system in its shutdown cooling mode when the RHR system interlocks are cleared (e.g., low reactor pressure) and will then terminate use of the main condenser.

In conclusion, the above event sequence shows that the main condenser remains available for the SORV at power event.

* MSIV closure can be initiated by:
1. High steam line radiation
2. Low steam line pressure ("run" mode only)
3. High steam line flow
4. Low RPV water level
5. High steam tunnel temperature
6. High steam line delta temperature
7. Low condenser vacuum

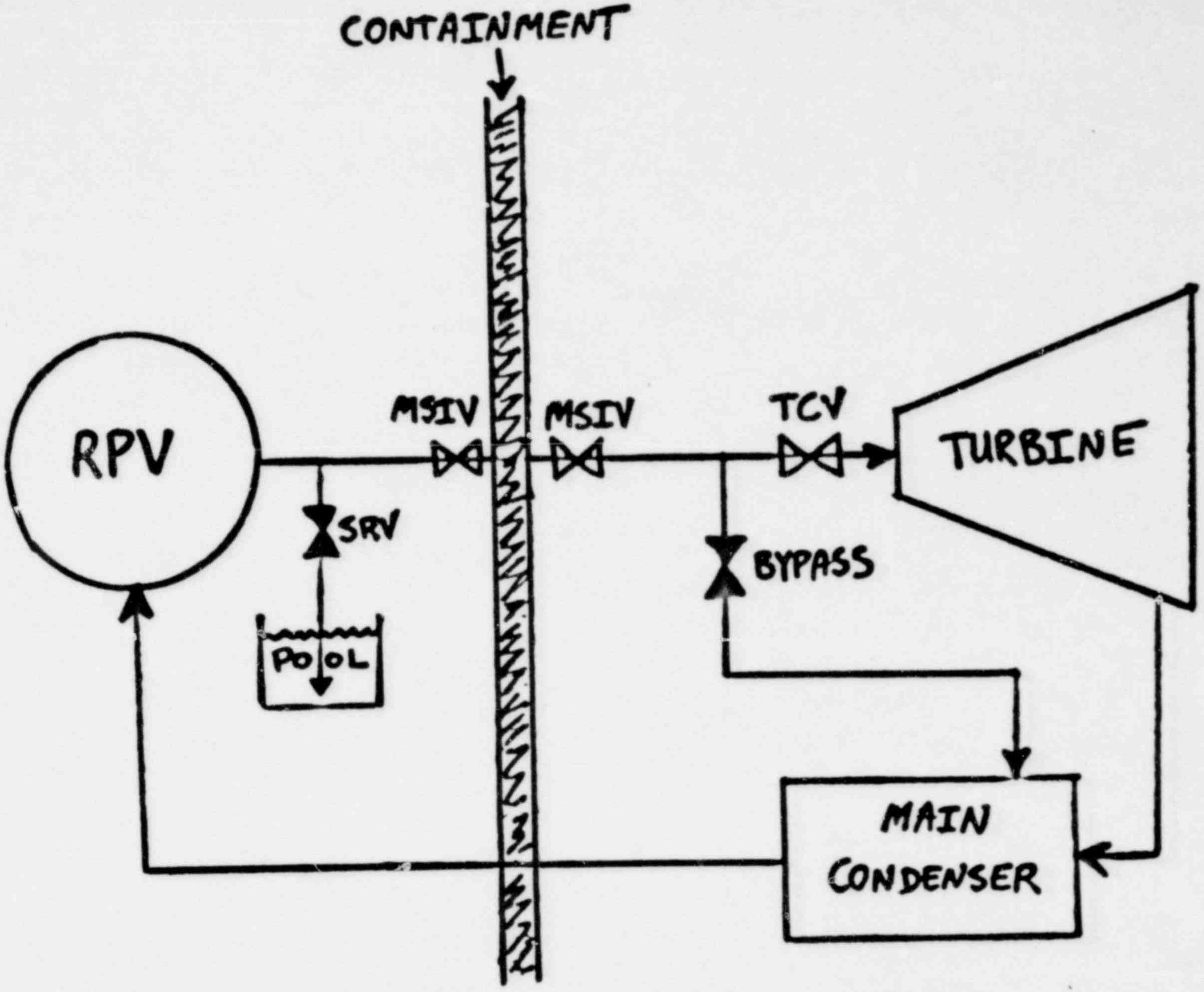

FIGURE C-1. Plant Schematic Diagram

APPENDIX D

SENSITIVITY STUDIES

This appendix is a summary of the suppression pool temperature sensitivity to the key parameters and assumptions used in the analysis.

A detailed set of sensitivity studies has been performed and is presented in Table D-1. The table includes the event selected for the sensitivity analysis, and tabulates the parameter, its base value, the sensitivity study value and the peak pool temperature difference (ΔT) from the base case peak pool temperature for that event. A negative ΔT is a decrease in the peak pool temperature while a positive ΔT is an increase.

The sensitivity of a parameter depends both upon the magnitude of the change in the base variable and the event for which it is analyzed.

While the sensitivity studies were performed for Zimmer, they are typical of the sensitivities of the Mark II plants.

