

The major factor in assessing the safety margin of any of the SEP facilities depends upon the ability to provide adequate protection for postulated design basis events (DBEs). The SEP topics provide a major input to the DBE review, both from the standpoint of assessing the probability of certain events and that of determining the consequences of events. As examples, the safe shutdown topics pertain to the listed DBEs (the extent of applicability will be determined during the SEP DBE review for Ginna):

<u>Topic</u>	<u>DBE Group*</u>	<u>Impact Upon Probability Or Consequences of DBE</u>
V-10.B	VII (Spectrum of Loss-of-Coolant Accidents)	Consequences
V-11.A	VII (Defined above)	Probability
V-11.B	VII (Defined above)	Probability
VII-3	All (Defined as a generic topic)*	Consequences
IX-3	III (Steam Line Break Inside Containment) (Steam Line Break Outside Containment)	Consequences
	IV (Loss of AC Power to Station Auxiliaries) (Loss of all AC Power)	Consequences
	V (Loss of Forced Coolant Flow) (Primary Pump Rotor Seizure) (Primary Pump Shaft Break)	Probability
	VII (Defined above)	Consequences

* For a listing of DBE groups and generic topics, see Reference 10.

TABLE 3.1 (Continued)

Components/Subsystems	Quality Group		Seismic		Remarks
	R.G. 1.26	Plant Design	R.G. 1.29	Plant Design	
Pressurizer Relief Valves	ASME III Class 1	7	Category 1	Class 1	
Pressurizer Safety Valves	ASME III Class 1	ASME III	Category 1	Class 1	
Pressurizer Heaters	NA	--	Category 1	Class 1	
<u>Component Cooling Water (CCW)</u>					
CCW pumps (2)	ASME III Class 3	7	Category 1	Class 1	
CCW heat exchangers	ASME III Class 3	ASME VIII	Category 1	Class 1	
Surge Tank	ASME III Class 3		Category 1	Class 1	
CCW piping and valves	ASME III Class 3	USAS III.1 & nuclear code cases	Category 1	Class 1	
<u>Residual Heat Removal (RHR) System</u>					
RHR pumps (2)	ASME III Class 2	7	Category 1	Class 1	RHR pumps provide LPSI and ECCS containment recirculation

TABLE 3.1 (Continued)

Components/Subsystems	Quality Group		Subsidiary		Remarks
	R.G. 1.26	Plant Design	R.G. 1.29	Plant Design	
RRR heat exchangers (tube side)	ASME III Class 2	ASME III Class C	Category I	Class I	
(shell side)	ASME III Class 3	ASME VIII	Category I	Class I	
Piping and valves to RRR pump suction from RWST, con- tainment sump, valve 701, and CVCS	ASME III Class 2	USAS 031.1 & nuclear code cases	Category I	Class I	
Piping and valves from RRR pump discharge to valves 1012 A,B and via RRR heat exchangers to RCS (valves 052 A,B, 720), CVCS, Sampling System, RWST, RPS1 pump #1C, and recirculation line to RRR pumps	ASME III Class 2	USAS 031.1 & nuclear code cases	Category I	Class I	
<u>Process Instrumentation and Controls</u>	NA	--	Category I	Class I	for safe shutdown systems only; see section 3.3.
<u>Emergency Power Supply System</u>	NA	--	Category I	Class I	
Diesel generators			Category I	Class I	
DC power supply system			Category I	Class I	
Distribution lines, switchgear, control boards, motor control centers			Category I	Class I	

TABLE 3.2

FUNCTIONS FOR SHUTDOWN AND COOLDOWN

<u>Function</u>	<u>Method</u>
1. Control of Reactor Power	a. Doration 1. CVCS 2. High Pressure Safety Injection b. Control Rods 1. Controlled Rod Insertion 2. Reactor Trip
2. Core Heat Removal	a. Forced Circulation (reactor coolant pumps) b. Natural Circulation (using steam generators) c. Residual Heat Removal d. CVCS Shutdown Heat Exchangers (CCW) e. Pressurizer Reliefs and Safety Injection
3. Steam Generator Heat Removal	a. Main Condenser (circulating water system) b. Atmospheric Dumps (manual actuation) c. Safety Valves d. Auxiliary Feed System Turbine e. Steam Generator Blowdown f. Water-Solid Steam Generator
4. Feedwater	a. Main Feedwater Pumps b. Steam- and Motor-Driven Auxiliary Feedwater Pumps c. Standby Auxiliary Feedwater Pumps
5. Primary System Control	a. CVCS b. Pressurizer Relief Valves

