SYSTEM 80+

SHUTDOWN RISK EVALUATION REPORT

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(DRAFT)

ABB-COMBUSTION ENGINEERING
NUCLEAR POWER SYSTEMS
WINDSOR, CONNECTICUT

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ABSTRACT

In engineering the System 80+ Standard Plant Design, ABB recognized the significance of addressing safety during shutdown operations. System 80+ is engineered with features that enhance shutdown safety: 1) by deliberate system engineering, equipment specification and plant arrangements for shutdown operation, 2) by mode dependent control logic that assists and limits operations, 3) by instrumentation, displays and alarms that clearly portray plant status in each mode and 4) by thorough procedural guidance and Technical Specifications that address important shutdown evolutions. This report presents these features and evaluates them in the context of the specific shutdown issues identified by the NRC. The report fulfills the ABB commitments to the NRC to 1) provide shutdown information in support of the System 80+ Design Certification and 2) provide responses to specific RAI's on shutdown operations.

DEFINITIONS

The following definitions of terms are employed throughout this report.

[This section will be provided in the June 15, 1992 updated submittal of this report.]

TABLE OF CONTENTS

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

TABLE OF CONTENTS (Cont'd)

LIST OF FIGURES

LIST OF TABLES

1.0 INTRODUCTION

1.1 PURPOSE

This report presents features of the System 80+ design which address the issues of shutdown risk. It further evaluates these features with respect to their ability to reduce and/or mitigate the consequences of this risk. It fulfills the commitment made to the NRC by ABB in Reference 1 to submit shutdown risk information in support of the System 80+ Design Certification.

1.2 SCOPE

Sections 2.1 through 2.13 present detailed discussions on the specific shutdown issues. Following the detailed discussions of these shutdown risk issues, the report provides a probabilistic risk assessment in Section 3.0. This is followed in Sections 4.0, 5.0 and 6.0 by an evaluation of the applicability of the analyses in CESSAR-DC Chapters 6 and 15 to LOCA and accident events that are initiated from shutdown modes. Section 7.0 evaluates the features of System 80+ that simplify shutdown operations and thereby reduce the potential for initiating shutdown events. Conclusions of this report are provided in Section 8.0. The scope of the information presented was discussed with the NRC at a presentation by ABB in Rockville, Maryland on December 18, 1991 and is outlined by the ABB slides enclosed with the NRC minutes of the meeting in Reference 2.

The report also addresses the RAI's from the NRC staff on CESSAR-DC that pertain to shutdown risk. Appendix A of this report lists the RAI's and provides either the response or a referral to sections of the report which encompass the response to each RAI.

1.3 BACKGROUND

In Generic Letter No. 88-17 (Reference 4) the NRC issued recommendations to all holders of licenses for PWR's to implement certain "expeditious actions" before operating their plants in a reduced inventory condition and to implement, as soon as practical, "program enhancements" concerning operations during shutdown cooling. The objective was to prevent the reoccurrence of events that had occurred and that had the potential for core damage and/or release of radiation. In NUREG-1449, NRC staff evaluations of shutdown operations indicate that recommendations have been implemented and/or are underway at operating plants.

ABB reviewed NRC evaluations and recommendations and searched the events which NRC considered significant. Generic characteristics of these events were compared with features of System 80+ that can prevent or mitigate the events. In this way ABB has demonstrated that the System 80+ design satisfies the intent of the ALWR program to benefit from past PWR experience and that the specific NRC concerns have been addressed in the design.

1.4 SYSTEM 80+ FEATURES

In this section, a comparison is made between the characteristics of past events and the System 80+ design features. The categories of shutdown events at operating plants are those used by the NRC in Chapter 2 of NUREG-1449 with little modification. These categories are mostly the same as the issues identified by Secy-91-283 and presented by ABB at the December 18, 1991 meeting with the NRC (Reference 2). Each category encompasses a group of similar events that have in common the type of event initiator. Ultimately, if left unmitigated by automatic or manual actions, all events might eventually lead to over heating and/or physical damage to fuel with consequent radiation release, but each scenario sequence may differ. Depending upon the importance pla:ed on each step in a sequence, the same events could be grouped differently. For example, the NUREG-1449 category, "Loss of Shutdown Cooling", includes the issues listed in Reference 2 as 1) Mid-Loop Operation, 2) Loss of Decay Heat Removal Capability and 3) Effect of PWR Upper Internals.

The categories employed in this section to group past events encompass (and for some categories are identical to) the issues which are listed by Reference 2 and which are presented in detail in this report with a few exceptions. The exceptions apply to postulated LOCA events initiated at high pressure and other significant events initiated at high pressure for which we do not have actual experience because they have not occurred in operating plants. They exist only as analyses for use as guidance to avoid the physical event and therefore are not included in the categories of past events.

Past events are grouped into the following ten categories:

- Loss of shutdown cooling
- Loss of electrical power
- Loss of reactor coolant
- Containment integrity
- Overpressurization
- Flooding and spills
- Boron and reactivity events
- Fire protection
- Heavy loads and fuel handling
- Mode change events

For each past event placed into a category an initiator is identified. The plant design objective is to prevent the occurrence of the event initiator, but realistically, absolute prevention is impracticable and may be impossible. A combination of prevention and mitigation is employed in the System 80+ design.

Table 1-1 provides an overview of the System 80+ features that avoid core damage during shutdown operating modes. It lists the ten shutdown event categories and for each category it lists event initiators for past events. These initiators are presented in a generic fashion; each initiator representing many specific events that have occurred. For each initiator, the features of the System 80+ design that are available to prevent occurrence of an initiator and/or to mitigate the consequences of an initiator are listed.

Table 1-2 provides a list of specific past events initiated from shutdown modes. Events were selected to include all ten event categories and all types of event initiators, but not all similar significant events that have occurred. Several information sources were utilized to compile this list. They include events listed in NUREG-1449 which were taken from the 1990 AEOD report (Reference 5) and which occurred mostly between January 1988 and July 1990 with some additional events. For events since July 1990, ABB searched the INPO database for LER's using a selection of keywords pertinent to the ten event categories and to shutdown operation. Various other INPO and NSAC documents were also reviewed for significant event reports dating from 1976 to 1990.

Events in Table 1-2 are grouped into the ten categories given above. For each specific event, the features of the System 80+ design that apply to prevent and mitigate the event are indicated. A review of this table serves as a design review of the System 80+ capabilities to avoid core damage and/or significant radiation release during shutdown modes. The design features are discussed in more detail in the following Sections of this report.

TABLE 1-1

SHUTDOWN EVENT CATEGORIES AND SYSTEM 80+ FEATURES FOR PREVENTION, DETECTION AND MITIGATION

EVENT CATEGORY	EVENT INITIATOR		SYSTEM 80+ FEATURES FOR PREVENTION, DETECTION AND MITIGATION
1.) Loss of Shutdown Cooling	SCS flow loss by pump suction vortex.	A)	Mid-loop lev:l maximized by locating SCS suction piping at the bottom of the hot leg.
		B)	Hard piped venting for SCS pumps relieves gas binding more quickly and conveniently.
		C)	One SCS suction line from each hot leg provides SCS redundancy with separation of pump suction sourcer.
		D)	Containment spray pumps identical to SCS pumps provide redundant capacity and may take suction from IRWST to refill RCS and to mitigate gas binding.
	Inaccurate mid-loop level leading to suction vortex.	E)	With head on, reactor vessel level monitoring system level indications from vessel head to a level below that required for SCS operation. Level indication is accurate for intended use.
		F)	Core exit thermocouples monitor coolant temperature down to 100°F prior to withdrawal of CETs during fuel shuffling. The RTDs and SCS temperatures are accurate during SCS operation.
		G)	With head off, level indication near hot leg elevation is provided by high resolution instruments.
		H)	SCS performance monitored on each of 2 SCS pumps by pump motor current, flow rate, discharge pressure and suctio.ı pressure. Possible SCS flow variance with decay heat to minimize potential for vortexing during mid-loop.
	Loss of flow while head off, upper internals in vessel and cavity flooded leads to core heatup.	I)	Internals design limits coolant flow from cavity to core. High availability of SCS system and/or backups assures forced convection.
	Various low level and loss of RHR events.	J)	Non-shared SCS system allows SCS maintenance and testing during Modes 1-4 prior to cold shutdown, increasing availability in Modes 5 and 6.
		K)	All SCS valves are motor operated, preventing failures on loss of air if electro-pneumatic operators were used.
		L)	Shutdown specific tech specs and procedural guidance reduces likelihood of personnel errors.

TABLE 1-1 (Continued)

SHUTDOWN EVENT CATEGORIES AND SYSTEM 80+ FEATURES FOR PREVENTION, DETECTION AND MITIGATION

EVENT CATEGORY	EVENT INITIATOR	SYSTEM 80+ FEATURES FOR PREVENTION, DETECTION AND MITIGATION
1.) Loss of Shutdown Cooling (Continued)		M) Inadvertent errors are reduced and early operator evaluation of failures is improved by 1.) IPSO overview display with critical function and system status specific to shutdown modes, 2.) CRT displays with system lineups and component status and 3.) alarms that are dependent on plant mode and equipment status.
		N) Prevention of inappropriate automatic actions from personnel errors by shutdown specific control logic (e.g., remove autoclosure interlocks from SCS suction valves.)
		O) CCW availability is increased by 2 redundant Divisions, each with two pumps and heat exchangers.
		P) Service water availability is increased by 2 Redundant Divisions, each with two pumps.
		Q) Each SCS Division has four potential sources of AC power for increased availability.
2.) Loss of Electric Power	Equipment failure and/or inadvertent personnel error leading to loss of power and shutdown cooling	A) Alternate AC gas turbine provides third on-site power source.
		B) Two switchyard interfaces provide flexibility.
		C) Shutdown specific tech specs and procedural guidance reduce likelihood of personnel errors.
		D) Normal power from safety transformer to safety buses does not depend on any non-safety components.
		E) Each safety division has a dedicated diesel generator.
		F) No equipment is shared between diesels.
		G) No equipment is shared with another unit.
3.) Loss of Reactor Coolant	From shutdown mode, equipment failure and/or personnel error leads to loss of coolant, usually through systems connected to RCS.	A) Inadvertent errors are reduced and early operator evaluation of failures is improved by 1.) IPSO overview display with critical function and system status specific to shutdown modes 2.) CRT displays with system lineups and component status and 3.) alarms that are dependent on plant mode and equipment status.

TABLE 1-1 (Continued)

SHUTDOWN EVENT CATEGORIES AND SYSTEM 80+ FEATURES FOR PREVENTION, DETECTION AND MITIGATION

EVENT CATEGORY	EVENT INITIATOR	SYSTEM 80+ FEATURES FOR PREVENTION, DETECTION AND MITIGATION
3.) Loss of Reactor Coolant (Continued)		
	Inadvertent RPV pressurization while connected systems are open causing coolant level drop in vessel.	B) Mid loop vent will not allow significant RV head pressurization. Thus, instruments are not affected.
		C) In-core instrument seal table evolutions are prohibited by procedural guidance while vessel head is on and mid-loop evolutions are in progress, preventing seal leaks.
		D) Coolant loss via RCP during seal maintenance reduced by pump impeller weight creating seal.
	Cavity draining exposes fuel being transferred.	E) Cavity draining limited by reinforced pool seal between vessel flange and cavity floor.
		F) Containment layout prevents total draining if seal fails.
4.) Containment Integrity	Loss of shutdown cooling and/or loss of reactor coolant results in core boiling requiring rapid containment closure to prevent radiological release.	A) Tech spec requires hatch and all penetrations closed during mid-loop evolutions. All temporary lines pass through penetrations, not through the open hatch. Containment configuration and size allow more outage activities within containment, resulting in less time without containment integrity.
		B) Redundancy in SCS system, electric power supply and support systems together with increased instrumentation reduce likelihood of an initiating event progressing to boiling.
	Personnel errors result in opening pathways from containment to atmosphere during shutdown evolutions.	C) Shutdown specific tech specs and procedural guidance reduce likelihood of personnel errors.
5. Overpressurization	Inadvertent high pressure safety injection actuation at low temperature pressurizes RCS and SCS system.	A) SCS system relief valves sized for maximum safety injection liquid flow.
		B) RCS is vented through two vents during Modes 5 and 6.

TABLE 1-1 (Continued)

SHUTDOWN EVENT CATEGORIES AND SYSTEM 80+ FEATURES FOR PREVENTION, DETECTION AND MITIGATION

	EVENT CATEGORY	EVENT INITIATOR	SYSTEM 80+ FEATURES FOR PREVENTION, DETECTION AND MITIGATION
5.)	Overpressurization (Continued)		C) Ring forged reactor vessel beltline and vessel material provide additional margin to pressurized thermal shock.
6.)	Flooding and Spills	Uncontrolled coolant flow from opened systems, typically combined with other inadvertent and/or poorly planned evolutions, floods essential equipment.	A) Inadvertent errors are reduced and early operator evaluation of failures is improved by 1.) IPSO overview display with critical function and system status specific to shutdown modes 2.) CRT displays with system lineups and component status and 3.) alarms that are dependent on plant mode and equipment status.
			B) Shutdown specific tech specs and procedural guidance reduce likelihood of personnel errors.
			C) Plant layout, including separation of redundant divisions, limits damage that may occur to affected division. No communication between divisions, including piping, electrical, HVAC, floor drains, etc.
7.)	Boron and Reactivity Events	Various CVCS misoperations and uncalibrated source range neutron monitors cause approach to criticality.	A) Shutdown specific tech specs and procedural guidance reduce likelihood of improper operation.
		CVCS misoperation causes boron dilution or potential boron precipitation.	B) Precipitation prevented by design that limits boron concentration to below cold precipitation concentration in most borated coolant lines, eliminating need for most heat tracing.
			C) Boron dilution alarm provides advanced warning.
8.)	Fire Protection	During shutdown evolutions, use of combustible materials plus ignition sources such as temporary power lines increases potential for fire damage to essential systems.	A) Plant layout and fire barriers separate redundant divisions and systems to limit potential fire damage.
			B) Combustible materials are limited in specific fire control areas.
9.)	Heavy Loads and Fuel Handling	Inadequate design and/or surveillance of lifting devices causes potential damage to fuel or essential equipment.	A) Shutdown specific guidance limits pathways for heavy lifts.

TABLE 1-1 (Continued)

SHUTDOWN EVENT CATEGORIES AND SYSTEM 80+ FEATURES FOR PREVENTION, DETECTION AND MITIGATION

	EVENT CATEGORY	EVENT INITIATOR		SYSTEM 80+ FEATURES FOR PREVENTION, DETECTION AND MITIGATION
9.)	Heavy Loads and Fuel Handling (Continued)		B)	Plant arrangement minimizes potential for damaging drops.
			C)	Proven design for fuel, core arrangement and fuel handling machine minimizes potential fuel drop.
10.)	Mode Change Events	Operator and/or procedural errors allow mode changes without satisfying entry requirements.	A)	Shutdown specific tech specs and procedural guidance reduce likelihood of personnel errors.
			B)	Inadvertent errors are reduced and early operator evaluation of failures is improved by 1.) IPSO overview display with critical function and system status specific to shutdown modes 2.) CRT displays with system lineups and component status and 3.) alarms that are dependent on plant mode and equipment status.

TABLE 1-2

SHUTDOWN EVENTS AND SYSTEM 80+
PREVENTION, DETECTION, AND MITIGATION FEATURES

LER. NO.	PLANT NAME	EVENT CATEGORY	DATE	EVENT SUMMARY	SYSTEM 80+ PREVENTION, DETECTION, AND MITIGATING FEATURES (SEE TABLE 1-1)
NOTE (1)	MILLSTONE 2	LOSS OF SHUTDOWN COOLING	12/09/81	IN MODE 5, SDC PUMP TRIPPED DUE TO TESTING AND RCS TEMPERATURE ROSE 118 DEG F TO 208 DEG F IN ABOUT 20 MINUTES. HEATUP AND COOLDOWN LIMITS WERE EXCEEDED. MODE 4 WAS ENTERED WITHOUT SATISFYING MODE 4 LCO'S.	1H, 1J, 1M
NOTE (1)	SALEM 1	LOSS OF SHUTDOWN COOLING	03/16/82	IN MODE 5, VITAL BUS WAS DE-ENERGIZED CAUSING LOSS OF POWER TO 1 CCW PUMP AND 2 SERVICE WATER PUMPS. THE OTHER CCS AND SERVICE WATER PUMPS WERE OUT OF SERVICE FOR MAINTENANCE. THUS, A COMPLETE LOSS OF CCW AND SERVICE WATER RESULTED. THE IMMEDIATE EFFECT WAS TO CAUSE ALL CHARGING PUMPS, BORON INJECTION PATHS, RHR TRAINS, AND D/G'S TO BE INOPERABLE. VITAL BUS WAS RESTORED WITHIN 1 HR RESTORING SW. CCW RESTORED 2 HRS LATER.	1M, 1O, 1P, 1Q, 2G
395/82004	V.C. SUMMER	LOSS OF SHUTDOWN COOLING	09/15/82	DURING COLD SHUTDOWN WHILE TESTING OVERPRESSURE PROTECTION SYSTEMS FOR RHR PROTECTION, THE INTERLOCKS CAUSED THE RHR SUCTION ISOLATION VALVE TO INADVERTENTLY CLOSE BECAUSE OF INADEQUATE TEST PROCEDURE, RESULTING IN LOSS OF RHR FLOW.	1C, 1N
334/83020	BEAVER VALLEY 1	LOSS OF SHUTDOWN COOLING	06/29/83	INADEQUATE COMMUNICATION AND WORK PROCEDURES DURING REFUELING OUTAGE CAUSED LOSS OF ELECTRICAL POWER TO THE RUNNING RHR PUMP.	1L, 1M, 1Q
NOTE (1)	PALISADES	LOSS OF SHUTDOWN COOLING	01/08/84	DE-FUELED, WITH BOTH THE MAIN TRANSFORMER AND ONE OF TWO D/G'S OUT OF SERVICE, A PROBLEM DEVELOPED ON AN INCOMING ELECTRICAL FEEDER. SINCE ONE D/G WAS OUT OF SERVICE, THE PLANT SHOULD NOT HAVE DISCONNECTED FROM THE PROBLEM FEEDER. HOWEVER, THE DISCONNECT WAS MADE AND THE PLANT WAS POWERED BY ONE D/G. WHEN THIS WAS DONE, ALL SW WAS LOST BECAUSE ONE SW PUMP WAS OUT OF SERVICE AND THE OTHER 2 SW PUMPS WERE POWERED BY THE INOPERABLE D/G. LACK OF SW CAUSED LOSS OF RUNNING D/G AND LOSS OF ALL AC.	1M, 2A, 2B, 2D, 2E
316/84014	D.C. COOK 2	LOSS OF SHUTDOWN COOLING	05/21/84	INADEQUATE PROCEDURE ALLOWED STARTING SECOND RHR PUMP BEFORE STOPPING THE RUNNING PUMP, LEADING TO VORTEXING AND AIR BINDING OF RHR PUMPS.	1A, 1B, 1C, 1D, 1H, 1M
295/84031	ZION 1	LOSS OF SHUTDOWN COOLING	09/14/84	LOSS OF RHR OCCURRENCE WHILE WATER LEVEL WAS BEING LOWERED. PERSONNEL ERROR WITH NITROGEN PURGE GAS VALVE CAUSED INACCURATE LEVEL INDICATION LEADING	1A, 1B, 1C, 1D, 1E, 1H, 1M

(1) NO LICENSEE EVENT REPORT (LER) NUMBER AVAILABLE. THIS EVENT SUMMARY WAS OBTAINED FROM ANOTHER SOURCE OF INFORMATION.