TABLE D-1

SUMMARY OF THE PEAK POOL TEMPERATURE SENSITIVITY STUDIES

EVENT	PLANT PARAMETER	VARIATIONS	CHANGE IN SUPPRESSION POOL PEAK TEMPERATURE (°F)
3A	Service Water Temperature	95°F	BASE
		85°F	-1.3
		75°F	-2.6
		65°F	-3.8
1A	Manual Scram Time	5 minutes (when Tpool = 110°F)	BASE
		3 minutes	-5.4
		10 minutes	+11.7
1A	Time at which Main Condenser is available	20 minutes	BASE
		10 minutes	-13.9
		30 minutes	+5.0
3A	Time for Initiation of RHR Pool Cooling	10 minutes	BASE
		20 minutes	+0.2
		30 minutes	+0.6
3B	Time for Initiation of Manual Depressurization	20 minutes (when Tpool = 120°F)	BASE
		15 minutes	0.0
		30 minutes	+0.1
3B	Manual Depressurization Rate	100°F/hr	BASE
		200°F/hr	-2.4
		125°F/hr	-0.7
		75°F/hr	+0.1
3A	Initial Pool Water Mass	5.83×10^6 lbm/(LWL)	BASE
		5.97×10^6 lbm/(HWL)	-1.1

APPENDIX E

FSAR CHAPTER 15 COMPARISON

This appendix provides a comparison of the Mark II long term pool temperature events analyzed with the transients reported in Chapter 15 of the plant specific Final Safety Analysis Report (FSAR).

The Mark II long term pool temperature events have a higher peak pool temperature than all the events analyzed in FSAR Chapter 15.

Table E-1 presents a list of all FSAR Chapter 15 events. It shows which events will result in a pool temperature increase and which will result in no pool temperature increase. For events which have a pool temperature increase, the similar long term pool temperature event is also identified. Those events which have a "N/A" are for PWRs only.

A detailed comparison of the FSAR Chapter 15 events and the Mark II long term pool temperature events, both assumptions and results, was performed. The results of the comparison indicate that the similar containment events bound all FSAR events for the calculation of the long term suppression pool peak temperature. The predominant number of FSAR Chapter 15 events define short term reactor response to various transient events in which the reactor returns to the normal state without significant increase in pool temperature.

Typical reasons that long term pool temperature events have a higher long term peak temperature are as follows:

1. Only one RHR system is available.
2. Stored energy in the feedwater system is added to the reactor.
3. Energy dump to the main condenser is minimized due to spurious main steam line isolation.

TABLE E-1

FSAR CHAPTER 15 EVENTS VERSUS

LONG TERM POOL TEMPERATURE EVENTS

FSAR CHAPTER 15 EVENTS		SIMILAR LONG TERM POOL TEMPERATURE EVENTS: SORV AT POWER	ISOLATION/ SCRAM	SMALL BREAK ACCIDENT	NO POOL TEMPERATURE INCREASE
15.1.1	Loss of Feedwater Heating				X
2	Feedwater Controller Failure - Maximum Demand		X		
3	Pressure Regulator Failure - Open		X		
4	Inadvertent Safety Relief Valve Opening	X			
5	PWR Steam Piping Break				N/A
6	Inadvertent RHR Shutdown Cooling Operation				X
15.2.1	Pressure Regulator Failure - Closed				X
2	Generator Load Reject		X		
3	Turbine Trip		X		
4	MSIV Closures		X		
5	Loss of Condenser Vacuum		X		
6	Loss of AC Power		X		
7	Loss of Feedwater Flow		X		
8	Feedwater Line Break		X		
9	Failure of RHR Shutdown Cooling		X		
15.3.1	Recirculation Pump Trip		X		
2	Recirculation Flow Control Failure - Decreasing Flow		X		
3	Recirculation Pump Seizure		X		
4	Recirculation Pump Shaft Break		X		

TABLE E-1 (CONTINUED)

FSAR CHAPTER 15 EVENTS VERSUS

LONG TERM POOL TEMPERATURE EVENTS

		SIMILAR LONG TERM POOL TEMPERATURE EVENTS			
FSAR CHAPTER 15 EVENTS		SORV AT POWER	ISOLATION/ SCRAM	SMALL BREAK ACCIDENT	NO POOL TEMPERATURE INCREASE
15.4.1	Rod Withdrawal Error - Low Power				X
2	Rod Withdrawal Error - At Power				X
3	Control Rod Maloperation				X
4	Abnormal Startup of Idle Recirculation Pump				X
5	Recirculation Flow Control Failure with Increasing Flow				X
6	Chemical and Volume Control System Malfunctions				N/A
7	Misplaced Bundle Accident				X
8	Spectrum of Rod Ejection Assemblies				N/A
9	Control Rod Drop Accident				X
15.5.1	Inadvertent HPCI/HPCS Startup				X
2	Chemical Volume Control System Malfunction				N/A
15.6.1	Inadvertent Safety Relief Valve Opening	X			
2	Instrument Line Break				X
3	Steam Generator Tube Failure				N/A
4	Steam System Piping Break Outside Containment		X		
5	Loss-of-Coolant Accidents			X	
6	Feedwater Line Break Outside Containment		X		