TABLE 3.3 LIST OF SAFE SHUTDOWN INSTRUMENTS

<u>Component/System</u>	<u>Instrument</u>	<u>Instrument Location</u>	<u>Reference</u>
Main Steam	Steam generator level LI & LI 460, 461 and 470, 471	LT Inside Containment LI Control Room ^A	DWG. 33013-544, Refs. ¹¹ 11 and 15
	Steam Pressure PT & PI 460, 469, 470, 479	PI Intermediate Building PI Control Room	DWG. 33013-534
	Reactor Coolant	Pressurizer level LT & LI, 426, 427, 428, 433	LT Inside Containment LI Control Room ^A
	Pressurizer pressure PT & PI 449, 429, 430, 431	PI Inside Containment PI Control Room ^A	DWG. 33013-424, Refs. ¹² 6 and 15
	RCS temperature TE & TI 409 A&B and 410 A&B	TE Inside Containment TI Control Room	DWG. 33013-424, Refs. ¹¹ 6 and 15
Auxiliary Feed	AFWS flow FI 2001, 2002, 2023, 2024	FI Intermed. Build. FI Control Room ^A	DWG 33013-544, Refs. ¹¹ 8 and 15
	FI 2021, 2022, 2023, 2024		
	SAFS flow FI & FI 4004, 4005	FI Aux. Build. Addition FI Control Room ^A	DWG D-302-071-E, Refs. ¹² 5 and 15
Service Water	Pump discharge press. PI 2160 & 2161, PI 2160 & 2161	PI Screen House PI Control Room	DWG 33013-529
Chemical and Volume Control	Charging flow FIT 120, FI 120	FIT Auxiliary Build. FI Control Room	DWG 33013-433
	RWST level LT 920, LI 920	LT Auxiliary Building LI Control Room	DWG 33013-425.

*Also indicators are available at local shutdown panels

TABLE 3.3 LIST OF SAIL SHUTDOWN INSTRUMENTS

<u>Component/System</u>	<u>Instrument</u>	<u>Instrument Location</u>	<u>Reference</u>
Component Cooling Water	System flow	FI Auxiliary Build.	DWG 33013-436
	FI 619	Low flow alarm in control room	
	Surge tank level	FI Auxiliary Build.	DWG 33013-435
	FI 610	FI Control Room	
Residual Heat Removal	System flow	FI Auxiliary Build.	DWG 33013-436
	FI 626, FI 626	FI Control Room	
Diesel Generator	Generator output voltage and current	Control Room	
Emergency AC Power	400V Buses 14, 16, 17, 18, voltage indication	Control Room	
Emergency DC Power	125 VDC Buses 1 and 2 voltage indication	Control Room	



TABLE 3.4 SAFE SHUTDOWN SYSTEMS POWER SUPPLY AND LOCATION

<u>System</u>	<u>Power Supply</u>	<u>Location</u>
Reactor Protection, Reactor Breakers Reactor Isolables	DC power, Instrument buses	Control Room (209 ^o)
Main Steam Safety valves Isolation valves Atmos. Dump valves	----- air (fail closed), air or manual	Intermediate Build. (270 ^o) Intermediate Build. (270 ^o) Intermediate Build. (270 ^o)
Auxiliary Feed Motor driven pumps A, B Turbine driven pump Standby pumps C, D	A-Bus 14, B-Bus 16 Steam driven C-Bus 14, D-Bus 16	Intermediate Build. (253 ^o) Intermediate Build. (253 ^o) Aux. Build. Addition (270 ^o)
Service Water pumps A, B, C, D	A, C-Bus 10 B, D-Bus 17	Screen House (253 ^o) Screen House (253 ^o)
Chemical and Volume Control pumps A, B, C	A-Bus 14 B, C-Bus 16	Auxiliary Build. (235 ^o) east
refueling water storage tank	-----	Auxiliary Build.
Component Cooling Water pumps A, B heat exchangers	A-Bus 14, B-Bus 16 -----	Auxiliary Build. (271 ^o) Auxiliary Build. (271 ^o)
Residual Heat Removal pumps A, B heat exchangers	A-Bus 14, B-Bus 16 -----	Auxiliary Build. (219 ^o) RHR pit Auxiliary Build. (219 ^o)
Diesel Generators IA IB	125VDC Control Power 125VDC Control Power	Diesel room N side of turbine build. (253 ^o) Diesel room N side of turbine build. (253 ^o)