TABLE 1-2

SHUTDOWN EVENTS AND SYSTEM 80+
PREVENTION, DETECTION, AND MITIGATION FEATURES

LER. NO.	PLANT NAME	EVENT CATEGORY	DATE	EVENT SUMMARY	SYSTEM 80+ PREVENTION, DETECTION, AND MITIGATING FEATURES (SEE TABLE 1-1)
				TO CAVITATION IN RHR PUMP.	
413/85020	CATAWBA 1	LOSS OF SHUTDOWN COOLING	04/22/85	INACCURATE LEVEL INDICATION DURING DRAINING IN MODE 5 CAUSED LOSS OF RHR PUMP SUCTION. PERSONNEL ERROR ALLOWED SECOND RHR TRAIN TO BE DOWN FOR MAINTENANCE WHILE DRAINING WAS IN PROGRESS. CHARGING RESTORED LEVEL, THE OPERABLE PUMP WAS VENTED AND RHR RESTORED.	1A, 1B, 1C, 1D, 1E, 1H, 1M
304/85028	ZION 2	LOSS OF SHUTDOWN COOLING	12/14/85	SHUTDOWN COOLING WAS LOST WHILE THE PLANT WAS IN COLD SHUTDOWN. IT WAS ATTRIBUTED TO VORTEXING. THE CAUSE OF THE EVENT WAS IDENTIFIED TO BE INADEQUATE PROCEDURE COUPLED WITH LEVEL MEASUREMENT AND THE LACK OF KNOWLEDGE ABOUT WHAT CONDITIONS, OR AT WHAT POINT, VORTEXING CAN OCCUR.	1A, 1B, 1C, 1D, 1E, 1G, 1H, 1M
302/86003	CRYSTAL RIVER 3	LOSS OF SHUTDOWN COOLING	02/02/86	AT MID-LOOP, RHR PUMP SHAFT BROKE AFTER CONTINUOUS OPERATION FOR ABOUT 30 DAYS. PLACING SECOND RHR PUMP IN OPERATION WAS DELAYED BECAUSE A TRIPPED BREAKER WAS POWERING THE SUCTION VALVE. TEMPERATURE ROSE 33 DEG F TO 131 DEG F.	1C, 1D, 1H, 1M
NOTE (1)	WATERFORD	LOSS OF SHUTDOWN COOLING	07/14/86	NEAR MID-LOOP, A DRAIN PATH WAS NOT CLOSED AND LEVEL DROPPED. OPERATING RHR PUMP BEGAN TO CAVITATE AND WAS SHUTDOWN. RHR LOST SEVERAL TIMES OVER A 3 AND 1/2 HOUR PERIOD. RCS TEMPERATURE ROSE 94 DEG F TO 232 DEG F.	1A, 1B, 1C, 1D, 1E, 1G, 1H, 1M
323/87005	DIABLO CANYON 2	LOSS OF SHUTDOWN COOLING	04/10/87	AT MID-LOOP, 7 DAYS AFTER SHUTDOWN WITH HEAD VENTED, A LEAK CAUSED LOSS OF RCS LEVEL AND THE RHR PUMP BECAME AIR BOUND. RHR WAS LOST FOR 1 HOUR, 28 MINUTES; RCS TEMPERATURE ROSE 87 DEG F TO 220 DEG F AND PRESSURE INCREASED TO 10 PSIG.	1A, 1B, 1C, [illegible], 1H, 1M
287/88005	OCONEE 3	LOSS OF SHUTDOWN COOLING	09/11/88	LOSS OF SHUTDOWN COOLING OCCURRED WHILE IN MODE 5 WHEN NON-LICENSED OPERATOR BECAME CONFUSED WHILE RACKING BREAKERS THEREBY RESULTING IN A LOSS OF POWER TO THE LOW PRESSURE INJECTION PUMPS. COOLANT TEMPERATURE INCREASED 15 DEGREES.	1D, 1J, 1L, 2M
454/88007	BYRON 1	LOSS OF SHUTDOWN COOLING	09/19/88	SHUTDOWN COOLING WAS LOST WHILE THE PLANT WAS IN THE REFUELING MODE. IT WAS ATTRIBUTED TO RHR PUMP CAVITATION CAUSED BY ENTRAINMENT OF AIR IN THE PUMP SUCTION DUE TO VORTEXING WHICH ADMITTED AIR WHEN THE REACTOR VESSEL WATER LEVEL LOWERED BELOW THE TOP OF THE REACTOR COOLANT HOT LEGS. WATER	1A, 1B, 1C, 1D, 1G, 1H, 1M

(1) NO LICENSEE EVENT REPORT (LER) NUMBER AVAILABLE. THIS EVENT SUMMARY WAS OBTAINED FROM ANOTHER SOURCE OF INFORMATION.

TABLE 1-2

SHUTDOWN EVENTS AND SYSTEM 80+
PREVENTION, DETECTION, AND MITIGATION FEATURES

LER. NO.	PLANT NAME	EVENT CATEGORY	DATE	EVENT SUMMARY	SYSTEM 80+ PREVENTION, DETECTION, AND MITIGATING FEATURES (SEE TABLE 1-1)
				LEVEL WAS MONITORED BY VISUALLY OBSERVING LEVEL IN REFUELING CAVITY, WHICH WAS 4 INCHES ABOVE THE FLANGE WITH THE UPPER INTERNALS IN PLACE WHEN CAVITATION OCCURRED.	
313/88014	ARKANSAS NUC ONE 1	LOSS OF SHUTDOWN COOLING	10/26/88	PERSONNEL ERROR CAUSED LOSS OF POWER TO THE DECAY HEAT REMOVAL COOLER OUTLET VALVES RESULTING IN LOSS OF SHUTDOWN COOLING. WITH LOSS OF POWER, THE VALVES, WHICH WERE DESIGNED TO "FAIL OPEN", WENT TO THE CLOSED POSITION. INVESTIGATION REVEALED THAT THE VALVES' ELECTRIC PNEUMATIC POSITIONER OUTPUT LINES HAD BEEN REVERSED CAUSING THE VALVES TO "FAIL CLOSED" ON LOSS OF POWER. COOLANT TEMPERATURE INCREASED 18 DEGREES.	1C, 1M
369/88049	MCGUIRE 1	LOSS OF SHUTDOWN COOLING	11/23/88	IN MODE 6, RHR PUMP LOST ITS SUCTION PRESSURE BECAUSE IT BECAME AIR BOUND WHILE STROKE TESTING VALVE IN SPRAY SYSTEM AT SUMP SUCTION, ALLOWING AIR TRAPPED IN INADEQUATELY VENTED HORIZONTAL PIPING TO BE FORCED INTO RHR SYSTEM. THE PUMP WAS MANUALLY STOPPED CAUSING A LOSS OF RHR. THE REACTOR COOLANT SYSTEM TEMP INCREASED FROM 90 DEG F TO 116 DEG F IN 39 MINUTES.	1B, 1C, 1D, 1H, 1J, 1M, 1N
313/88024	ARKANSAS 1	LOSS OF SHUTDOWN COOLING	12/19/88	RELAY PROBLEM CAUSED RHR SUCTION VALVE IN SINGLE SUCTION LINE TO CLOSE. RHR LOST FOR 12 MINUTES; RCS TEMPERATURE ROSE 12 DEG F TO 147 DEG F.	1B, 1C, 1D, 1H, 1M
457/89001	BRAIDWOOD 2	LOSS OF SHUTDOWN COOLING	02/23/89	DURING AN ESF RESPONSE TIME SURVEILLANCE, INADEQUATE PROCEDURE CAUSED THE RHR PUMP ISOLATION VALVE TO INADVERTENTLY CLOSE RESULTING IN LOSS OF RHR.	1C, 1D, 1H, 1M, 1N
317/89007	CALVERT CLIFFS 1	LOSS OF SHUTDOWN COOLING	03/17/89	DAMAGE TO SCS PIPING AND SUPPORTS RESULTING FROM THE SLAMMING SHUT OF CHECK VALVES WHEN ONE SCS/LPSI PUMP IS STOPPED WHILE ANOTHER IS RUNNING. SUPPORTS WERE STIFFENED. IT IS NOT CERTAIN IF THIS OCCURRENCE IS UNIQUE TO MODES 2,3,4,5,&6; IT COULD BE DUE TO LOWER PRESSURES.	1C, 1J
328/90013	SEQUOYAH 2	LOSS OF SHUTDOWN COOLING	09/11/90	LOSS OF SHUTDOWN COOLING OCCURRED, WHILE IN MODE 5, WHEN A SUCTION ISOLATION VALVE WAS INADVERTENTLY CLOSED WHEN POWER WAS REMOVED FROM AN AUTOCLOSURE INTERLOCK. THE OPERATOR FAILED TO IDENTIFY THE EFFECTS OF REMOVING RPS RACKS FROM SERVICE.	1H, 1M, 1N

TABLE 1-2

SHUTDOWN EVENTS AND SYSTEM 80+
PREVENTION, DETECTION, AND MITIGATION FEATURES

LER. NO.	PLANT NAME	EVENT CATEGORY	DATE	EVENT SUMMARY	SYSTEM 80+ PREVENTION, DETECTION, AND MITIGATING FEATURES (SEE TABLE 1-1)
275-91003	DIABLO CANYON 1	LOSS OF SHUTDOWN COOLING	02/07/91	PERSONNEL ERROR ALLOWED TRIP OF RHR PUMPS DURING CAVITY FILL AND SUBSEQUENT FAILURE TO COMMUNICATE OPEN SHIFT TURNOVER.	1L, 1M
295/91005	ZION 1	LOSS OF SHUTDOWN COOLING	04/21/91	LOSS OF INSTRUMENT INVERTER OUTPUT POWER CAUSES MOMENTARY HIGH RCS PRESSUE SIGNAL THAT INITIATES CLOSE OF RHR SUCTION VALVE RESULTING IN LOSS OF RESIDUAL HEAT REMOVAL SYSTEM DURING COLD SHUTDOWN (MODE 5).	1H, 1M, 1N
456/91006	BRAIDWOOD 1	LOSS OF SHUTDOWN COOLING	04/27/91	PLANT WAS ENTERING MODE 4 FROM MODE 5. SDC TRAIN 1A WAS OPERATING, BUT THE PUMP STOPPED AS A RESULT OF A MULTITUDE OF MANUAL AND AUTOMATIC ACTIONS AND REACTIONS IN THE ELECTRICAL POWER SUPPLY DURING SURVEILLANCE TESTING. SDC WAS RESTORED USING TRAIN 1B.	1H, 1M, 1J
382/91005	WATERFORD 3	LOSS OF SHUTDOWN COOLING	05/25/91	THE PLANT WAS IN MODE 5. SHUTDOWN COOLING TRAIN B WAS OPERATING WHEN THE PUMP BECAME AIRBOUND. SDC WAS RESTORED USING TRAIN A. THE PUMP LOST SUCTION AS A RESULT OF MAINTENANCE ON A HPSI CHECK VALVE IN A BRANCH LINE.	1C, 1D, 1H, 1M
251-91003	TURKEY POINT 4	LOSS OF SHUTDOWN COOLING	06/26/91	INADEQUATE PROCEDURE CONTROLS LED TO LOSS OF POWER TO COMPONENT COOLING WATER PUMPS AND LOSS OF SPENT FUEL POOL COOLING WHILE CORE WAS OFFLOADED.	1M, 1D
NOTE (1)	TROJAN 1	LOSS OF SHUTDOWN COOLING	07/23/91	IMPROPERLY ADJUSTED RELIEF VALVES IN THE COMPONENT COOLING WATER SYSTEM OPENED UPON A PRESSURE SPIKE CAUSED BY STARTUP OF A SECOND CCW PUMP AND SUBSEQUENTLY FAILED TO RESEAT RESULTING IN LEAKAGE OF CCW INVENTORY AND POTENTIAL LOSS OF RHR.	1M
250-91005	TURKEY POINT 3	LOSS OF SHUTDOWN COOLING	08/30/91	PERSONNEL ERROR CAUSED CAVITATION IN COMPONENT COOLING WATER PUMPS WITH POTENTIAL LOSS OF RHR DURING MODE 6.	1M, 1D
424/91-009	VOGTLE 1	LOSS OF SHUTDOWN COOLING	10/26/91	INADEQUATE PRESSURIZER VENT CAUSED INACCURATE RCS LEVEL INDICATION WHILE DRAINING REFUELING CAVITY BELOW FLANGE LEVEL LEADING TO VORTEXING AND LOSS OF RHR PUMPED FLOW.	1A, 1B, 1C, 1D, 1G, 1H, 1M, 7B

(1) NO LICENSEE EVENT REPORT (LER) NUMBER AVAILABLE. THIS EVENT SUMMARY WAS OBTAINED FROM ANOTHER SOURCE OF INFORMATION.

TABLE 1-2

SHUTDOWN EVENTS AND SYSTEM 80+
PREVENTION, DETECTION, AND MITIGATION FEATURES

LER. NO.	PLANT NAME	EVENT CATEGORY	DATE	EVENT SUMMARY	SYSTEM 80+ PREVENTION, DETECTION, AND MITIGATING FEATURES (SEE TABLE 1-1)
NOTE (1)	INDIAN POINT 2	LOSS OF ELECTRICAL POWER	11/16/84	WHILE IN COLD SHUTDOWN FOR MORE THAN 1 MONTH, LOSS OF OFFSITE POWER OCCURRED DUE TO SHEET METAL BLOWING ACROSS PHASES OF BUSWORK. 1 D/G OUT OF SERVICE, OTHER 2 D/G S STARTED BUT AN OUTPUT BREAKER T. ONE VITAL BUS DID NOT CLOSE BECAUSE THE BREAKER FOR NORMAL OFFSITE POWER HAD NOT OPENED. OTHER ATTEMPTS TO ENERGIZE THAT BUS FAILED DUE TO CONTROL POWER FUSES BEING BLOWN.	2A, 2B, 2D, 2E, 2F
285/87008	FORT CALHOUN	LOSS OF ELECTRICAL POWER	03/21/87	DUE TO PERSONNEL ERROR, ALL AC OFFSITE ELECTRICAL POWER WAS LOST FOR 40 MINUTES AND THE ONE OPERABLE DIESEL GENERATOR DID NOT START BECAUSE CONTROL SWITCH WAS IN "OFF" POSITION DUE TO ERECTION OF SCAFFOLDING AROUND DIESEL. THE DIESEL WAS MANUALLY STARTED AND SHUTDOWN COOLING WAS THEN RESTORED AFTER 5 MINUTES.	2A, 2B, 2C, 2D, 2E, 2F
NOTE (1)	MCGUIRE 1	LOSS OF ELECTRICAL POWER	09/08/87	AT MID-LOOP, VALVE BETWEEN DIESEL FUEL STORAGE TANK AND DAY TANK FOR ONE DIESEL CLOSED IN ERROR. OTHER D/G WAS INOPERABLE. DIESEL WOULD HAVE TRIPPED HAD OPERATORS NOT FOUND PROBLEM. DAY TANK WAS AVAILABLE.	2A, 2E, 2F
400/87059	HARRIS	LOSS OF ELECTRICAL POWER	10/11/87	ONE INCOMING LINE TO ONE OF THE SAFETY BUSES WAS OUT OF SERVICE FOR MODIFICATION. THE ONE REMAINING INCOMING LINE BREAKER TRIPPED DUE TO ACCIDENTAL JARRING OF PROTECTION RELAYS. THE DIESEL GENERATOR ON THIS BUS STARTED AND LOADED APPROPRIATELY.	2A, 2B, 2D, 2E, 2F
NOTE (1)	WOLF CREEK	LOSS OF ELECTRICAL POWER	10/15/87	A SERIES OF ENGINEERED SAFETY FEATURE ACTUATIONS OCCURRED DUE TO REDUCED BATTERY VOLTAGE. THE 480 VOLT AC BUS WHICH POWERS THE BATTERY CHARGERS WAS REMOVED FROM SERVICE FOR MAINTENANCE. AT THE ONSET OF THE WORK, A BATTERY LIFE TIME ESTIMATE WAS MADE. HOWEVER, THIS TIME WAS EXCEEDED AND THE 125 VOLT DC SOURCE WAS LOST.	2C, 2D
302/87025	CRYSTAL RIVER 3	LOSS OF ELECTRICAL POWER	10/16/87	WHILE SHUTDOWN FOR REFUELING, THE STARTUP TRANSFORMER WAS ACCIDENTALLY SHORTED IN PREPARATION FOR MAINTENANCE. ONE OF TWO VITAL BUSES LOST POWER, AN ESF ACTUATION COMPLICATED POWER RESTORATION. POWER TO THE CONTROL ROOM ANNUNCIATORS AND EVENT RECORDER WAS LOST. AN ALERT WAS DECLARED.	2A, 2B, 2C, 2D

(1) NO LICENSEE EVENT REPORT (LER) NUMBER AVAILABLE. THIS EVENT SUMMARY WAS OBTAINED FROM ANOTHER SOURCE OF INFORMATION.