TABLE 3.4 SAFE SHUTDOWN SYSTEMS POWER SUPPLY AND LOCATION

400 V Bus 14	Diesel 1A or offsite power	Auxiliary Build. (271')
400 V Bus 16	Diesel 1B or offsite power	Auxiliary Build. (263')
400 V Bus 17	Diesel 1B or offsite power	Screen House (253')
400 V Bus 18	Diesel 1A or offsite power	Screen House (253')
Instrument Buses 1A, 1B, 1C, 1D	1A-Inverter 1, 1B-400V MCC 1C-Inverter 2, 1D-400V MCC	Control Room (209')
Battery and Inverter 1A	---	Battery room (253')
Battery and Inverter 1B	---	Battery room (253')

4.0 SPECIFIC RESIDUAL HEAT REMOVAL AND OTHER REQUIREMENTS OF BRANCH
TECHNICAL POSITION 5-1

BTP 5-1 contains the functional requirements discussed in Section 3.0 and also detailed requirements applicable to specific systems or areas of operation. Each of these specific requirements is presented below with a description of the applicable Ginna system or area of operation.

4.1 "B. RHR System Isolation Requirements

The RHR system shall satisfy the isolation requirements listed below.

1. The following shall be provided in the suction side of the RHR system to isolate it from the RCS:
 - (a) Isolation shall be provided by at least two power-operated valves in series. The valve positions shall be indicated in the control room.
 - (b) The valves shall have independent diverse interlocks to prevent the valves from being opened unless the RCS pressure is below the RHR system design pressure. Failure of a power supply shall not cause any valve to change position.
 - (c) The valves shall have independent diverse interlocks to protect against one or both valves being open during an RCS increase above the design pressure of the RHR system.
2. One of the following shall be provided on the discharge side of the RHR system to isolate it from the RCS:
 - (a) The valves, position indicators, and interlocks described in item 1(a)-(c).
 - (b) One or more check valves in series with a normally closed power-operated valve. The power-operated valve position shall be indicated in the control room. If the RHR system discharge line is used for an ECCS function the power-operated valve is to be opened upon receipt of a safety injection signal once the reactor coolant pressure has decreased below the ECCS design pressure.

(c) Three check valves in series, or

(d) Two check valves in series, provided that there are design provisions to permit periodic testing of the check valves for leaktightness and the testing is performed at least annually."

The RHR suction and discharge valves connecting this system to the primary coolant system are shown on Figure 3.3-1 of the R. E. Ginna FSAR. The reactor coolant system suction supply to the RHR pumps is from the hot leg of loop A through motor-operated valves MOV 700 and MOV 701 in series. The RHR pump discharge return to the loop B cold leg of the reactor coolant system is through two series motor-operated valves, MOV 720 and MOV 721. There are no check valves in series with MOV 720 and MOV 721.

Permissive interlocks required to open the four RHR system isolation valves are listed below.

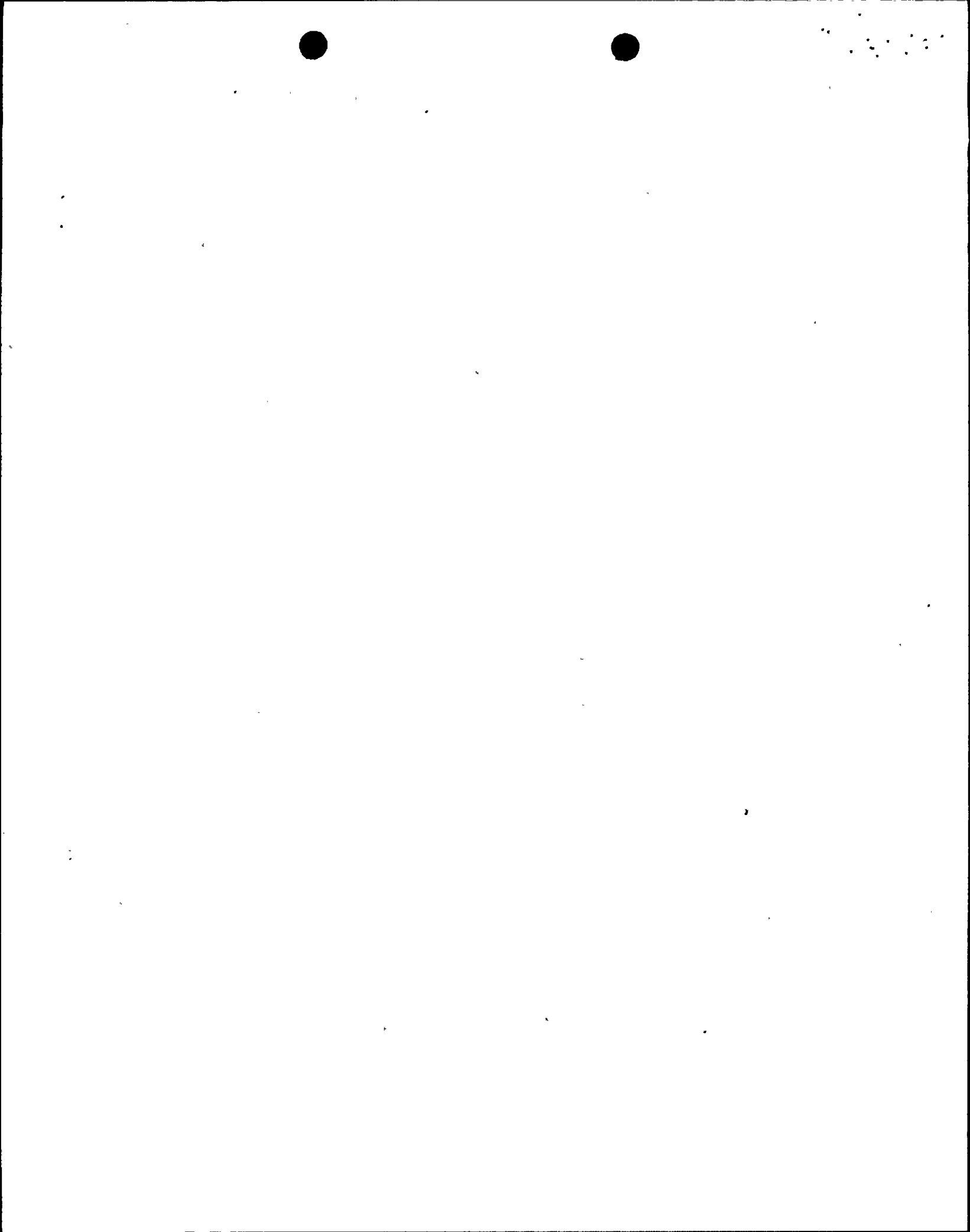
- | | |
|---------|---|
| Mov 700 | (1) Reactor coolant system pressure must be less than +10 psig |
| | (2) RHR suction valves MOV 350A and MOV 350B from the containment sump must be closed |
| Mov 701 | (1) RHR suction valves MOV 350A and MOV 350B from the containment sump must be closed |
| | (2) The valve is operated by a key switch |
| Mov 720 | (1) No interlocks exist but the valve is operated by a key switch |
| Mov 721 | (1) Reactor coolant system pressure must be less than +10 psig |

No interlocks are associated with valve closure. There are no automatic functions which close the valves and no alarms generated by the valves (Reference 5). The valves fail "as is" upon loss of power supply and have remote position indication in the control room.

The RHR system discharge line is not used for an ECCS function that would require MOV 720 or MOV 721 to open; however, a branch of the RHR discharge line provides low pressure safety injection (LPSI) to the reactor vessel via parallel lines with one normally closed motor-operated valve and one check valve in each line. The check valves are periodically tasted. The motor-operated valve position indication is provided in the control room and these valves receive an open signal coincident with the safety injection (SI) signal.