TABLE I-2

SHUTDOWN EVENTS AND SYSTEM 80+
PREVENTION, DETECTION, AND MITIGATION FEATURES

LER. NO.	PLANT NAME	EVENT CATEGORY	DATE	EVENT SUMMARY	SYSTEM 80+ PREVENTION, DETECTION, AND MITIGATING FEATURES (SEE TABLE I-1)
247/87013	INDIAN POINT	LOSS OF ELECTRICAL POWER	11/05/87	WHILE IN COLD SHUTDOWN, WITH ALL 3 DIESEL GENERATORS DOWN FOR MAINTENANCE AND ALMOST ALL ESF TAGGED OUT OF OPERATION, A TECHNICIAN INITIATED A PARTIAL SIS CAUSING LOSS OF 480V VITAL POWER BECAUSE THE DIESELS WERE TAGGED OUT. ADDITIONAL BATTERY CABLE WORK CAUSED LOSS OF INSTRUMENT BUSES WHICH [illegible] EVENT. NORMAL OFFSITE POWER WAS RE[illegible] MINUTES.	2A, 2B, 2C, 2D
336/88005	MILLSTONE 2	LOSS OF ELECTRICAL POWER	02/04/88	IN PREPARATION FOR A TEST, THE AUXILIARY CONTACT OF A BREAKER WAS ENERGIZED CAUSING LOSS OF ONE TRAIN ON VITAL 4160 V AC. THE OTHER TRAIN WAS OUT OF SERVICE FOR MAINTENANCE. THE DIESEL GENERATOR STARTED BUT BECAUSE OF A SEQUENCER FAILURE DID NOT LOAD. OPERATOR ACTION WAS REQUIRED TO RE-ENERGIZE THE BUS AND REESTABLISH SHUTDOWN COOLING.	2A, 2B, 2C, 2D
029/88010	YANKEE ROWE	LOSS OF ELECTRICAL POWER	11/16/88	PLANT WAS IN MODE 5 WITH GENERATOR CIRCUIT TESTING IN PROGRESS. DUE TO A MAINTENANCE ERROR, POWER WAS LOST TO TWO EMERGENCY 480 VOLT BUSES. ONE SOURCE OF OFFSITE POWER AND ONE DIESEL GENERATOR WERE UNAVAILABLE DUE TO MAINTENANCE. THE PLANT HAS THREEE DIESEL GENERATORS. A TIE BREAKER PROBLEM DELAYED RESTORING POWER TO ONE OF THE BUSES.	2A, 2B, 2C, 2D
313/89040	ARKANSAS NUC ONE 1	LOSS OF ELECTRICAL POWER	12/15/89	PERSONNEL ERROR CAUSED LOSS OF POWER RESULTING IN AUTOMATIC ACTUATION OF AN EMERGENCY DIESEL GENERATOR AND THE LOSS OF RHR FOR APPROXIMATELY 9 MINUTES DURING WHICH TIME THE REACTOR COOLANT SYSTEM TEMPERATURE INCREASED 17 DEGREES.	2B, 2C, 2D
285/90006	FORT CALHOUN 1	LOSS OF ELECTRICAL POWER	02/26/90	ONE OF TWO DIESEL GENERATORS AND ONE OF TWO SOURCES OF OFFSITE POWER WERE OUT OF SERVICE. THE CONTAINMENT EQUIPMENT HATCH WAS OPEN. POWER TO 4160 SAFETY BUSES WAS LOST WHEN A TRIP CIRCUIT BREAKER OPENED FOR UNKNOWN REASONS. THE OPERABLE DIESEL GENERATOR STARTED. BECAUSE OF A DESIGN FEATURE WOULD NOT LOAD ON THE BUS UNTIL THE RHR PUMP MOTOR CONNECTED TO THE BUS WAS TRIPPED. THE PUMP MOTOR BREAKER WAS OPENED AND THE BUS WAS RE-ENERGIZED. A 2 DEG F RCS TEMPERATURE INCREASE OCCURRED.	2A, 2B, 2D, 2E, 2F
424/90006	VOGTLE 1	LOSS OF ELECTRICAL POWER	03/20/90	AT MID-LOOP, A LOSS OF OFFSITE POWER TO OPERATING	2A, 2B, 2D, 2E

TABLE 1-2

SHUTDOWN EVENTS AND SYSTEM 3PH
PREVENTION, DETECTION, AND MITIGATION FEATURES

LER. NO.	PLANT NAME	EVENT CATEGORY	DATE	EVENT SUMMARY	SYSTEM 3PH PREVENTION, DETECTION, AND MITIGATING FEATURES (SEE TABLE 1-1)
				RHR PUMP AND THE FAILURE OF DIESEL GENERATOR TO CONTINUE TO RUN OCCURRED. RHR RESTORED AFTER 41 MINUTES; RCS TEMPERATURE ROSE 46 DEG F TO 126 DEG F.	
336/90016	MILLSTONE 2	LOSS OF ELECTRICAL POWER	10/09/90	LOSS OF POWER TO THE CONTAINMENT RADIATION MONITOR CAUSES INADVERTENT ESAS ACTUATION OF CONTAINMENT PURGE ISOLATION SYSTEM DURING REFUELING.	2C
275-91004	DIABLO CANYON 1	LOSS OF ELECTRICAL POWER	07/07/91	PERSONNEL ERROR OUTSIDE CONTAINMENT SHORTED OFFSITE POWER LINE CAUSING POTENTIAL FOR LOSS OF RHR.	2A, 2B, 2C, 2D, 2E, 2F, 2G
251-91003	TURKEY POINT 4	LOSS OF ELECTRICAL POWER	03/13/91	POOR HOUSEKEEPING ... CORE OFF LOADED CAUSED WASTE MATERIAL TO SHORT OUT POWER SUPPLY WITH CONSEQUENT LOSS OF SPENT FUEL POOL COOLING AND 3 DEGREE TEMPERATURE RISE.	2A, 2B, 2C, 2E
247-91006	INDIAN POINT 2	LOSS OF ELECTRICAL POWER	03/20/91	LOSS OF NORMAL OFF-SITE POWER WHILE 2 OF 3 DIESELS OUT OF SERVICE WITH CONSEQUENT LOSS OF ONE DC BUS, WHILE CORE OFF LOADED. LOSS OF COMPONENT COOLING WATER PUMPS, BUT SPENT FUEL POOL COOLING CONTINUED FROM AN ALTERNATE OFF-SITE SOURCE.	2A, 2B, 2C
456-91006	BRAIDWOOD 1	LOSS OF ELECTRICAL POWER	04/22/91	WHILE IN MODE 5, DURING RELAY REPAIR, RHR AND COMPONENT COOLING PUMP POWER WAS LOST.	2A, 2B, 2C, 2E, 3D
247-91010	INDIAN POINT 2	LOSS OF ELECTRICAL POWER	06/22/91	DIESEL TRIP WHILE OFFSITE BREAKER OPEN FOR TEST DURING COLD SHUTDOWN RESULTED IN LOSS OF SPENT FUEL POOL COOLING FOR 36 MINUTES.	2A, 2B, 2C, 2D, 2E, 2F

TABLE 1-2

SHUTDOWN EVENTS AND SYSTEM RHR PREVENTION, DETECTION, AND MITIGATION FEATURES

LER. NO.	PLANT NAME	EVENT CATEGORY	DATE	EVENT SUMMARY	SYSTEM RHR PREVENTION, DETECTION, AND MITIGATING FEATURES (SEE TABLE 1-1)
364/87008	FARLEY 2	LOSS OF REACTOR COOLANT	11/27/87	DURING A REFUELING OUTAGE, NEGLECT BY PERSONNEL TO REFILL A SECTION OF PIPE BETWEEN THE CONTAINMENT SUMP SUCTION ISOLATION VALVES AFTER DRAINING RESULTED IN A PRESSURE PULSE THAT OPENED A RHR LOOP SUCTION PRESSURE RELIEF VALVE DURING STROKING OF CONTAINMENT SUMP SUCTION VALVES THEREBY CAUSING A DECREASE IN THE REACTOR COOLANT SYSTEM PRESSURE AND PRESSURIZER LEVEL.	3A
327/88021	SEQUOYAH 1	LOSS OF REACTOR COOLANT	05/23/88	IN COLD SHUTDOWN WITH THE RCS PARTIALLY DRAINED, IMPROPER VALVE ALIGNMENT CAUSED BY POOR COMMUNICATIONS RESULTED IN A LOSS OF REACTOR COOLANT WATER INVENTORY, RHR PUMP CAVITATION, AND LOSS OF RHR COOLING. RHR RECOVERED AFTER GRAVITY FILLING FROM RWST.	1B, 1C, 1D, 1M
NOTE (1)	SURRY 1	LOSS OF REACTOR COOLANT	05/27/88	ABOUT 25,000 GALLONS OF REACTOR COOLANT WERE DRAINED FROM THE REACTOR CAVITY DUE TO LEAKING REACTOR CAVITY RING DUE TO ISOLATION OF THE INSTRUMENT AIR FOR MAINTENANCE.	1C, 1D, 1M
361/88017	SAN ONOFRE 2	LOSS OF REACTOR COOLANT	06/22/88	FAILURE TO IMPLEMENT UPDATED FINAL SAFETY ANALYSIS REPORT (UFSAR) COMMITMENT RESULTS IN DRAINING OF APPROXIMATELY 7000 GALLONS FROM THE SPENT FUEL POOL INTO THE REACTOR CAVITY. UFSAR REQUIRES THAT ALL CONNECTIONS TO THE SPENT FUEL STORAGE POOL ARE MADE SO AS TO PRECLUDE THE POSSIBILITY OF SIPHON DRAINING OF THE POOL.	3A, 3D
255/89033	PALISADES	LOSS OF REACTOR COOLANT	11/21/89	DUE TO PRESSURIZER PORV DESIGN CHARACTERISTICS, WHEN THE PORV BLOCK VALVE OPENED, THE PORV ALSO OPENED AND RCS PRESSURE DROPPED FROM 2154 TO 1565 PSI.	3A
456/89016	BRAIDWOOD 1	LOSS OF REACTOR COOLANT	12/01/89	AT 350 PSIG, APPROX 67,000 GALLONS OF REACTOR COOLANT WAS LOST WHEN A RHR PUMP SUCTION RELIEF VALVE LIFTED. ABOUT 2 AND 1/2 HOURS WAS REQUIRED TO ISOLATE THE LEAK.	6A, 6C
457/90002	BRAIDWOOD 2	LOSS OF REACTOR COOLANT	03/18/90	FROM SHUTDOWN AT 350 PSIG, RHR SUCTION VALVE WAS OPENED BEFORE CLOSING THE RHR PUMP DISCHARGE VALVE TO THE RWST WHICH HAD BEEN PREVIOUSLY OPENED ON RECIRCULATION FOR BORON SAMPLING. WITH THE RHR PUMP OFF, COOLANT DRAINED FROM THE PRESSURIZED RCS TO THE RWST.	3A

(1) NO LICENSEE EVENT REPORT (LER) NUMBER AVAILABLE. THIS EVENT SUMMARY WAS OBTAINED FROM ANOTHER SOURCE OF INFORMATION.

TABLE [illegible]

SHUTDOWN EVENTS AND SYSTEM B--
PREVENTION, DETECTION, AND MITIGATION FEATURES

LER. NO.	PLANT NAME	EVENT CATEGORY	DATE	EVENT SUMMARY	SYSTEM B-- PREVENTION, DETECTION, AND MITIGATING FEATURES (SEE TABLE 3-1)	
[illegible]	YANKEE 1	LOSS OF REACTOR COOLANT	[illegible]	FAILURE OF SEAL IN AN INCORE INSTRUMENTATION TUBE AT THE SEAL TABLE RESULTS IN PRESSURE BOUNDARY LEAKAGE AND POTENTIAL FOR CORE UNCOVERY.	[illegible]	

TABLE 1-2

SHUTDOWN EVENTS AND SYSTEM 80+
PREVENTION, DETECTION, AND MITIGATION FEATURES

LER. NO.	PLANT NAME	EVENT CATEGORY	DATE	EVENT SUMMARY	SYSTEM 80+ PREVENTION, DETECTION, AND MITIGATING FEATURES (SEE TABLE 1-1)
NOTE (1)	SALEM 2	CONTAINMENT INTEGRITY	05/25/83	IN REFUELING, CONTAINMENT INTEGRITY COULD NOT BE ESTABLISHED WITHIN THE REQUIRED 8 HOURS WITH THE ELECTRICAL POWER AVAILABLE. CONDITION LASTED ABOUT 9 HOURS.	4A, 4B, 4C, 2A, 2B
NOTE (1)	SAN ONOFRE 1	CONTAINMENT INTEGRITY	02/13/85	IN MODE 3, A SECURITY OFFICER REQUIRED TO UNLOCK BOTH NORMAL AND EMERGENCY HATCHES, MANIPULATED THE EMERGENCY HATCH SO THAT BOTH DOORS WERE OPEN. CONDITION EXISTED ABOUT 21 HOURS.	4C
NOTE (1)	FARLEY 1	CONTAINMENT INTEGRITY	04/11/85	DURING REFUELING, BOTH DOORS OF PERSONNEL AIR LOCK WERE INADVERTENTLY LEFT OPEN FOR ABOUT 4 HOURS.	4C
NOTE (1)	MCGUIRE 1	CONTAINMENT INTEGRITY	05/25/85	IN REFUELING, DURING CORE ALTERATIONS, CONTAINMENT VENTILATION COOLING WATER SYSTEM VENT VALVE WAS FOUND LOCKED OPEN. THIS CONFIGURATION CREATED A PATH FROM THE UPPER CONTAINMENT TO THE AUXILIARY BUILDING THROUGH THE OPEN VALVE INSIDE CONTAINMENT.	4A
NOTE (1)	FARLEY 1	CONTAINMENT INTEGRITY	09/30/89	ON TWO SEPARATE OCCASIONS CORE ALTERATIONS WERE PERFORMED WITHOUT CONTAINMENT INTEGRITY REQUIRED BY TECHNICAL SPECIFICATIONS. A PATHWAY EXISTED FROM THE STEAM GENERATOR SECONDARY SIDE MANWAY IN CONTAINMENT THROUGH THE STEAM GENERATOR ATMOSPHERIC RELIEF VALVE.	4A
266/90011	POINT BEACH 1	CONTAINMENT INTEGRITY	08/29/90	ENGINEERING EVALUATION INDICATES THAT THE RHR PUMPS MAY NOT PROVIDE ADEQUATE NPSH TO THE CONTAINMENT SPRAY PUMPS WHEN THE ECCS IS IN THE RECIRCULATION MODE. CORRECTIVE ACTIONS INCLUDE PROCEDURE CHANGES.	1J
412/90013	BEAVER VALLEY 2	CONTAINMENT INTEGRITY	09/15/..	WATER TURBULENCE DURING THE REACTOR CAVITY FILL OPERATION CAUSES AN INCREASE IN CONTAINMENT AIRBORNE ACTIVITY RESULTING IN AUTOMATIC CONTAINMENT PURGE ISOLATION.	--
336-90018	MILLSTONE 2	CONTAINMENT INTEGRITY	10/02/90	PERSONNEL ERROR WHILE DRAINING STEAM GENERATOR DURING REFUELING OUTAGE CAUSED OPEN PATHWAY FROM CONTAINMENT THROUGH MANWAY AND ADV TO THE OUTSIDE ATMOSPHERE.	4A, 4C
336-999161	MILLSTONE 2	CONTAINMENT INTEGRITY	10/09/90	DURING MODE 6 CORE ALTERATIONS, MAINTENANCE PERSONNEL ERROR RESULTED IN UNAVAILABILITY OF	4A, 4C

(1) NO LICENSEE EVENT REPORT (LER) NUMBER AVAILABLE. THIS EVENT SUMMARY WAS OBTAINED FROM ANOTHER SOURCE OF INFORMATION.