Based on the above description, the RHR system deviates from these BTP provisions:

- (a) The power-operated valves in the LPSI lines open on an SI signal before RCS pressure drops below RHR design pressure.
- (b) The RHR discharge and suction isolation valves do not have independent diverse interlocks to prevent opening the valves until RCS pressure is below 410 psig. Only the inboard valves (700, 721) have this interlock. The outboard valves (701, 720) are manually controlled with key-locked switches. By procedure, MOV 701 and MOV 720 are not opened until RCS pressure is less than 410 psig.



(c) The RHR isolation valves have no interlock feature to close them when RCS pressure increases above the design RHR pressure.

The staff has concluded that the deviation regarding the independent, diverse interlocks to prevent opening of the RHR isolation valves until pressure is below 410 psig is acceptable. The RHR isolation valves are designed such that they are physically unable to open against a differential pressure of greater than 500 psi. The inboard isolation valves are provided with a pressure interlock. By administrative procedure, the RHR valves are key-locked closed, with power removed. In addition, a relief valve (RV203), set at 600 psig, is available. The staff therefore has concluded that the probability of an intersystem LOCA is acceptably low.

The deviation regarding the LPSI isolation valve is considered acceptable since the check valve testing provides sufficient assurance that these valves will perform their isolation function until RCS pressure decreases below RHR pressure. The staff's position on these deviations is given in Section 5.2.

The deviation regarding lack of automatic closure for the RHR isolation valves is acceptable based on the administrative controls which the licensee provides for the operation of these valves, coupled with the RHR system high pressure alarm at 550 psig and the RCS interlock pressure alarm at 410 psig (Reference 5). These alarms provide adequate assurance that the operator action required by procedure will be taken to shut the

isolation valves when RCS pressure is increasing towards the RHR design pressure. (See the following discussion of BTP provision C.1, "Pressure Relief Requirements.")

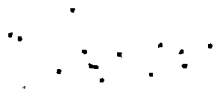
4.2 "C. Pressure Relief Requirements

The RHR system shall satisfy the pressure relief requirements listed below.

1. To protect the RHR system against accidental overpressurization when it is in operation (not isolated from the RCS), pressure relief in the RHR system shall be provided with relieving capacity in accordance with the ASME Boiler and Pressure Vessel Code. The most limiting pressure transient during the plant operating condition when the RHR system is not isolated from the RCS shall be considered when selecting the pressure relieving capacity of the RHR system. For example, during shutdown cooling in a PWR with no steam bubble in the pressurizer, inadvertent operation of an additional charging pump or inadvertent opening of an ECCS accumulator valve should be considered in selection of the design basis.

The RHR relief valve has a setpoint of 600 psig and a capacity of 70,000 lb/hr. The RHR system is provided with a 550 psig high pressure alarm and a reactor coolant system interlock pressure alarm at 410 psig. The RHR system is connected to the loop A hot leg on the suction side and the loop B cold leg on the discharge side. The design pressure and temperature of the RHRS are 600 psig and 400°F. The design basis with regard to overpressure protection for Ginna Station's RHRS is to prevent opening of the RHR isolation valves when RCS pressure exceeds 450 psig and to provide relief capacity sufficient to accommodate thermal expansion of water in the RHR and/or leakage past the system isolation valves.

An analysis of incidents which might lead to overpressurizing the RHR system was performed (Reference 5). Three events were considered in the analysis:



- (a) With RCS in solid condition and RHR and charging pumps operating, the letdown line from the RCS is isolated.
- (b) During cooldown using two RHR trains, one RHR train suffers a failure at a time when the core heat generation rate exceeds the heat removal capability of one train.
- (c) Pressurizer heaters are energized with RHR in operation and RCS solid.

The results of these analyses show that the RHR system is provided adequate relief capacity provided certain procedural changes are implemented. These changes have been implemented in the licensee's operating procedures.

Overpressure transients more severe than the three listed above have been analyzed by the licensee in conjunction with the reactor vessel overpressurization protection system (OPS) (Reference 4). To successfully mitigate these worst case transients, the licensee has modified the pressurizer power operated relief valve (PORVs) to provide a low pressure relief setpoint of 435 psig during plant cold shutdown conditions and has implemented several administrative controls changes. The PORVs also provide overpressure protection for the RHR system when the RHR is aligned to the RCS for shutdown cooling.