TABLE 1-2

SHUTDOWN EVENTS AND SYSTEM 80+
PREVENTION, DETECTION, AND MITIGATION FEATURES

LER. NO.	PLANT NAME	EVENT CATEGORY	DATE	EVENT SUMMARY	SYSTEM 80+ PREVENTION, DETECTION, AND MITIGATING FEATURES (SEE TABLE 1-1)
				CONTAINMENT PURGE VALVE AUTOMATIC ISOLATION CAPABILITY.	
309/90009	MAINE YANKEE 1	CONTAINMENT INTEGRITY	10/20/90	A DESIGN DEFICIENCY IN THE CONTAINMENT ISOLATION VALVE POSITION INDICATION SYSTEM CAUSED IMPROPER ILLUMINATION OF SEVERAL ISOLATION VALVE POSITION INDICATOR LIGHTS DURING HOT SHUTDOWN.	---
213/90024	CONNECTICUT YANKEE 1	CONTAINMENT INTEGRITY	10/31/90	DURING SURVEILLANCE TESTING WITH PLANT IN COLD SHUTDOWN (MODE 5) A DESIGN DEFICIENCY IN THE CONTAINMENT ISOLATION VALVE WHICH SUPPLIES COMPONENT COOLING WATER TO THE NEUTRON SHIELD TANK COOLER CAUSED THE VALVE TO BE INOPERABLE.	1D
255/91004	PALISADES 1	CONTAINMENT INTEGRITY	01/05/91	INADEQUATE PREVENTATIVE MAINTENANCE PROCEDURES RESULTS IN EXCESSIVE LEAKAGE THROUGH VIEWING PORT GASKET ON INNER DOOR OF CONTAINMENT EMERGENCY AIRLOCK. LEAKAGE WAS DETERMINED DURING COLD SHUTDOWN MODE WITH REACTOR COMPLETELY DEFUELED.	4C
413/91005	CATAWBA 1	CONTAINMENT INTEGRITY	01/10/91	EQUIPMENT FAILURE-MALFUNCTION RESULTS IN UNLATCHING OF EMERGENCY PERSONNEL HATCH DURING HOT STANDBY (MODE 3).	---
338/91005	NORTH ANNA 1	CONTAINMENT INTEGRITY	02/09/91	INADEQUATE PROCEDURES RESULTED IN BREACH OF CONTAINMENT BUILDING PENETRATIONS DURING REFUELING.	4A, 4L
272/91006	SALEM 1	CONTAINMENT INTEGRITY	02/27/91	EVENT OCCURRED ON 2/16, 2/20, AND 2/27. INADEQUATE ADMINISTRATIVE CONTROLS PLUS EQUIPMENT FAILURE OF THE RADIATION MONITORING SYTEM RESULTS IN ACTUATION OF ESF SIGNALS (CONTAINMENT PURGE - PRESSURE - VACUUM RELIEF) FOR THE CONTAINMENT VENT ISOLATION SYSTEM DURING COLD SHUTDOWN (MODE 5) AND REFUELING (MODE 6).	4C
213/91006	CONNECTICUT YANKEE 1	CONTAINMENT INTEGRITY	03/05/91	DURING COLD SHUTDOWN (MODE 5), AN ENGINEERING EVALUATION OF CONTAINMENT INSTRUMENTATION PENETRATIONS REVEALED THAT INADEQUATE PROCEDURES PERTINENT TO CALIBRATION OF PRESSURE TRANSMITTERS RESULTED IN A VIOLATION OF CONTAINMENT INTEGRITY. CORRECTIVE ACTION CONSISTED OF REVISING THE PERTINENT PROCEDURES.	4C
213/91010	CONNECTICUT YANKEE 1	CONTAINMENT INTEGRITY	03/15/91	PERSONNEL ERROR CAUSED A MANUAL CONTAINMENT ISOLATION VALVE TO REMAIN OPEN DURING HOT STANDBY	4C

TABLE 1-2

SHUTDOWN EVENTS AND SYSTEM 8H+
PREVENTION, DETECTION, AND MITIGATION FEATURES

LER. NO.	PLANT NAME	EVENT CATEGORY	DATE	EVENT SUMMARY	SYSTEM 8H+ PREVENTION, DETECTION, AND MITIGATING FEATURES (SEE TABLE 1-1)
				(MODE 3) PROVIDING A BREACH OF CONTAINMENT INTEGRITY.	
275/91006	DIABLO CANYON 1	CONTAINMENT INTEGRITY	03/26/91	SAMPLE PUMP OF RADIATION MONITORING SYSTEM SEIZED, RESULTING IN ACTUATION OF ESF SIGNAL FOR THE CONTAINMENT VENT ISOLATION SYSTEM DURING COLD SHUTDOWN.	--
266/91004	POINT BEACH 1	CONTAINMENT INTEGRITY	05/10/91	DURING STEAM GENERATOR CREVICE FLUSHING, THE PRIMARY SYSTEM TEMPERATURE EXCEEDED 200 DEG F FOR APPROX. 10 MINUTES BEFORE BEING COOLED TO LESS THAN 200 DEG F USING THE RHR SYSTEM.	4C
369-91017	MCGUIRE 1	CONTAINMENT INTEGRITY	09/21/91	INADEQUATE PROCEDURES ALLOWED A PRESSURE BOUNDARY DOOR TO THE ANNULUS VENTILATION SYSTEM TO REMAIN OPEN DURING MODES 1, 2, 3, AND 4.	4C
301-91003	POINT BEACH 2	CONTAINMENT INTEGRITY	10/10/91	PERSONNEL ERROR ALLOWED OPEN CONTAINMENT HATCH DURING RFUELING SHUTDOWN.	4A, 4C
335/91008	ST. LUCIE 1	CONTAINMENT INTEGRITY	11/10/91	FAILURE OF MAINTENANCE PERSONNEL TO FOLLOW APPROVED PROCEDURE RESULTED IN IMPROPER MANIPULATION OF RELIEF AND DRAIN VALVES WHICH CREATED A DIRECT FLOW PATH FROM CONTAINMENT TO THE REACTOR AUXILIARY BUILDING THEREBY COMPROMISING CONTAINMENT INTEGRITY DURING REFUELING.	4C
213/91025	CONNECTICUT YANKEE 1	CONTAINMENT INTEGRITY	11/14/91	INADEQUATE PROCEDURES RESULTED IN IMPROPER ALIGNMENT OF ALL CONTAINMENT PENETRATIONS DURING REFUELING.	4A, 4C
529/91007	PALO VERDE 2	CONTAINMENT INTEGRITY	12/05/91	INADEQUATE PROCEDURES RESULTS IN OPEN CONTAINMENT ISOLATION VALVE DURING CORE ALTERATIONS WHICH IS CONTRARY TO TECH SPEC REQUIREMENTS THAT EACH PENETRATION PROVIDING DIRECT ACCESS FROM THE CONTAINMENT ATMOSPHERE TO THE OUTSIDE ATMOSPHERE SHALL BE CLOSED BY AN ISOLATION VALVE.	4A, 4C

TABLE 1-7

SHUTDOWN EVENTS AND SYSTEM 8H+
PREVENTION, DETECTION, AND MITIGATION FEATURES

LER. NO.	PLANT NAME	EVENT CATEGORY	DATE	EVENT SUMMARY	SYSTEM 8H+ PREVENTION, DETECTION, AND MITIGATING FEATURES (SEE TABLE 1-1)
423/88005	MILLSTONE 3	OVERPRESSURIZATION	01/19/88	FAILURE OF THE LOW TEMPERATURE OVERPRESSURE PROTECTION SYSTEM TO OPERATE. THE EVENT WAS MITIGATED BY OPERATOR ACTION. ISOLATION OF THE RHR LETDOWN PA CAUSED A PRESSURE EXCURSION. THE PEAK RCS PRESS. . REACHED WAS 526 PSIA. THE LOW TEMPERATURE OVERPRESSURE PROTECTION SYSTEM DID NOT WORK BECAUSE A POWER SUPPLY WHICH WAS REQUIRED FOR SYSTEM OPERATION HAD ITS FUSES REMOVED WITHOUT KNOWING WHAT EFFECT THIS REMOVAL WOULD HAVE ON THE SYSTEM.	5A, 3F
NOTE (1)	SURRY 1	OVERPRESSURIZATION	04/15/88	BOTH PRESSURIZER PORV'S FAILED TO OPEN MANUALLY. TORQUE VALUES FOR DIAPHRAGM HOLD DOWN BOLTS WERE NOT SPECIFIED AND ACTUATOR OPERATION WAS INTERMITTENT. REQUIRED TO BE OPERABLE FOR LTOP.	5A, 5C
311/89021	SALEM	OVERPRESSURIZATIONS	10/27/89	ONE TRAIN THE RHR SUCTION PIPING WAS PRESSURIZED TO 600 PSIG (150 PSIG ABOVE THE DESIGN). THE EVENT OCCURRED BECAUSE THE RHR COLD LEG INJECTION VALVE DID NOT SEAT PROPERLY AND ALLOWED RCS COOLANT TO PRESSURIZE THE LINE.	5A, 5C, 3M
255/90017	PALISADES 1	OVERPRESSURIZATION	09/26/90	PALISADES HAS BOTH SPRING-LOADED SAFETY VALVES AND POWER-OPERATED RELIEF VALVES (PORV) WHICH ARE PARALLEL. THE PORV IS SET AT A LOWER PRESSURE THAN THE SAFETY VALVE. THERE ARE BLOCK VALVES UPSTREAM OF THE PORV'S. THE PORV'S WERE NOT OPERABLE IN COLD SHUTDOWN AND ON SHUTDOWN COOLING AS IS REQUIRED WITH RCS TEMPERATURE LESS THAT 430 DEG F. THE PORV'S WOULD NOT OPEN WITH A FAILURE OF THE TEMPERATURE INPUT RESULTING IN LOSS OF LOW TEMPERATURE OVERPRESSURE PROTECTION FOR THE RCS.	5A, 5C

(1) NO LICENSEE EVENT REPORT (LER) NUMBER AVAILABLE. THIS EVENT SUMMARY WAS OBTAINED FROM ANOTHER SOURCE OF INFORMATION.

TABLE 3-2

SHUTDOWN EVENTS AND SYSTEM 80+
PREVENTION, DETECTION, AND MITIGATION FEATURES

LER. NO.	PLANT NAME	EVENT CATEGORY	DATE	EVENT SUMMARY	SYSTEM 80+ PREVENTION, DETECTION, AND MITIGATING FEATURES (SEE TABLE 3-1)
287/76019	OCONEE 3	FLOODING AND SPILLS	10/09/76	WHILE IN COLD SHUTDOWN, WITH THE CONDENSER WATERBOX MANWAYS OPEN FOR MAINTENANCE, VALVE FAILURES ALLOWED LAKE WATER TO FLOW BY GRAVITY OUT THE OPEN MANWAYS, FLOODING THE TURBINE BUILDING TO A DEPTH OF 24 INCHES AND THREATENING THE AUXILIARY BUILDING CONTAINING SAFETY RELATED EQUIPMENT.	6A, 6B, 6C
247/84011	INDIAN POINT 2	FLOODING AND SPILLS	08/13/84	DURING MAINTENANCE, A VALVE HAD BEEN REMOVED FROM ONE SERVICE WATER TRAIN LEAVING AN OPEN PIPE, LOCATED IN THE COMPONENT COOLING WATER PUMP ROOM. SUBSEQUENTLY, ANOTHER OPERATOR CLOSED VALVES WHICH DIVERTED WATER THROUGH A PARTIALLY OPEN CROSS-CONNECT VALVE BETWEEN THE OPERATING TRAIN AND THE OUT-OF-SERVICE TRAIN. LEAKAGE FROM THE OPEN PIPE FLOODED THE ROOM, COVERING ALL THREE CCW PUMP MOTORS CAUSING LOSS OF ALL CCW FLOW.	6A, 6B, 6C
213/84-13	CONNECTICUT YANKEE	FLOODING AND SPILLS	08/21/84	THE REFUELING CAVITY SEAL FAILED AND ABOUT 200,000 GALLONS OF REACTOR COOLANT SPILLED INTO THE CONTAINMENT IN ABOUT 20 MINUTES. CONTAINMENT SUMP OVERFLOWED AND LEVEL IN CONTAINMENT REACHED ABOUT 18 INCHES. IT WAS POSTULATED THAT FOR A WORST CASE SCENARIO, THAT THE FUEL POOL COULD HAVE BEEN DRAINED. THE FUEL TRANSFER CAN WAS NOT IN USE AND THE FUEL POOL WAS ISOLATED FROM THE CAVITY AT TIME OF THE EVENT. NRC BULLETIN 84-03 WAS ISSUED.	7E, 7F, 6C
NOTE (1)	SALEM 1	FLOODING AND SPILLS	12/22/87	IN MODE 5 FOR REFUEL OUTAGE, WITH TWO BLANK FLANGES INSTALLED ON SERVICE WATER PIPING - ONE WHERE A VALVE WAS REMOVED AND ONE AT ANOTHER LOCATION, A LEAK OCCURRED. FLANGES HAD BEEN LOOSENED AFTER ORIGINAL INSTALLATION TO PROVIDE A NECESSARY LEAK PATH. DUE TO VALVE FAILURE AND FLANGES BEING LOOSE, A LARGE AMOUNT OF SERVICE WATER (ABOUT 144,000 GAL) ENTERED ONE SERVICE WATER BAY. THE WATER LEVEL WAS ABOVE THE SERVICE WATER PUMP MOTORS IN THE BAY. SOME ELECTRICAL EQUIPMENT AND THE CARDOX ROOM OUTSIDE THE BAY WERE WETTED.	6A, 6B, 6C
370/88018	MCGUIRE 2	FLOODING AND SPILLS	09/05/88	ABOUT 20,000 GALLONS OF WATER ENTERED THE AUXILIARY BUILDING WHEN ONE TRAIN OF CONTAINMENT SPRAY WAS OVERPRESSURIZED AND THE BOTTOM FLANGE GASKET OF A HEAT EXCHANGER FAILED.	6A, 6B, 6C

TABLE 1-2

SHUTDOWN EVENTS AND SYSTEM 80+
PREVENTION, DETECTION, AND MITIGATION FEATURES

LER NO.	PLANT NAME	EVENT CATEGORY	DATE	EVENT SUMMARY	SYSTEM 80+ PREVENTION, DETECTION, AND MITIGATING FEATURES (SEE TABLE 1-1)
NOTE (1)	SOUTH TEXAS	FLOODING AND SPILLS	04/09/90	WHILE THE REACTOR CAVITY WAS BEING FILLED, ONE OF TWO REQUIRED SPOOL PIECES HAD NOT BEEN INSTALLED. THIS CREATED A DRAIN PATH FROM THE LOWER EQUIPMENT STORAGE AREA TO THE CONTAINMENT AND 17,000 GAL OF WATER ENTERED CONTAINMENT.	6A, 6B
348/91004	FARLEY 1	FLOODING AND SPILLS	04/23/91	PERSONNEL ERROR CAUSED 5000 GALLONS OF WATER TO DRAIN FROM RWST TO CONTAINMENT FLOOR WHEN FIVE VALVES WERE IMPROPERLY OPENED IN CONTAINMENT SPRAY SYSTEM DURING VALVE TESTING.	6A, 6B

(1) NO LICENSEE EVENT REPORT (LER) NUMBER AVAILABLE. THIS EVENT SUMMARY WAS OBTAINED FROM ANOTHER SOURCE OF INFORMATION.

TABLE 1-2

SHUTDOWN EVENTS AND SYSTEM 84+
PREVENTION, DETECTION, AND MITIGATION FEATURES

LER. NO.	PLANT NAME	EVENT CATEGORY	DATE	EVENT SUMMARY	SYSTEM 80+ PREVENTION, DETECTION, AND MITIGATING FEATURES (SEE TABLE 5-1)
NOTE (1)	SAN ONOFRE 2	BORON AND REACTIVITY EVENTS	07/14/87	IMPROPER VALVE ALIGNMENT DURING NITROGEN BACKFLUSHING IN CCES RESULTED IN THE LPSI PUMPS BECOMING GAS BOUND. SDC WAS REESTABLISHED IN ABOUT 90 MINUTES. DURING THE RESTORATION, LPSI SUCTION WAS ALIGNED TO THE RWST. DUE TO BORON STRATIFICATION IN THE RWST, MEASURED BORON CONCENTRATION IN THE TANK WAS HIGHER THAN THAT PUMPED INTO THE RCS. THUS, THE ADDITION RESULTED IN DILUTION TO LESS THAN 2000 PPM.	7A, 7E
NOTE (1)	SALEM 1	BORON AND REACTIVITY EVENTS	07/16/84	0% POWER, ONE CHARGING PUMP SEIZED. METAL FILINGS AND RESIN PARTICLES WERE FOUND IN THE PUMP. SIMILAR MATERIAL WAS FOUND IN THE SUCTION LINES TO ALL CHARGING PUMPS. POTENTIAL EXISTED TO CAUSE LOSS OF ALL CENTRIFUGAL CHARGING PUMPS.	7A
NOTE (1)	CALLAWAY 1	BORON AND REACTIVITY EVENTS	07/15/84	INADEQUATE SURVEILLANCE PROCEDURES MADE BOTH TRAINS OF THE SRM DOUBLING CIRCUIT INOPERABLE FOR ABOUT 5 HOURS IN MODE 5. THE SRM TRIP AND HIGH FLUX ALARMS WERE STILL OPERATIONAL.	7A, 7E
250/87017	TURKEY POINT 3	BORON AND REACTIVITY EVENTS	05/29/87	NITROGEN GAS BINDING OF ALL BORIC ACID TRANSFER PUMPS, CAUSED BY FAILURE OF MECHANICAL SEAL, RESULTED IN LOSS OF THE NORMAL AND EMERGENCY BORIC ACID FLOW PATHS TO THE VOLUME CONTROL TANKS. CROSS TIE BETWEEN TURKEY POINT 3 AND 4 ALLOWED GAS LEAK FROM SEAL FAILURE ON UNIT 4 TO DISABLE PUMP ON UNIT 3.	2B
NOTE (1)	VOGTLE 1	BORON AND REACTIVITY EVENTS	06/06/87	DURING STARTUP, DUE TO ERROR IN THE ESTIMATED CRITICAL POSITION, REACTOR BECAME CRITICAL BEFORE EXPECTED. TRAINEE WAS PERFORMING STARTUP AND SRO WAS INATTENTIVE. PROCEDURE FAILED TO ADEQUATELY TREAT MODERATOR TEMPRATURE'S EFFECT ON BORON WORTH. REACTOR TRIP OCCURRED DUE TO HIGH SOURCE RANGE FLUX.	7A
368/88008	ARKANSAS NUC ONE 2	BORON AND REACTIVITY EVENTS	05/04/88	DURING HOT STANDBY, GAS BINDING OF CHARGING PUMPS CAUSED BY INADVERTENT EMPTYING OF VCT, DUE TO ERRONEOUS VCT LEVEL INDICATION, RESULTED IN LOSS OF RCS NORMAL MAKEUP/EMERGENCY BORATION CAPABILITY. ERRONEOUS LEVEL INDICATION WAS CAUSED BY LEAK ON THREADED FITTING OF LEVEL TRANSMITTER.	—
482/88025	WOLF CREEK 1	BORON AND REACTIVITY EVENTS	11/25/88	DIESEL GENERATOR ON ONE TRAIN OF EMERGENCY	7A, 2A

(1) NO LICENSEE EVENT REPORT (LER) NUMBER AVAILABLE. THIS EVENT SUMMARY WAS OBTAINED FROM ANOTHER SOURCE OF INFORMATION.

TABLE 1-2

SHUTDOWN EVENTS AND SYSTEM 80+
PREVENTION, DETECTION, AND MITIGATION FEATURES

LER. NO.	PLANT NAME	EVENT CATEGORY	DATE	EVENT SUMMARY	SYSTEM 80+ PREVENTION, DETECTION, AND MITIGATING FEATURES (SEE TABLE 1-1)
				CHARGING AND BORATION WAS REMOVED FROM SERVICE WHILE CHARGING PUMP IN THE REDUNDANT TRAIN WAS INOPERABLE RESULTING IN NO OPERABLE BORON INJECTION FLOW PATH DURING CORE ALTERATIONS.	
206/89002	SAN ONOFRE 1	BORON AND REACTIVITY EVENTS	01/23/89	FOLLOWING REFUELING, PERSONNEL ERROR RESULTED IN DECONTAMINATION OF REACTOR REFUELING CAVITY WITH APPROXIMATELY 440 GALLONS OF UNBORATED WATER WHICH REDUCED THE BORON CONCENTRATION OF THE REACTOR COOLANT BY 37 PPM. THE MINIMUM SHUTDOWN MARGIN OF 5% (1900 PPM BORON CONCENTRATION) WAS NOT APPROACHED. EVENT OCCURRED WHILE SOURCE RANGE MONITORS WERE INOPERABLE.	7A, 7C
250/89008	TURKEY POINT 3	BORON AND REACTIVITY EVENTS	02/20/89	WITH TURKEY POINT 4 IN MODE 5 AND TURKEY POINT 3 AT POWER, HEAT TRACED SUCTION LINES OF BORIC ACID TRANSFER PUMPS SHARED BY THE TWO UNITS WERE INOPERABLE BECAUSE OF LOW TEMPERATURE, JEOPARDIZING BORATION CAPABILITY.	7B
280/89016	SURRY 1	BORON AND REACTIVITY EVENTS	04/14/89	DURING COLD SHUTDOWN, A LEAK IN THE RCS STANDPIPE MAKEUP VALVE CAUSED AN INCREASE IN RCS INVENTORY RESULTING IN A DESCREASE IN RCS BORON CONCENTRATION. BORON CONCENTRATION WAS REDUCED BY LESS THAN 80 PPM OVER A PERIOD OF NINE DAYS. VALVE LEAKAGE WAS CAUSED BY IMPROPER ADJUSTMENT OF OPERATOR TRAVEL STOPS DURING MAINTENANCE.	7A, 7C
281/89015	SURRY 2	BORON AND REACTIVITY EVENTS	10/25/89	DUE TO PROBLEMS WITH THE BORIC ACID BLENDER IN THE CVCS, THE MODE OF OPERATION FOR THE BLENDER WAS CHANGED TO MANUAL. RCS FILLING WAS IN PROGRESS AND THE FLOWS WERE INCREASED TO EXPEDITE FILLING OF THE RCS. THIS RESULTED IN DECREASED BORON CONCENTRATION. ADJUSTMENT CAUSED THE BORON CONCENTRATION TO DECREASE BELOW THE REQUIRED 2000 PPM.	7A, 7C
414/90009	CATAWBA 2	BORON AND REACTIVITY EVENTS	07/23/90	CORE ALTERATIONS WERE IN PROGRESS WITHOUT AUDIBLE COUNT RATE INDICATION IN CONTROL ROOM OR CONTAINMENT FROM SOURCE RANGE NEUTRON FLUX MONITORS. ATTRIBUTED TO OPERATOR ERROR.	7A

TABLE 3-2

SHUTDOWN EVENTS AND SYSTEM 8-4
PREVENTION, DETECTION, AND MITIGATION FEATURES

LER. NO.	PLANT NAME	EVENT CATEGORY	DATE	EVENT SUMMARY	SYSTEM 8-4 PREVENTION, DETECTION, AND MITIGATING FEATURES (SEE TABLE 3-3)
368/90017	ARKANSAS NUC. ONE 2	FIRE PROTECTION	07/18/90	PERSONNEL ERROR RESULTS IN AN INOPERABLE FIRE BARRIER PENETRATION SEAL.	8A, 6B
261-91002	ROBINSON 2	FIRE PROTECTION	02/14/91	OVERLOADED ELECTRIC EXTENSION CORDS IGNITED COMBUSTIBLE MATERIALS DURING REFUELING OUTAGE. NO SAFETY RELATED EQUIPMENT IN FIRE AREA.	3A, 8B
368-91002	ARKANSAS NUC ONE 2	FIRE PROTECTION	02/23/91	FIRE IN EMERGENCY SAFETY FEATURES MOTOR CONTROL CENTER DURING HOT STANDBY DISABLED EQUIPMENT IN ONE ESF TRAIN. REDUNDANT SAFETY RELATED EQUIPMENT MAINTAINED SAFE SHUTDOWN CONDITION.	8A
344-91021	TROJAN 1	FIRE PROTECTION	05/24/91	UNABLE TO VERIFY THAT PLANT COULD BE PLACED INTO RHR CONDITION FOLLOWING WORST CASE FIRE BECAUSE NECESSARY DEPRESSURIZATION CONTROL SYSTEMS COULD NOT BE CREDITED.	8A
269-91008	OCONEE 1	FIRE PROTECTION	06/12/91	FOLLOWING CRAFT WORK AREA, FIRE BARRIER DOORS WERE LEFT OPEN, IN VIOLATION OF PROCEDURE.	8A
266-91007	POINT BEACH 1	FIRE PROTECTION	06/26/91	FOLLOWING WORK IN AREA, FIRE BARRIER SEALS IN EMERGENCY SAFETY FEATURES ROOMS WERE NOT SECURE IN VIOLATION OF PROCEDURE. POTENTIAL FOR LOSS OF ESF.	8A
423-91029	MILLSTONE 3	FIRE PROTECTION	11/21/91	INADEQUATE PROCEDURES DURING CONSTRUCTION RESULTED IN UNSEALED FIRE PORTECTION BARRIER BETWEEN RHR AND CONTAINMENT RECIRCULATION SPRAY CUBICLES IN ENGINEERED SAFETY FEATURES BUILDING.	8A

TABLE 1-2

SHUTDOWN EVENTS AND SYSTEM 80+
PREVENTION, DETECTION, AND MITIGATION FEATURES

LER. NO.	PLANT NAME	EVENT CATEGORY	DATE	EVENT SUMMARY	SYSTEM 80+ PREVENTION, DETECTION, AND MITIGATING FEATURES (SEE TABLE 1-1)
286-90008	INDIAN POINT 3	HEAVY LOADS AND FUEL HANDLING	/ /	PRIOR DAMAGE TO INTERNALS WHILE LIFTING LED TO STUCK FUEL ASSEMPLIES DURING REFUELING AND POTENTIAL RADIOACTIVE RELEASE DURING CONSEQUENT RETRIEVAL HANDLING.	9C
369-91016	MCGUIRE 1	HEAVY LOADS AND FUEL HANDLING	/ /	INADEQUATE PROCEDURE ALLOWED MISPLACEMENT OF SPENT FUEL CAUSING POTENTIAL FOR LOCAL OVERHEATING IN SPENT FUEL POOL.	10A
455-90008	BYRON 2	HEAVY LOADS AND FUEL HANDLING	09/29/90	INADEQUATE PROCEDURES LED TO A FUEL ASSEMBLY DROP ONTO TOP OF FUEL RACK IN SPENT FUEL POOL.	9C
316-90010	COOK 2	HEAVY LOADS AND FUEL HANDLING	10/05/90	DESIGN ERROR IN POLAR CRANE COULD LEAD TO FAILURE AT MUCH LOWER LOADS THAN SPECIFIED DESIGN.	9A, 9B
313-90015	ARKANSAS NUC. ONE 1	HEAVY LOADS AND FUEL HANDLING	10/26/90	INADEQUATE PROCEDURE LED TO LACK OF SURVEILLANCE FOR SPENT FUEL BUILDING VENTILATION SYSTEM AND POTENTIAL FOR RADIATION RELEASE SHOULD EVENT OCCUR.	10A
247/91004	INDIAN POINT 2	HEAVY LOADS AND FUEL HANDLING	02/12/91	SINGLE OVERLOAD EVENT OR CORROSION FATIGUE/STRESS CORROSION CRACKING CAUSES FAILURE OF HOLD DOWN BOLTS FOR POLAR CRANE RAIL. SUFFICIENT MARGIN EXISTED TO ACCEPT REDUCED BOLT CONFIGURATION.	9A, 9B
275/91014	DIABLO C.	HEAVY LOADS AND FUEL HANDLING	02/13/91	INADEQUATE PROCEDURE ALLOWED MOVEMENT OF HEAVY LOADS OVER FUEL STORAGE POOL WHILE ALL THREE DIESELS WERE INOPERABLE CAUSING POTENTIAL FOR RADIATION RELEASE UPON DROP EVENT WHEN VENTILATION UNAVAILABLE.	9A, 9B
498-91005	SOUTH TEXAS 1	HEAVY LOADS AND FUEL HANDLING	02/18/91	INADEQUATE PROCEDURE CONTROLS ALLOWED OPEN DOOR IN FUEL HANDLING BUILDING DURING FUEL MOVEMENT PROVIDING POTENTIAL FOR RADIATION RELEASE.	9A, 9B, 9C
382-91004	WATERFORD 3	HEAVY LOADS AND FUEL HANDLING	03/04/91	PERSONNEL ERROR OMITTED REQUIRED SURVEILLANCE OF FUEL HANDLING MACHINE PRIOR TO USE DURING REFUELING.	9C
NOTE (1)	PALISADES	HEAVY LOADS AND FUEL HANDLING	02/29/92	DURING REFUELING, WHILE LIFTING THE UPPER GUIDE STRUCTURE, A FUEL ASSEMBLY STUCK TO THE BOTTOM OF THE GUIDE STRUCTURE AND WAS SUSPENDED ABOVE THE CORE. CONTAINMENT INTEGRITY WAS VERIFIED AND NON-ESSENTIAL PERSONNEL WERE EVACUATED.	9C

(1) NO LICENSEE EVENT REPORT (LER) NUMBER AVAILABLE. THIS EVENT SUMMARY WAS OBTAINED FROM ANOTHER SOURCE OF INFORMATION.

TABLE 1-2

SHUTDOWN EVENTS AND SYSTEM BU+
PREVENTION, DETECTION, AND MITIGATION FEATURES

LER. NO.	PLANT NAME	EVENT CATEGORY	DATE	EVENT SUMMARY	SYSTEM BU+ PREVENTION, DETECTION, AND MITIGATING FEATURES (SEE TABLE 1-1)
NOTE (1)	TROJAN	MODE CHANGE EVENTS	08/18/82	BOTH TRAINS OF SI WERE BLOCKED FOR ABOUT 43 HOURS WHILE IN MODES 3 AND 4. WHILE IN MODE 5, BOTH TRAINS WERE BLOCKED DUE TO TESTING. THE PLANT STATUS PANEL INDICATED THIS CONDITION, HOWEVER, IT WAS NOT LOGGED AND PROPER PROCEDURES WERE NOT FOLLOWED FOR REMOVAL OF THE SYSTEM FROM SERVICE.	1M, 10A, 10B
NOTE (1)	NORTH ANNA 1	MODE CHANGE EVENTS	12/06/82	AN INADVERTENT AUTOMATIC SI OCCURRED WHICH BLOCKED ADDITIONAL SI ACTUATIONS UNTIL RESET BY CLOSING THE REACTOR TRIP BREAKERS. THE PROCEDURE USED TO RECOVER FROM THE SI ACTUATION, IMPLIED IT WAS ACCEPTABLE TO LEAVE THE SI BLOCKED FOR UP TO 30 HOURS WHICH IS NOT CORRECT. THE SI WAS UNBLOCKED ABOUT 22 H[illegible]RS LATER.	1M, 10A, 10B
NOTE (1)	OCONEE 3	MODE CHANGE EVENTS	04/19/87	HEATING UP, BREAKERS FOR HPI SUCTION VALVES FROM BWST WERE LEFT TAGGED OUT. THIS IS A TECHNICAL SPECIFICATION VIOLATION [illegible]USE TWO FLOW PATHS TO HPI ARE REQUIRED. CAU[illegible]S FAILURE TO REVIEW TAGOUT LOGBOOK AND POOR COMMUNICATION BETWEEN OPERATIONS STAFF AND SUPPORT.	1M, 10A, 10B
454/91005	BYRON 1	MODE CHANGE EVENTS	10/27/91	PERSONNEL ERROR RESULTS IN PLANT ENTERING HOT SHUTDOWN (MODE 4) FROM COLD SHUTDOWN (MODE 5) WITH INOPERABLE CONTAINMENT SPRAY SYSTEM AND ECCS.	10A, 10B

(1) NO LICENSEE EVENT REPORT (LER) NUMBER AVAILABLE. THIS EVENT SUMMARY WAS OBTAINED FROM ANOTHER SOURCE OF INFORMATION.

2.0 SHUTDOWN RISK ISSUES

Sections 2.1 through 2.13 present detailed evaluations of specific shutdown issues that were identified at the December 18, 1991 meeting with the NRC and that are listed in Reference 2. Each section is subdivided into four subsections. The first subsection states the issue consistent with the interpretation and evaluation in NUREG-1449 and the appropriate RAIs. The second subsection lists the acceptance criteria that are employed to evaluate the System 80+ design to prevent and/or mitigate unacceptable consequences related to each shutdown issue. The third subsection discusses the postulated plant scenarios, the analyses and the evaluations considered by ABB to assure that the shutdown issue is adequately addressed. Finally, the fourth subsection states how the issue is resolved by the System 80+ design. Depending upon each issue and its significance in evaluating System 80+, the content of these subsections varies. Where appropriate, reference is made to RAI's on the issue, both outstanding and previously submitted. Appendix A contains the responses to all the RAI's.

2.1 PROCEDURES

2.1.1 ISSUE

The operational guidance provided by the plant designer to the owner/operator might not be sufficient to insure that procedures to avoid, detect, mitigate, and/or recover from abnormal events initiated from shutdown operations can be developed by the plant owner/operator.

2.1.2 ACCEPTANCE CRITERIA

The operational guidance provided by the plant designer to the owner/operator shall be sufficient to properly utilize design features that are available to detect, mitigate and assist recovery from abnormal events initiated during shutdown operations.

2.1.3 DISCUSSION

The System 80+ design incorporates advanced features which promote safer and simpler plant operation. The features include redundancy and diversity of components and systems, dedicated and/or permanently aligned systems, and an advanced information system which better informs the operations staff of plant status, potential adverse system interactions, and available recovery paths if an abnormal event occurs. These features also contribute to improved operability and maintainability that should significantly reduce the initiating situations that have contributed to increased shutdown risk.

The plant owner/operator is responsible for preparing detailed procedures for normal, abnormal, and emergency operations using guidance developed by the plant designer and plant site specific information. The plant designer's guidance generally is in the form of suggested operational sequences that preserve the safety bases of the design. Since shutdown operations must be intimately connected to an outage strategy, specific procedures cannot be imposed by the plant designer to cover the array of possible shutdown events. However, the plant designer can provide guides which instruct the plant owner-operator in the use of design features which can detect, mitigate, and assist recovery from abnormal events that can occur during shutdown operations.

Discussion and explanation of the procedural guidance that is appropriate to the shutdown modes and that will be provided to the owner operator will be presented in the June 15, 1992 updated submittal of this report.

2.1.4 RESOLUTION

The issue of procedures for shutdown operation is resolved for System 80+ by providing operational guidance to address use of advanced design features to detect, mitigate, and assist recovery from abnormal events initiated from shutdown operations. The information that will be provided to the owner/operator as operational guidance during shutdown modes will be presented in the June 15, 1992 updated submittal of this report.