The staff has evaluated the effects of the worst case mass and heat input events to establish the capability of the OPS and RHR relief to prevent RHR overpressurization. For the mass input case presented in Reference 4, the OPS alone prevents pressure from exceeding the RHR design pressure. For the heat input case, the Reference 4 data was extrapolated to include a 50°F steam generator to RCS temperature difference at an RCS temperature of 300°F. (The data in Reference 4 only applied to heat input transients at RCS temperatures from 180°F to 250°F.) 300°F was chosen because, this is the maximum temperature for which the steam generator to RCS temperature difference is 50°F based on RHR initiation at 350°F. The staff determined that pressure transients, at an RCS temperature of 300°F which would result from heat addition, would not exceed 110% of RHR design pressure even assuming the failure of one PORV. No credit is taken for action of relief valve RV-203. The staff then considered the potential for initiating a heat input transient at Ginna when RCS temperature is between 300°F and 350°F. For a heat input transient to occur, the heat from the steam generators must be rapidly transferred to a cooler, water-solid RCS. The means of rapid heat transfer is forced convection caused by a reactor coolant pump start. In its review of overpressurization transients, the staff considered steam generator to RCS temperature differences in excess of 50°F to be unlikely occurrences.

The administrative measures proposed by the licensee to reduce the probability of heat input transient were to (1) require an acceptable RCS temperature profile prior to reactor coolant pump startup with a water-solid RCS, (2) require one coolant pump to be run until RCS temperature

is less than or equal to 150°F, and (3) minimize plant operation in a water-solid condition. Although items (1) and (3), above, would not necessarily preclude a heat addition event, item (2) would. Also, the staff examined the potential for initiating a heat input event during plant cooldown, which is the time that steam generator temperature may exceed RCS temperature with RCS temperature above 300°F. The licensee initiates RHR cooling at 350°F after cooling down to that point with the steam generators. Continuing the cooldown with the RHR system and with the reactor coolant pumps secured (in violation of procedures), would result in the 50°F difference being fully developed at an RCS temperature of 300°F. As noted before, a heat input event at this temperature would not result in RHR overpressurization even with an assumed single failure.

Based on the above discussion, we conclude that the OPS and RHR relief provide sufficient RHR overpressure protection for RCS temperatures of 300°F or less and that the licensee's procedures acceptably minimize the likelihood of a heat addition overpressure transient at RCS temperatures above 300°F. Therefore, the OPS and the RHR relief meet the pressure relief requirements of the RTP. The OPS and related Technical Specifications were approved by the staff in Reference 17.

By procedure, the OPS is enabled at the same time as RHR cooling is initiated during plant cooldown, so the RHR system is afforded the additional overpressure protection of the OPS. The licensee will be required to incorporate, into the plant Technical Specifications, a requirement

for enabling of the OPS whenever RHR cooling is in progress to assure this safety margin is maintained for the life of the plant. The licensee has agreed to incorporate this change (Reference 20).

- 4.2.1 "2. Fluid discharged through the RHR system pressure relief valves must be collected and contained such that a stuck open relief valve will not:
- "(a) Result in flooding of any safety-related equipment.
 - "(b) Reduce the capability of the ECCS below that needed to mitigate the consequences of a postulated LOCA.
 - "(c) Result in a nonisolatable situation in which the water provided to the RCS to maintain the core in a safe condition is discharged outside of the containment."

Fluid discharged through the 2-inch RHR relief valve (RV203) is directed to the pressure relief tank (PRT) inside the reactor containment. The PRT has a rupture disc which is designed to rupture at 100 psig and allow the contents of the tank to overflow to the containment sump, where it would be available for recirculation. Should flow from a stuck RHR relief valve cause the rupture disc to rupture, the consequences to safety-related equipment would be less severe than the consequences of post-LOCA containment flooding which has been previously analyzed and found acceptable (Reference 6).