2.2 TECHNICAL SPECIFICATION IMPROVEMENTS

2.2.1 ISSUE

When a plant is operated within the limiting conditions for operation provided by the technical specifications, the consequences of design basis events should be bounded by the results of the safety analyses. However, limiting conditions for operation developed for power operation might not be sufficient to insure that the consequences of events initiated from shutdown modes are bounded by the analyses. Technical specification should include the necessary limiting conditions for operation that are applicable to shutdown modes.

2.2.2 ACCEPTANCE CRITERIA

Technical specification shall insure that when the plant is operated within the limiting conditions for operation applicable to the mode of operation, consequences of design basis events shall be bounded by the results of safety analyses for that mode.

2.2.3 DISCUSSION

The System 80+ design incorporates advanced features which promote safer operation and greater margins to operating limits. The features include redundancy and diversity of components and systems and an advanced information system which better informs the operations staff of plant status, potential adverse system interactions, and available recovery paths if an abnormal event occurs. These features also contribute to improved operability and maintainability that should significantly reduce the initiating situations that have contributed to increased shutdown risk.

One objective of the plant designer is to reduce the operational constraints that limit the plant owner's flexibility to operate the plant as efficiently as possible. Another objective is to formally impose the operational constraints required to insure the plant remains within analyzed bounds for operation through the initial set of technical specifications. Overly restrictive technical specifications especially for shutdown modes may unnecessarily complicate operations and may increase risks by prolonging the shutdown period and adding to staff stress. The objective of this assessment of shutdown risk for the System 80+ relative to technical specifications is to modify existing technical specifications to the extent necessary to address event initiators not fully covered by analysis of the traditional design basis events. Discussion and explanation of proposed changes or additions to the technical specifications will be presented in the June 15, 1992 updated submittal of this report.

2.2.4 RESOLUTION

The issue of shutdown specific technical specifications is resolved for System 80+ by modifications and additions to the technical specifications based upon safety analyses performed for modes 2 through 6. These changes will be presented in the June 15, 1992 updated submittal of this report.

2.3 REDUCED INVENTORY OPERATION AND GL 88-17 FIXES

2.3.1 ISSUE

The NRC has voiced increasing concern over the safety of operations during plant shutdowns. Plant events which have occurred in the industry have highlighted the need for a close examination of operations during reduced inventory conditions in the reactor coolant system. Following the Diablo Canyon incident, the NRC published Generic Letter 88-17, which required that holders of operating licenses or construction permits address a number of deficiencies in order to enhance the safety of shutdown operations and reduce the risk to the public. Specific areas of concern include:

1. containment closure issues,
2. instrumentation which would greatly improve the operator's monitoring capability during reduced inventory operations,
3. alternate ways to add inventory to keep the core covered should SCS be lost,
4. administrative procedures that would avoid RCS perturbations during reduced inventory operations, and
5. nozzle dam installation procedures which would ensure a vent pathway is available so that RCS pressurization can be minimized if shutdown cooling is lost.

The NRC has specified that programmed enhancements should accomplish a comprehensive improvement in the plant's ability to cope with shutdown operations. The NRC asserted that plants are not well designed for reduced inventory operations, that procedures are incomplete for shutdown cooling recovery or alternate actions and that mitigating features may not be available under shutdown conditions. Therefore licensees should implement means to prevent accident initiation, to monitor a progression that may lead to core damage and to evaluate consequences and, where needed, to provide mitigation.

2.3.2 ACCEPTANCE CRITERIA

The System 80+ design shall reflect a comprehensive consideration of shutdown and lower power risk, by adequately addressing all GL 88-17 recommendations and other issues relevant to reduced inventory, especially in the areas of instrumentation, technical specifications, procedures, equipment availability and analyses.

2.3.3 DISCUSSION

During plant shutdowns, certain maintenance and testing activities require a draindown of the RCS to a partially filled condition. Normal maintenance activities include the replacement of RCP seals and journal bearings. A testing activity requiring RCS draindown is the Technical Specification for inservice inspection of the steam generator tubes. The use of nozzle dams during maintenance and testing activities minimizes the time during which the RCS must be operated in a partially filled condition. To minimize operating time at mid-loop level, nozzle dams are installed on the steam generators and the RCS is reflooded to continue maintenance and testing.

While the RCS coolant level is lowered to within the hot leg, the risk of loosing shutdown cooling is increased due to the possibility of vortexing at the SCS suction line interface with the hot leg. In the worst scenario, subsequent to vortexing in the SCS suction line, a large percentage of air is entrained into the SCS suction piping, shutdown cooling pump cavitation occurs, and the SCS pump is damaged or lost entirely. If SCS operation is not reestablished, core boiling and pressurization can produce very rapid core uncovery, sometimes in as little as 15 to 20 minutes. This phenomenon, and the high probability of it occurring, prompted the NRC to issue the recommendations of GL 88-17.

The System 80+ design features will realize practical and significant benefits during reduced inventory operations. These design features are outlined in Sections 2.3.3.1 through 2.3.3.5 which follow. Details of the capabilities of these System 80+ design features to enhance safety during reduced inventory conditions and of the analytical bases for changes to Technical Specifications and procedure guidance to the owner/operator will be presented in the June 15, 1992 updated submittal of this report.

2.3.3.1 <u>Instrumentation for Shutdown Operations</u>

Diverse, accurate, and redundant instrumentation, including control room CRT displays, gives continuous system status, and provide the operations staff with precise information to respond to loss of shutdown cooling events, should they occur (See also Section 2.8 of this Report). Phenomena which can affect instrumentation operation are considered. Analyses form the basis for instrument design, calibration, and operation so as to assure correct instrument operability during reduced inventory states. Instrumentation availability during shutdown will be assured via the plant Technical Specifications that willbe provided in Section 2.2 of the final submittal of this report.

The types of instrumentation used are outlined below.

1. Independent wide and narrow range level sensors are provided for continuous monitoring of RCS level during draindown operations. The level indicators provide monitoring capability from the pre-drain down normal level in the pressurizer to a point lower than that required for SCS operation. The level indicators are calibrated for low temperature operation and they provide a high degree of accuracy. Indication in the main control room and low and high level alarms are provided.

   The wide range level instrument covers draining from the pressurizer to the bottom of the hot leg and will be available with the head on and off the vessel. The narrow range level instrument covers reduced inventory operations. Two instrument types are being considered for level indication during reduced inventory operations. The final instrument design will be presented in the June 15, 1992 submittal. Either of the two instrument types being considered is accurate for measuring level within the hot leg. During a draindown, level monitoring would be transitioned from the wide range level instrument to the narrow range instrument when the greatest degree of accuracy is required during operations with level within the hot leg region.

2. Several independent diverse temperature measurements representative of core exit temperature are provided during reduced inventory operations. Temperature indication is available when the head is located both on and off the vessel. Since temperature is valuable in guiding SCS restoration actions and in monitoring the success of recovery actions, alarm setpoints are based on integrated response times necessary to support SCS recovery, event mitigation, time to boil, and containment closure.

3. SCS operation monitoring instrumentation is provided that assures precise knowledge of the status of the operating SCS loop; including pressure, temperature, flow and pump performance indications.

2.3.3.2 SCS Design

The functional design of the Shutdown Cooling System (SCS) is substantially complete for System 80+. Some of the design features that improve the performance during shutdown operation are listed below. These features will be evaluated in the context of the results pending from activities described in Section 2.4 on decay heat removal and other sections of this report and will be presented in the final submittal of this report.

1. The System 80+ SCS suction lines do not contain any loop seals. An improved suction piping layout allows self venting (in as much as possible). Entrained air travels back up to the hot

leg without the possibility of being trapped anywhere in the SCS suction line. This feature allows the SCS pumps to be restarted without requiring complicated venting procedures, assuring flooded suction conditions at the shutdown cooling pump.

2. The two SCS suction lines are independent and redundant to each other. Problems associated with a specific suction line would not limit the other shutdown cooling train from being operated, after level recovery (if necessary), for continued decay heat removal.

3. The two containment spray system pumps are interchangeable with the SCS pumps and are designed to back up the SCS pumps in the event of a non-electrical pump failure. Thus, there are four pumps available for shutdown cooling provided support systems are available. Plant Technical Specifications will assure pump availability during shutdown operations.

4. There are no interlocks on the shutdown cooling suction piping which have the potential for disturbing shutdown cooling. Although previous designs (e.g., System 80) included interlocks to isolate the SCS in the event of an unanticipated RCS pressurization during shutdown cooling, this interlock has been deleted from the System 80+ design per the EPRI ALWR Utility Requirements Document. This reasonably reduces the likelihood of losses of SCS.

2.3.3.3 Steam Generator Nozzle Dam Integrity

The System 80+ design addresses the NRC concern for preventing pressurization in the upper plenum of the reactor vessel during core boiling scenarios. In the System 80+ design, the ability of the RCS to withstand pressurization during reduced inventory operations with the nozzle dams installed is limited by the design pressure of the nozzle dams. Based on field hydrostatic tests performed on nozzle dams, a conservative value of 40 psia is assumed for this pressure limit. In order to assure that the nozzle dam design pressure is not exceeded during reduced inventory operations with boiling conditions in the reactor vessel, the System 80+ design includes two blind-flanged connections off the primary safety valve inlet piping to be used as mid loop operation vent pathways. These connections, when opened to the containment atmosphere, provide sufficient vent capacity to prevent RCS pressurization and subsequent nozzle dam failure. These vents will be opened prior to draining the RCS for reduced inventory operations and will not be closed until after the final RCS refill.

Analyses have shown that two 6 inch vents connected to the pressurizer safety valve piping and relieving to the pressurizer cubicle will be sufficient for venting the RCS during RCS boiling.

These vents will allow an equilibrium pressure of approximately 70% of nozzle dam design pressure at 4 days after shutdown. The final analysis will provide details that will define the equilibrium pressure versus days after shutdown for the conservative case where no steam generators are available for reflux boiling. This data will be incorporated into guidance for the owner/operator to employ when planning outage evolutions. The guidance will address the safety and risk impacts of nozzle dam installation timing and will be provided in Section 2.1 of the final submittal of this report.

Based on operating plant data, the time required to achieve the mid-loop condition from full power operation varies, depending on planned outage activities. Typically, a minimum for four days is required to cool down and then drain down the RCS from a full power condition. As a result of the analyses performed for Section 2.4 of this report, a Technical Specification regarding the earliest time after shutdown for entry to reduced inventory operations will be provided in Section 2.2 of the final submittal of this report. Such restrictions could minimize the consequences of a loss of shutdown cooling event during reduced inventory operations.

If multiple operator errors result in significant reactor vessel head pressurization, vessel water could be displaced into the cold legs. In operating plants, this situation has been postulated to have it's greatest negative effect during RCP seal changeout, since it can lead to inventory loss thru the RCP. The System 80+ RCP design includes a shaft stop seal which uses the weight of the shaft and impeller to effect a seal between the RCP shaft and casing when uncoupled from the motor during seal replacement. During seal replacement, when RCS fluid is present in the cold leg, any displaced inventory (and subsequent spillage) caused from manometric effects due to vessel head pressurization is minimized.

2.3.3.4 Alternate Inventory Additions and DHR Methods

The effective management of time and efforts is crucial to coping with a loss of shutdown cooling. Awareness of time constraints provides information that is useful in deciding how to allocate effort. If shutdown cooling cannot be restored within the time to core uncovery, getting a source of water lined up to keep the core covered becomes a first priority. Inventory makeup directly extends the margin of safety prior to uncovering the core.

The recommended plan for coping with a loss of shutdown cooling includes the following steps:

1. verify the proper functioning of level indications

2. verify reactor coolant pressure boundary integrity

3. establish the availability of makeup sources

4. if no makeup sources are available, remove RCS decay heat via reflux boiling, if steam generators are not isolated.

The System 80+ design ensures that adequate operable equipment is available for mitigation of the effects of a total, sustained loss of shutdown cooling. Mitigating actions, such as operating redundant shutdown cooling pumps and containment spray pumps, are being developed and will be provided in the final submittal of this report. Safety injection system pumps will be employed for those scenarios where RCS pressure is too high for gravity feed.

2.3.3.5 Operations

Procedural and Technical Specification changes necessary to support the program will be identified and implemented into the plant design basis. Procedural guidance for the conduct of mid-loop draindowns will be provided to assure that no testing or maintenance activity adversely affects the NSSS during mid-loop operations. This guidance will be reported in Section 2.1 of the final submittal of this report.

Restrictions on maintenance and testing activities while the RCS is in a mid-loop or a reduced inventory condition are provided in plant Technical Specifications. The outage schedule is derived from Technical Specification Requirements, necessary plant maintenance and testing activities, and plant and equipment conditions required for these activities. Procedural guidance will specify that the plant operator suspend any activities which could alter the NSSS state or result in losing shutdown cooling during reduced inventory operations. This guidance will be provided in Section 2.1 of the final submittal of this report.

Due to the Diablo Canyon incident and other industry events, operations in Mode 5 will be evaluated, including the requirements for evacuation of personnel from the containment building, closing of the containment equipment hatch and containment air lock doors, and isolation of penetrations to the outside atmosphere, as appropriate, based on time to boil and time to core uncovery criteria. A description of the containment closure conditions referred to, along with a description of containment closure design features, is contained in other Sections of this report. This evaluation and description will be included in Section 2.5 of the final submittal of this report.

Appropriate operating and emergency response guidance directs the operator in the proper conduct of reduced inventory operations. Training guidance aids in operator detection of abnormal conditions, and mitigation sequences. This guidance will be provided in Section 2.1 of the final submittal of this report.

2.3.4 RESOLUTION

The resolution of the reduced inventory issue on System 80+ will comprise the results of the analyses outlined above, related evaluations in Section 2.4 on availability of decay heat removal, Technical Specifications in Section 2.2 and procedural guidance in Section 2.1. These will be integrated and focused on the reduced inventory issue in the June 15, 1992 updated submittal of this report.

2.4 LOSS OF DECAY HEAT REMOVAL CAPABILITY

2.4.1 ISSUE

Events that have occurred at operating plants demonstrate the vulnerability during shutdown Modes to loss of decay heat removal. The variety of maintenance activities taking place at shutdown combined with the possible system and equipment interactions that may occur lead to many conceivable scenarios for experiencing a loss of decay heat removal. Three dominant design objectives have evolved from the emphasis placed on prevention of shutdown events:

1. Provide redundant Shutdown Cooling System capacity and identify alternate decay heat removal capability.
2. Provide instrumentation to effectively monitor shutdown operations, including critical plant configurations such as mid-loop.
3. Provide flexible redundancy in AC power.

The System 80+ features that address these issues are presented below in the context of demonstrating an integrated design capable of avoiding unacceptable consequences from the entire spectrum of potential event scenarios.

2.4.2 ACCEPTANCE CRITERIA

All event scenarios may be characterized by initiation, detection, mitigation and consequence. To measure the success of the integrated response of System 80+ to events initiated from Modes 2 through 6, two criteria related to the potential for radiological release are adopted here. Significant release can only occur from fuel cladding rupture resulting from heatup after the coolant level drops below the top of the active core. Therefore, the first acceptance criterion is that there shall be no fuel cladding failure resulting from postulated events, excluding LOCA, initiated from Modes 2 through 6. The second criterion is that the radiological exposure of the public to events resulting in the loss of decay heat removal shall be limited to a fraction of the 10CFR100 limits that will be specified in the final submittal of this report.

2.4.3 DISCUSSION

In this section, an evaluation is presented of the System -80+ features that are designed to prevent violation of the above criterion. Section 2.4.3.1 examines events and event initiators which potentially result in loss of shutdown cooling leading to boiling. Causes of past events considered include mid-loop operation, power failure and operator error. Thermal-hydraulic analyses specific to the System 80+ configurations confirm the flexibility afforded the operator to mitigate these events. Appropriate Technical Specification limitations and procedural guidance are identified by the analyses and will be provided in the final submittal of this report.

Section 2.4.3.2 presents the features of System 80+ which help prevent a loss of decay heat removal due to the loss of AC power. This is one of the specific concerns identified in NUREG-1410. The discussion in this section is directly related to the means of coping with a loss of decay heat removal. This concern is also identified in NUREG-1410 and evaluated in Section 2.4.3.1.

Section 2.4.3.3 presents the features of System 80+ that help assure the availability of the diesel generator. This issue was also identified in NUREG-1410. Availability of the diesel generator has been a significant factor in numerous past events.

Taken together, these sections demonstrate the integrated capability of the System 80+ to prevent and mitigate a loss of decay heat removal to ensure that the acceptance criteria are not violated.

2.4.3.1 Shutdown Event Initiation and Analyses

This section will present event initiators and the means of prevention and mitigation on System 80+. Thermal-hydraulic analyses will evaluate scenario time sequences and identify available action times. Results will be presented in the June 15, 1992 updated submittal of this report.

2.4.3.2 System 80+ AC Power Reliability

2.4.3.2.1 Introduction

This section presents the System 80+ features that assure the availability of electrical power to supply the Class 1E buses and the capability to restore power if the electrical source is interrupted. The electrical distribution system provides redundant and diverse sources of power to the Class 1E buses during shutdown modes and reduced inventory in the reactor coolant system and provides redundancy and flexibility to insure re-energizing the Class 1E buses is possible if power is interrupted.

2.4.3.2.2 Discussion

Electrical power sources need to be carefully managed during shutdown operations to maintain a desired level of safety. This is especially true during reduced inventory operations. Reduced inventory requires heightened awareness to manage the risks of maintaining an electrical source to the Class 1E buses and of insuring an alternate source is available. The potential for a complete loss of decay heat removal due to the loss of electrical power is lowered when the electrical supply requirements for shutdown modes and reduced inventory are managed properly.