If RV203 were to stick open in a post-LOCA scenario, RHR flow to the RCS for both low head recirculation and low head safety injection modes would be affected. This is because a flow path would exist from the RHR system to RV203 via valves HCV-133 and 703 in either of these RHR operating modes. HCV-133 fails shut following loss of instrument air on containment

isolation following a LOCA, but a flow path would still exist to RV203 via the 3/4-inch locked open manual valve 703. The effect of this flow diversion would not reduce the capability of the ECCS below that needed to mitigate the consequences of a postulated LOCA. This is because the design flow rate through RV203 (70,000 lb/hr, which is a conservative number in this case since HCV-133 is shut) is much less than the flow rate of an RHR pump in the low pressure safety injection (LPSI) mode (776,000 lb/hr). Each RHR pump has the capacity to provide 100% of the required LPSI flow. Therefore, the leakage through RV203 would not be as severe an event as the loss of an RHR pump which has been postulated as a single failure in the ECCS analysis.

- 4.2.2 "3. If interlocks are provided to automatically close the isolation valves when the RCS pressure exceeds the RHR system design pressure, adequate relief capacity shall be provided during the time period while the valves are closing."

As noted above, these interlocks are not provided. However, the procedures for coordination of the overpressure protection and RHR systems as described above provide adequate relief capacity to prevent the RCS pressure from exceeding RHR design pressure.

4.3 "D. Pump Protection Requirements

"The design and operating procedures of any RHR system shall have provisions to prevent damage to the RHR system pumps due to overheating, cavitation or loss of adequate pump suction fluid."

The features designed into the Ginna RHR system to prevent damage to the system centrifugal pumps are provision for pump cooling, a pump mini-flow

recirculation flow path, and system design to prevent loss of net positive suction head (NPSH).

The CCW system provides cooling for the RHR pumps to prevent damage from overheating. The RHR pumps are provided with a recirculation line to recycle a portion of the pump discharge fluid to the pump suction. This prevents overheating caused by operating the pumps under no flow conditions. NPSH calculations were performed for the RHR pumps by the licensee. The RHR operating modes evaluated were normal plant shutdown cooling, low pressure safety injection, and post-LOCA recirculation. Recirculation operation developed the most limiting NPSH requirements, but the calculations indicated a 43% NPSH margin is available during recirculation (Reference 7, page 6.2-37). The RHR NPSH requirements will be reevaluated during the SEP under Topic VI-7.E, "ECCS Sump Design and Test for Recirculation Mode Effectiveness."

The above protection features provide adequate protection to prevent RHR pump damage.

4.4 "E. Test Requirements

"The isolation valve operability and interlock circuits must be designed so as to permit online testing when operating in the RHR mode. Testability shall meet the requirements of IEEE Standard 338 and Regulatory Guide 1.22. The preoperational and initial startup test program shall be in conformance with Regulatory Guide 1.58. The programs for PWRs shall include tests with supporting analysis to (a) confirm that adequate mixing of borated water added prior to or during cooldown can be achieved under natural circulation conditions and permit estimation of the times required to achieve such mixing, and (b) confirm that the cooldown under natural circulation

conditions can be achieved within the limits specified in the emergency operating procedures. Comparison with performance of previously tested plants of similar design may be substituted for these tests."

The RHR isolation valve operability and interlocks cannot be tested during the RHR cooling mode of operation. This test requirement is not applicable to the Ginna facility, since the installed interlocks function only when the RHR isolation valves are shut.

Regulatory Guide 1.68 was not in existence when the Ginna preoperational and initial startup testing was accomplished. However, tests have been performed to confirm that cooldown under natural circulation can be achieved (Reference 8). The core flow rates achieved under natural circulation were more than adequate for decay heat removal. The calculated core flow at approximately 2% reactor power was 4.2% of nominal full power flow. At approximately 4% reactor power, calculated core flow was 5.2% of nominal. Flow rates of this magnitude should provide adequate mixing of boron added to the RCS during cooldown. An incident at Ginna Station on July 5, 1970, provides further indication that natural circulation will provide uniform mixing of boron in the RCS (Reference 9). During that incident, while steam system maintenance was in progress with no RCPs operating, natural circulation was indicated by incore thermocouple readings. While the RCPs were secured, 1365 gallons of water were added to the RCS to dilute the boron concentration. When an RCP was restarted, reactor power, which was being maintained at a low power level corresponding to 10^{-7} amps on the intermediate range channel, did not change. This

indicates that the natural circulation flow had uniformly mixed the boron throughout the RCS.