The management and operation of these electrical sources will be guided by Technical Specifications for shutdown operations and reduced inventory. Technical Specifications will be written to identify the minimum acceptable electrical distribution system alignments for operating in shutdown modes and reduced inventory. The operation of the electrical distribution system during shutdown modes and reduced inventory can be guided by procedures for normal alignments and for aligning alternate electrical sources if normal sources are interrupted.

The electrical distribution system design will provide flexibility and redundancy to allow for the management of competing priorities during shutdown. These competing priorities include the need to perform maintenance on electrical system equipment versus the need to have electrical sources available to provide power to the Class 1E buses.

The System 80+ electrical system design (see Figure 2.4-1) provides the redundancy and flexibility to insure the risks associated with shutdown modes and reduced inventory operations are lowered to acceptable levels. This is accomplished by providing two independent divisions of AC Electrical Power. Each division has two 4.16 KW Safety Buses with three sources of electrical power. Two of these are Class 1E sources. The two Class 1E sources are: (1) Normal - the Safety System Transformer being powered from the Switchyard Interface II, and (2) Emergency - the Diesel Generator. The third source is a backup from the Permanent Non-Safety (PNS) Bus.

This backup source (PNS-Bus) of power to the Safety Bus has three sources of electrical power. The three sources are: (1) Normal - The division related Unit Auxiliary Transformer (UAT) being powered from Switchyard Interface I through the Unit Main Transformer (UMT), (2) Alternate - The opposite divisions UAT being powered from Switchyard Interface I through the UMT, and (3) Backup - the Combustion Turbine.

Therefore, the Class 1E Safety Buses have the potential to be fed from four different ultimate sources during shutdown modes and reduced inventory operations. These sources are:

1. Switchyard Interface I,
2. Switchyard Interface II,
3. Diesel Generator, and
4. Combustion Turbine.

This distribution system provides the shutdown management team with the flexibility to perform shutdown activities on one source of power to a division 4.16 KV Safety Bus and still maintain three diverse sources of reliable electrical power to the 4.16 KV Safety Bus.

Along with the electrical system design features, the System 80+ Technical Specifications include shutdown modes and reduced inventory operation Limiting Conditions for Operations (LCOs). The LCOs provide minimum acceptable electrical distribution alignments. Guidance is also provided by procedure to the operation staff to insure available source alignments are identified whenever shutdown activities are in progress. Additional procedural guidance is provided for aligning any available source(s) to the Safety Bus(es) if power to the bus(es) is interrupted. The procedure guidance and Technical Specifications will be provided in Sections 2.1 and 2.2 of the final submittal of this report.

2.4.3.2.3 Conclusion

The System 80+ electrical distribution system design features provide the necessary redundancy, flexibility, and diversity to reduce the likelihood of losing decay heat removal due to a loss of electrical power. The features of the design, the Technical Specifications, and the procedure guidance allow shutdown activities within certain limits and provide operational guidance for system flexibility and assurance that a loss of the decay heat removal is unlikely.

2.4.3.3 <u>System 80+ Diesel Generator Availability</u>

2.4.3.3.1 Introduction

The availability of the Diesel Generator and the Diesel Loading Sequencer to automatically start and load during shutdown modes of operation is one of the issues identified in NUREG-1410. The availability of Diesel Generator instrumentation and control system to provide reliable indications and automatic trip signals for Diesel Generator protection during emergency operation (e.g. automatic start while in shutdown modes) and the availability of adequate information and indications to identify, diagnose, and correct Diesel Generator operational problems are significant to the overall maintenance of decay heat removal as presented in Section 2.4.3.1.

The Diesel Generator (D/G) and Diesel Load Sequencer (DLS) provide emergency power to the Class 1E buses during shutdown modes of operation with the same methods used during power modes of operation. The Instrumentation and Control (I&C) system for the D/G provides signals to start the diesel for emergency operation, applicable protective trips to prevent or limit damage to the D/G at all times and D/G status to the Control Room and to the local control panel. This status includes trip signals (alarms, indications and recordings), parameter indications, and alarms for abnormal parameters. Also, controls for starting, stopping, synchronizing, and loading the D/G are provided in the Control Room and at the local control panel.

2.4.3.3.2 Discussion

The Diesel Generator (D/G) and Diesel Load Sequencer need to maintain a consistent means of operation independent of the plant operation condition. This ensures the operating staff is not required to learn different operating schemes and therefore reduces potent [illegible] error.

The [illegible] and DLS provides this simplicity of operation. The [illegible] rgency source of power to the Class 1E bus. The D/G [illegible] are available for operation during shutdown condi[illegible] undergoing maintenance. The Class 1E buses are monit[illegible]dervoltage and degraded voltage conditions. If either [illegible]n is sensed, the D/G is started and the DLS is initiated (see attached CESSAR-DC Figure 7.3-5). For a loss of power to the Class 1E bus, the response of the D/G and DLS is not dependent on plant operational modes. Therefore, the response of the System 80+ equipment provides the operator with the same parameters and indication to be monitored whether shutdown or operated at power. This design characteristic provides a basis for consistency in operating procedures and operator training. This eliminates the necessity of two sets of procedures dependent on plant operating conditions. It also eliminates extra required training for the operation staff. (Detail on the Emergency Diesel Generators can be found in CESSAR-DC Section 8.3.1.1.4).

The D/G I&C system needs to ensure the diesel is protected during all modes of operation. However, certain protective trips need to be bypassed during emergency operation.

The System 80+ Diesel Generator protection system provides automatic trips to prevent or limit damage to the D/G. The protection trips provided during emergency operation are:

1. Engine Overspeed,
2. Generator Differential Protection,
3. Low-Low Lube Oil Pressure, and
4. Generator Voltage - Controlled Overcurrent.

These trips are provided in accordance with Reg. Guide 1.9 Position 7. All other trips are bypassed during emergency operation. (See CESSAR-DC Section 8.3.1.1.4.4 for a complete description of trips bypassed during emergency operation). The protection circuitry is dependent on the initiating signal and not dependent on plant operational modes. The sensing of an undervoltage or degraded voltage condition during shutdown causes an automatic D/G start, activates the protective circuitry, and bypasses all non-emergency trips. This circuitry allows for consistency in the operational response to an emergency start of the D/G independent of plant operating mode.

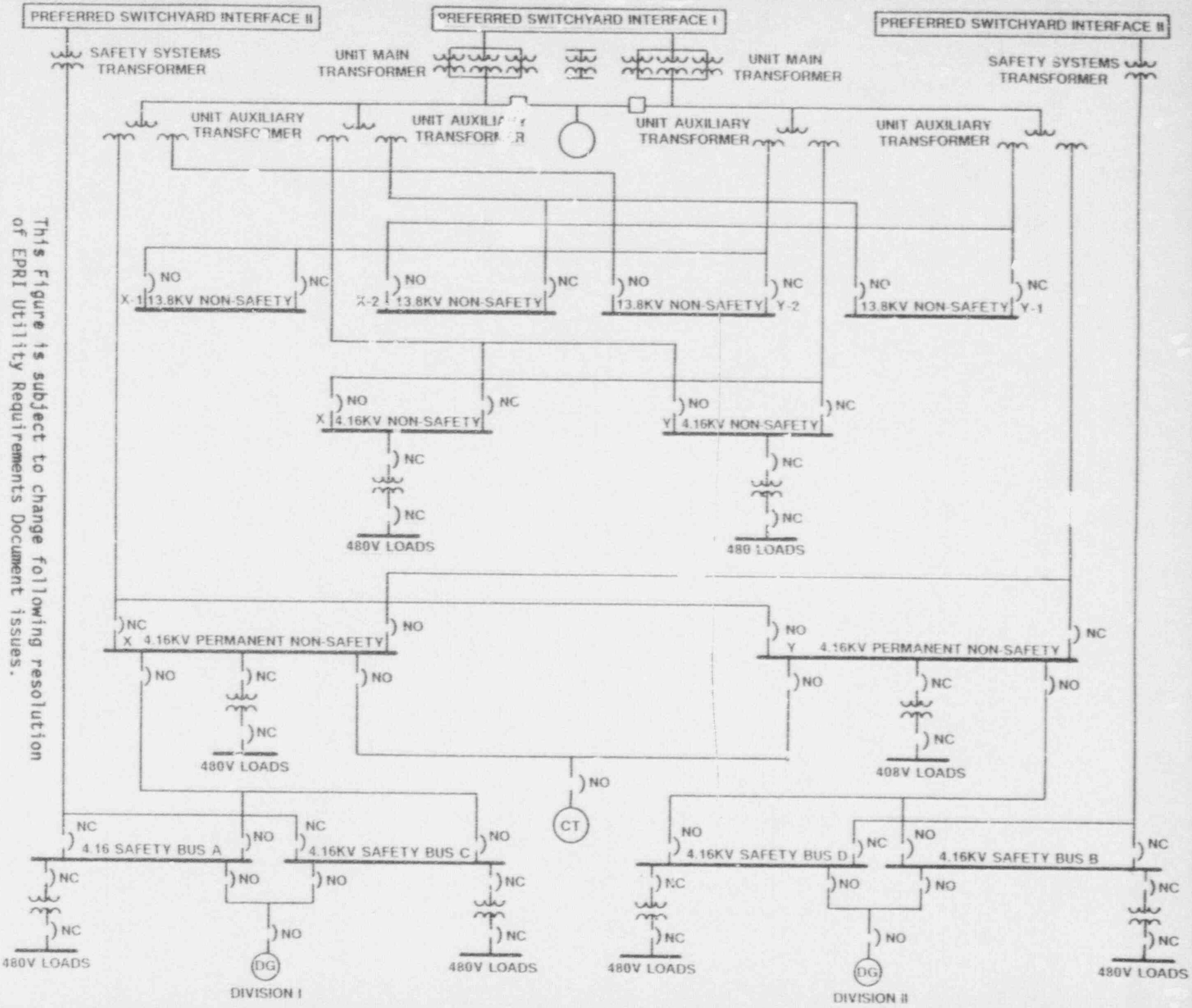
The I&C system needs to ensure the operator is informed of the D/G's operational status. This status includes parameter indications and alarms. The I&C systems need to provide controls to allow the operator to start and load the diesel to provide power to the Class 1E buses. This status and control scheme needs to be provided locally and in the control room.

The System 80+ control room is designated as the Nuplex 80+ Advanced Control Complex (ACC). The Nuplex 80+ ACC presents the operator with the information and controls necessary to complete any tasks identified in a task analysis process. The task analysis for D/G operation identifies the parameters, alarms, and controls required to operate the D/G from the Nuplex 80+ ACC. This identified status and control scheme is presented to the Control Room Operator on the Electrical Distribution Auxiliary Console. The presentation of this information is accomplished in accordance with a structural and hierarchial format discussed in CESSAR-DC Section 18.7.1. This formatting provides the operator with parameter displays, alarm status, alarm categorization, and alarm priority. This method of information presentation provides the Control Room Operator (CRO) with the tools necessary to monitor and/or diagnose D/G status.

The System 80+ local control panels for the D/G provides the Plant Equipment Operator with the same information and controls as is available to the CRO. The D/G status information and control scheme on the local control panel utilizes the same Man-Machine Interface (MMI) features used in the Nuplex 80+ ACC. These features meet the System 80+ human factors standards and guidelines.

2.4.3.3.3 Conclusion

The System 80+ Diesel Generator instrumentation and control systems design features provide starting signals for the D/G and DLS initiation and protective trip signals for D/G emergency operation and provide D/G status information to the control room and local control panel which allows the operator to operate, monitor, and diagnose D/G and DLS operation. These features of the System 80+ design enhance the operator's interface with the emergency equipment and reduces the potential of human error.

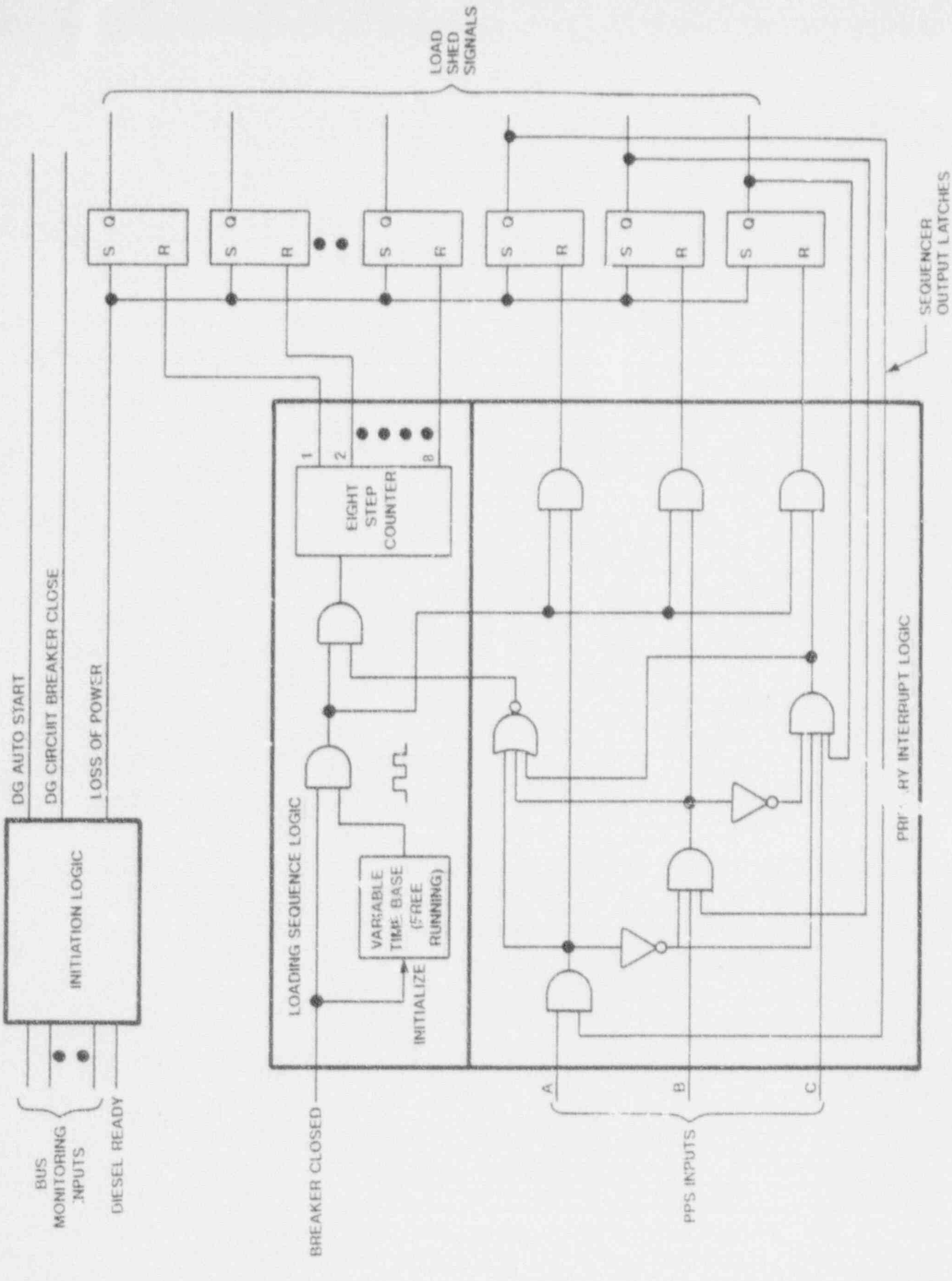


This Figure is subject to change following resolution of EPRI Utility Requirements Document issues.

SYSTEM 80+™

SYSTEM 80+ ELECTRICAL DISTRIBUTION

Figure 2.4-1



SYSTEM 80+™

DIESEL LOAD SEQUENCE SIMPLIFIED LOGIC DIAGRAM

Figure 2.4-2

2.5 PRIMARY/SECONDARY CONTAINMENT CAPABILITY AND SOURCE TERM

2.5.1 ISSUE

This section addresses the ability of the containment to protect the public from the consequences of a release of radiation during the time the containment is open.

This issue is related to events initiated in Modes 5 or 6 which have the potential for radiological release. The events considered are the loss of decay heat removal capability initiated by either a loss of shutdown cooling or by a loss of coolant accident caused by either a pipe break or operator error.

Following a loss of decay heat removal that is not the result of a pipe break, a radiological release from an open containment can occur when the time for the core to reach saturation is less than the time to restore RCS cooling and, failing this, the additional time to evacuate, close and isolate the containment. The time for the coolant to reach saturation is a function of plant conditions at the time the event is initiated.

Time to restore includes the time to detect that decay heat removal has been lost plus the time either to restore shutdown cooling or to initiate alternative means of cooling. Time to detect depends on the instrumentation available to detect that primary system cooling has been lost. The time to restore decay heat removal depends on the available systems and procedures.

Once primary system cooling has been lost, measures must be taken to evacuate and seal the containment before the system begins to boil. The time to close and isolate the containment depends on:

- Design, operation, condition and status of equipment to close penetrations, equipment hatches and personnel air-locks;
- Procedures for routing material and lines through these openings;
- Training of personnel; and
- Conditions of pressure, temperature and radiation within the containment as the core uncovers.

2.5.2 ACCEPTANCE CRITERIA

The following acceptance criteria apply to the issue addressed in this section:

1. Radiological exposure of the public to any event resulting in a loss of decay heat removal, shall be limited to a fraction of the 10CFR100 limits that will be specified in the final submittal of this report.

2. Radiological exposure of the public to any event resulting in a pipe break shall be limited to the limit stated in 10CFR100.

2.5.3 DISCUSSION

Evaluations of postulated initiators, event progression and timing, mitigating equipment, operator actions and source terms are in progress and will be presented in the June 15, 1992 updated submittal of this report.