4.5 "F. Operational Procedures

"The operational procedures for bringing the plant from normal operating power to cold shutdown shall be in conformance with Regulatory Guide 1.33. For pressurized water reactors, the operational procedures shall include specific procedures and information required for cooldown under natural circulation conditions."

Operational procedures reviewed in this comparison of the Ginna Station to BTP RSB 5-1 are discussed in Section 2.0. All of the procedures required the use of nonsafety-grade equipment for portions of the shutdown operation. The licensee performed a review of a plant shutdown utilizing safety-grade equipment only; this procedure would require remote hand operation of certain air-operated valves because the control air system is not safety-grade. The procedures for shutdown and cooldown should provide instructions as to how safety-grade equipment could be used to perform the cooldown. No procedure exists for proceeding to cold shutdown conditions from outside the control room. The need for procedures for these evolutions stems from the provisions of BTP RSB 5-1 and SEP Topic VII-3 to provide assurance that the capability for decay heat removal with safety-grade equipment exists. The staff will consider requiring the licensee to develop these procedures during the integrated SEP assessment of the Ginna plant. We conclude that the procedures for safe shutdown and cooldown at Ginna are in conformance with Regulatory Guide 1.33. The plant operating procedures also include a procedure for cooldown using natural circulation.

4.6 "G. Auxiliary Feedwater Supply

"The seismic Category I water supply for the auxiliary feedwater system for a PWR shall have sufficient inventory to permit operation at hot shutdown for at least four hours, followed by cooldown to the conditions permitting operation of the RHR system. The inventory needed for cooldown shall be based on the longest cooldown time needed with either only onsite or only offsite power available with an assumed single failure."

The Category I water supply for the auxiliary feed system (AFS) is the service water system (SWS). The SWS, which must be manually aligned to the AFS system, receives its water supply from Lake Ontario via the seismic Class I screen house. This source of water, which has never been interrupted in the nine years of plant operation, provides sufficient AFS water supply with an assumed single failure regardless of the loss of offsite or onsite power:

The SEP will reexamine the adequacy of the screen house to provide water during emergency shutdown and maintenance of safe shutdown during resolution of SEP topics on seismic design and flooding.

The SEP has reevaluated the capability of the Ginna plant to achieve cold shutdown conditions within a reasonable period of time in Appendix A.

5.0 RESOLUTION OF SEP TOPICS

The SEP topics associated with safe shutdown have been identified in the INTRODUCTION to this assessment. The following is a discussion of how the Ginna Station meets the safety objectives of these topics.

5.1 Topic V-10.B RHR System Reliability

The safety objective for this topic is to ensure reliable plant shutdown capability using safety-grade equipment using the guidelines of SRP Section 5.4.7, Regulatory Guide 1.139, and BTP RSB 5-1. The Ginna Station systems have been compared with these criteria, and the results of these comparisons are discussed in Sections 3.0 and 4.0 of this assessment. Based on these discussions, we have concluded that the Ginna systems fulfill the topic safety objectives except for the requirement for procedures to shutdown and cooldown using safety-grade systems.

The licensee will be required to ensure that their operating procedures contain sufficient information to enable plant operators to perform required functions, such as decay heat removal, with safety-grade systems.

5.2 Topic V-11.A Requirements for Isolation of High and Low Pressure Systems

The safety objective of this topic is to assure adequate measures are taken to protect low pressure systems connected to the primary system

from being subjected to excessive pressure which could cause failures and in some cases potentially cause a LOCA outside of containment.

This topic is assessed in this report only with regard to the isolation requirements of the RHR system from the RCS. As discussed in Sections 4.1 and 4.2, adequate overpressure protection for the RHR system will exist when the plant technical specifications are modified to require enabling the overpressure protection system whenever RHR cooling is in progress. The licensee agreed to this change in a letter dated January 13, 1981.

5.3 Topic V-11.B RHR Interlock Requirements

The safety objective of this topic is identical to that of Topic V-11.A. The staff conclusion regarding the Ginna RHR interlocks, as discussed in Section 4.1, is that adequate interlocks exist subject to completion of the above modification.