2.5.4 RESOLUTION

The resolution of the open containment radiological release issue for System 80+ will be presented in the June 15, 1992 updated submittal of this report.

2.6 RAPID BORON DILUTIONS

2.6.1 ISSUE

The issues of the rapid boron dilution can be broken down into three categories as follows:

1. The introduction of deborated water into the RCS via Shutdown Cooling System (SCS), which flows into the RCS through the Direct Vessel Injection (DVI) lines, during maintenance of inline components.

2. Introduction of a water slug into the RCS during startup or refueling operations, including a specific example from NUREG-1449 (Reference 3). In that example, a loss of offsite power has occurred and the charging pumps are returned on line, powered by the Emergency Diesel Generators. If the plant were in startup mode - i.e., deboration in progress - the charging pumps could continue to operate causing a "slug" of unborated water to collect in the lower plenum of the reactor vessel. If it is then assumed that offsite power is restored and the RCP's are restarted, then a water slug of deborated water can be injected into the core.

3. A potential boron dilution resulting from inleakage from the secondary side of a steam generator during a SGTR event.

All the above issues will be addressed in the discussion and resolution sections of this report.

2.6.2 ACCEPTANCE CRITERIA

The acceptance criteria for the rapid boron dilution event should be consistent with the acceptance criteria that are necessary to meet the relevant requirements of GDC 10, 15 and 26. Specifically, these criteria are as follows:

1. Pressure in the reactor coolant and main steam systems should be maintained below the RCS P/T limits (see Figure 3.4.3-1 of Technical Specification 3.4.3) or below 110% of the design value, whichever is less.

2. Fuel cladding integrity shall be maintained by ensuring that the minimum DNBR remains above the 95/95 DNBR limit for PWRs and CPR remains above the MCPR safety limit for BWRs based on acceptable correlations (see SRP Section 4.4).

3. An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

4. An incident of moderate frequency in combination with any single active component failure, or single operator error, shall be considered and is an event for which an estimate of the number of potential fuel failures shall be provided for radiological dose calculations. For such accidents, the number of fuel failures must be assumed for all rods for which the DNBR or CPR falls below those values cited above for cladding integrity unless it can be shown, based on an acceptable fuel damage model (see SRP Section 4.2), that fewer failures occur. There shall be no loss of function of any fission product barrier other than the fuel cladding.

The above criteria are the same requirements as the acceptance criteria for the Inadvertent Boron Dilution (IBD) event as stated in NUREG-0800 Section 15.4.6, Reference 6, with the exception of Item 5. This criteria states that the available operator action time be 30 minutes for an IBD event during refueling conditions and 15 minutes for startup, cold shutdown and power operation. This requirement is not applicable to a "rapid" boron dilution event.

2.6.3 DISCUSSION

Analyses of the rapid boron dilution event initiated from a shutdown mode are in progress and will be presented in the June 15, 1992 updated submittal of this report.

2.6.4 RESOLUTION

The resolution of the rapid boron dilution issue will be presented in the June 15, 1992 updated submittal of this report.

2.7 FIRE PROTECTION

2.7.1 ISSUE

The risk of fire during shutdown operations is higher than when the plant is in power operation. This increase in risk is due to the presence of transient combustibles and ignition sources such as welding, grinding, and cutting operations necessary to support shutdown maintenance activities. Another risk is the reduced level of fire protection for systems such as the shutdown cooling and fuel pool cooling systems when the plant is in a shutdown mode, resulting in a higher susceptability of failure due to fire.

2.7.2 ACCEPTANCE CRITERIA

A defense in depth philosophy shall be employed in the design of the fire protection system in order to reduce the overall shutdown risk due to fire. The elements in this defense in depth philosophy are:

1. Prevent a fire from occurring,

2. Promptly detect and suppress a fire,

3. Mitigate the consequences of a fire.

The fire protection features shall be independent from other features or systems which are routinely taken out of service during shutdown modes of operation.

2.7.3 DISCUSSION

For clarity the three elements of the defense in depth philosophy outlined above will be discussed in reverse order. Only Division 1 of a system is discussed; Division 2 is identical to Division 1.

2.7.3.1 Mitigation of Fire Consequences

DIVISIONAL SEPARATION

Shutdown Cooling System components for each division are completely separated from each other with 3-hour rated fire barriers with no communicating openings (see CESSAR-DC Figure 9.5.1-2 reproduced here as Figure 2.7-1). All penetrations within these barriers are sealed with assemblies that are qualified to maintain the integrity of the 3-hour rating. This assures that a fire involving one division of Shutdown Cooling System components will not effect the redundant division.

INTERDIVISIONAL SEPARATION

Within each division, the containment spray pump and the shutdown cooling pump can be interchanged with each other. These pumps can be used interchangeably with valve manipulations guided by approved procedures. For each division, the shutdown cooling pump is separated from the containment spray pump with 3-hour rated fire barriers and 3-hour rated fire doors for openings. The valve which allows switchover from one pump to the other is located in a separate fire area. This will enable operators to make the switchover without being exposed to a fire involving either the Containment Spray or Shutdown Cooling Systems. Finally, the containment spray pump is powered from a safety bus separate from the shutdown cooling pump. The safety buses are separated from each other with 3-hour rated fire walls. For example, the Division 1 Safety Bus A is located in Fire Area 65 and the Division 1 Safety Bus C is located in Fire Area 70 (see CESSAR-DC Figure 9.5.1-3 reproduced here as Figure 2.7-2).

This interdivisional mechanical and electrical separation assures the operating of shutdown cooling can be maintained if a fire occurs concurrent with the redundant division being out of service.

2.7.3.2 Detection and Suppression of Fires

DETECTION

Fire Area 38 contains the Division 1 containment spray pump and heat exchanger and Fire Area 41 contains the shutdown cooling pump and heat exchanger. These areas were evaluated during the recently completed System 80+ Fire Hazards Assessment. This assessment considered the fixed and transient combustible loads in these areas and the importance of the components to plant shutdown. Both areas will be equipped with full area coverage ceiling mounted ionization smoke detectors. These detectors provide an early warning alarm at the central fire alarm console in the event of a fire. Detector location and spacing is based on engineering analysis to optimize detector effectiveness. This analysis will be referenced in the System 80+ Fire Hazards Analysis to be completed later in the design process.

The detection system is highly reliable and will be kept in service at all times, even during shutdown modes of operation.

SUPPRESSION

The System 80+ Fire Hazards Assessment concludes that a fixed automatic suppression in the form of automatic sprinklers is not warranted. This is due to the minimal combustible loadings in these areas. This will be verified later by engineering analysis, and will be referenced in the System 80+ Fire Hazards Analysis to be completed by the plant designer before operations.

Portable fire extinguishers and fixed manual fire hose stations provide manual fire fighting capability. The fire hoses are supplied from a dedicated fire protection water supply. Because of the fire barrier arrangement discussed previously, manual fire fighting activities can be accomplished without exposing either the redundant division equipment or interdivisional equipment to the effects of smoke or hot gases from a fire.

MANUAL FIRE FIGHTING

A fully trained and equipped on-site fire brigade would provide fire fighting activities for the System 80+. (See CESSAR-DC Section 9.5.1.9.3.) The brigade would be thoroughly familiar with the plant layout and will conduct sufficient fire drills and fire pre-planning to effectively control and suppress any credible fire. A documented pre-fire plan which outlines the necessary fire fighting strategies, will be prepared prior to plant start-up.

MAINTAINED LEVEL OF FIRE PROTECTION

The System 80+ fire protection system is not degraded or reduced during plant shutdown. There will be no reason to breach the fire boundaries, interrupt the detection system, or impair the fire hose (standpipe) system. All of these features are provided specifically for fire protection and are not shared with or dependent on any other systems or features.

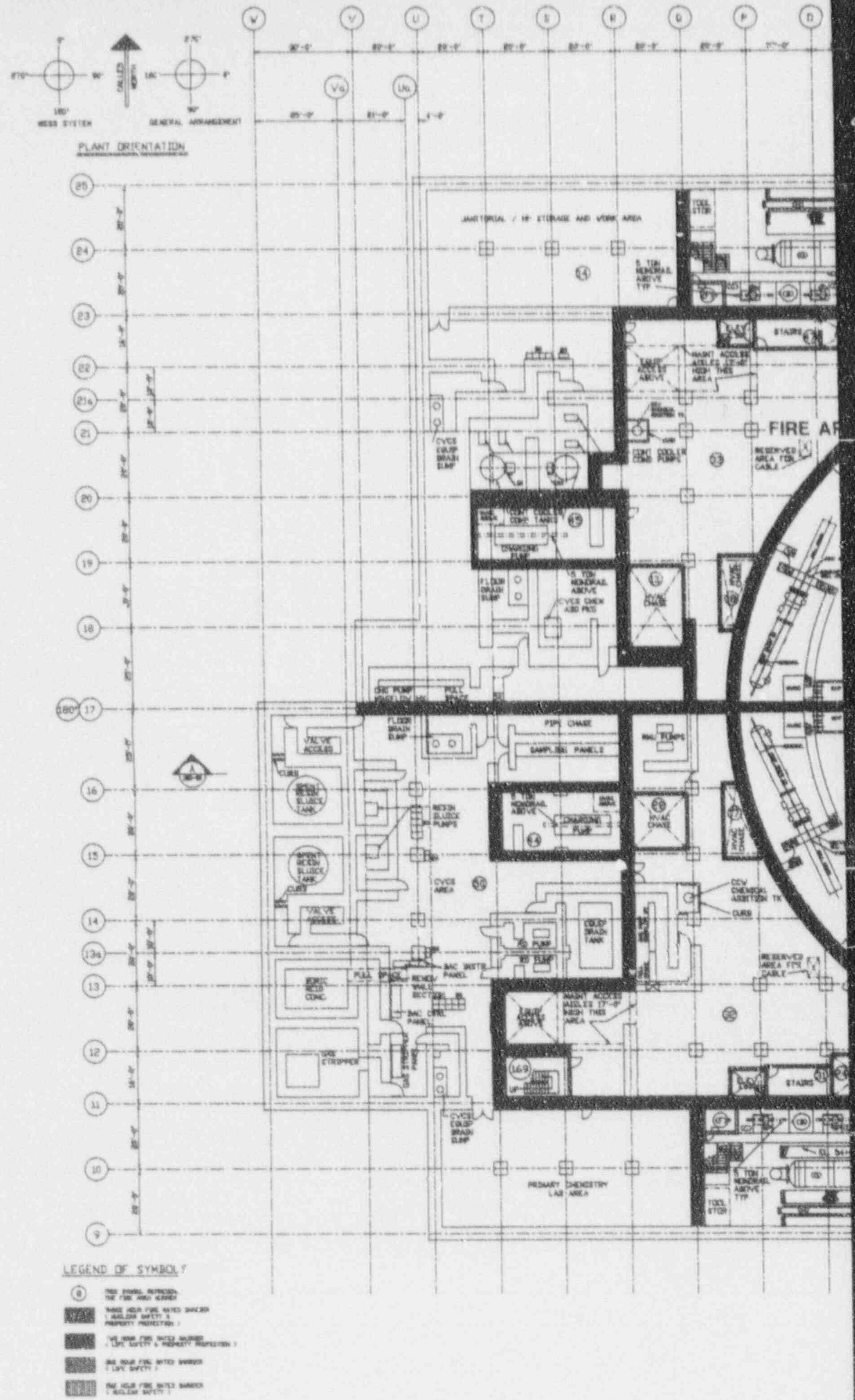
2.7.3.3 <u>Prevention of Fires</u>

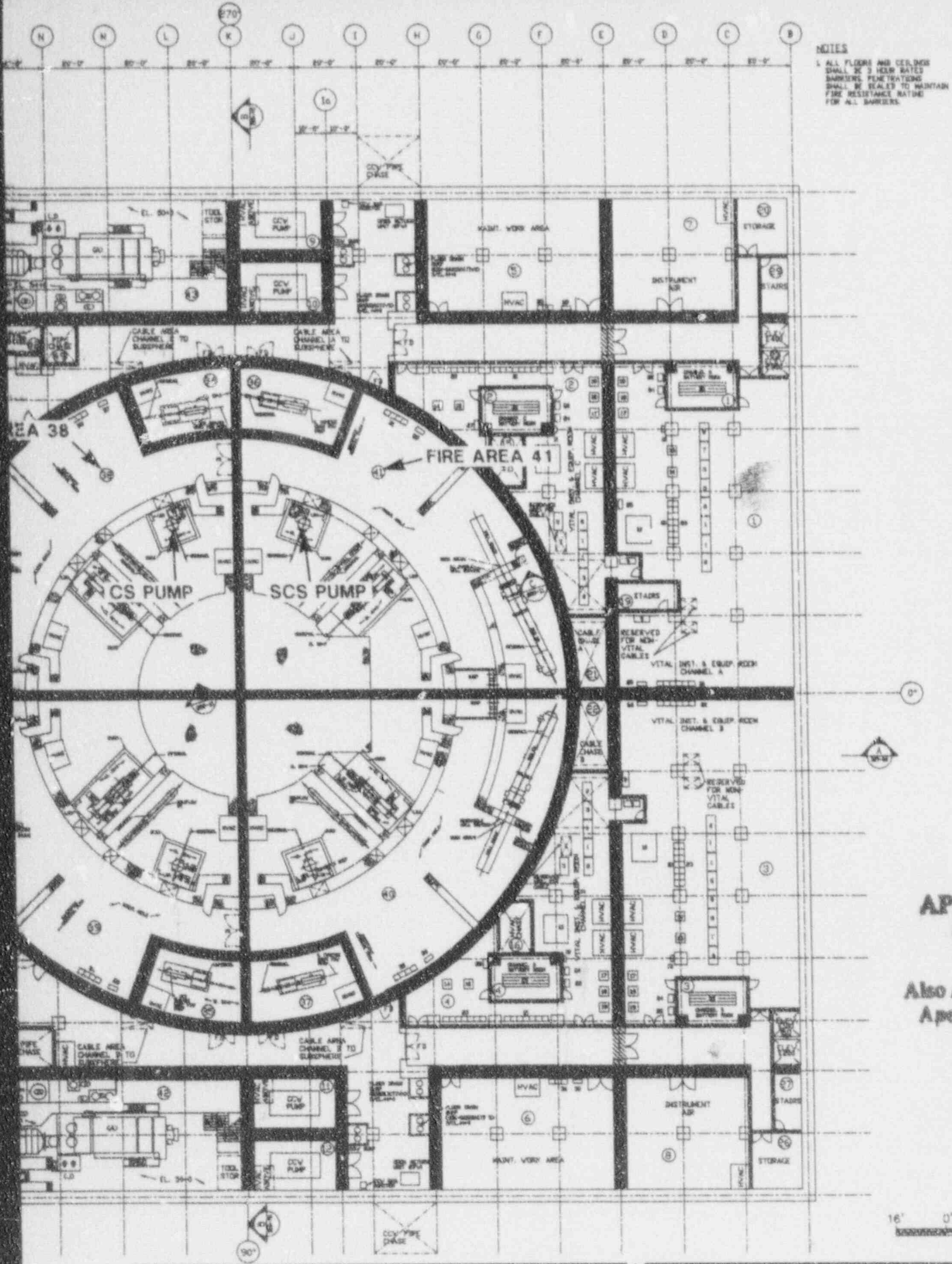
Prevention is the most important element in the defense in depth philosophy. When this element is successful there is no need to employ the other elements. To facilitate the implementation of this element, work place procedures and guidelines will be established by the owner-operator based on guidance provided by the plant designer. Procedural guidance would include control of combustibles, housekeeping, and control of hotwork. The preparation of these procedures will consider those areas in which a fire during shutdown modes of operation could pose a risk. The procedures will include requirements to reduce the risk of fire ignition during shutdown. For example, the control of combustibles procedure may establish a maximum amount and configuration of combustible materials that may be left unattended in any of these areas. This will not be based solely on an arbitrary "good engineering practice" approach, but will consider the amount of combustibles necessary to result in a fire that could cause unacceptable damage. The control of hotwork and housekeeping procedures will be developed by the owner-operator and implemented so as to not place unnecessary restrictions on shutdown maintenance activities, yet will provide a high level of fire prevention.

2.7.4 RESOLUTION

The fire protection features provided by the System 80+ design are consistent with the acceptance criteria outlined in Section 2.7.2. These features will reduce the risk due to fire during shutdown operation to an acceptable level. The combination of fire protection features resulting from employing the defense in depth philosophy will minimize the potential for fire damage to systems required for shutdown operations.

This issue has been resolved by the design features of System 80+.


PLANT ORIENTATION
GENERAL ARRANGEMENT
JANITORIAL / HP STORAGE AND WORK AREA
5 TON MONORAIL ABOVE TYP
TOOL STOR
STAIRS
FIRE AR
RESERVED AREA FOR CABLE
CVCS EQUIP DRAIN SUMP
CHARGING PUMP
FLOOR DRAIN SUMP
5 TON MONORAIL ABOVE
CVCS CHEM ADD PKG
HVAC CHASE
PIPE CHASE
SAMPLING PANELS
VALVE ACCESS
SPENT RESIN SLUICE TANK
RESIN SLUICE PUMPS
CVCS AREA
CCW CHEMICAL ADDITION TK
CURB
EQUIP DRAIN TANK
BORIC ACID CONC.
GAS STRIPPER
PULL SPACE
PRIMARY CHEMISTRY LAB AREA
LEGEND OF SYMBOLS



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SYSTEM 80+™	NUCLEAR ISLAND FIRE BARRIER LOCATIONS PLAN AT ELEVATION 50+0	Figure 2.7-1

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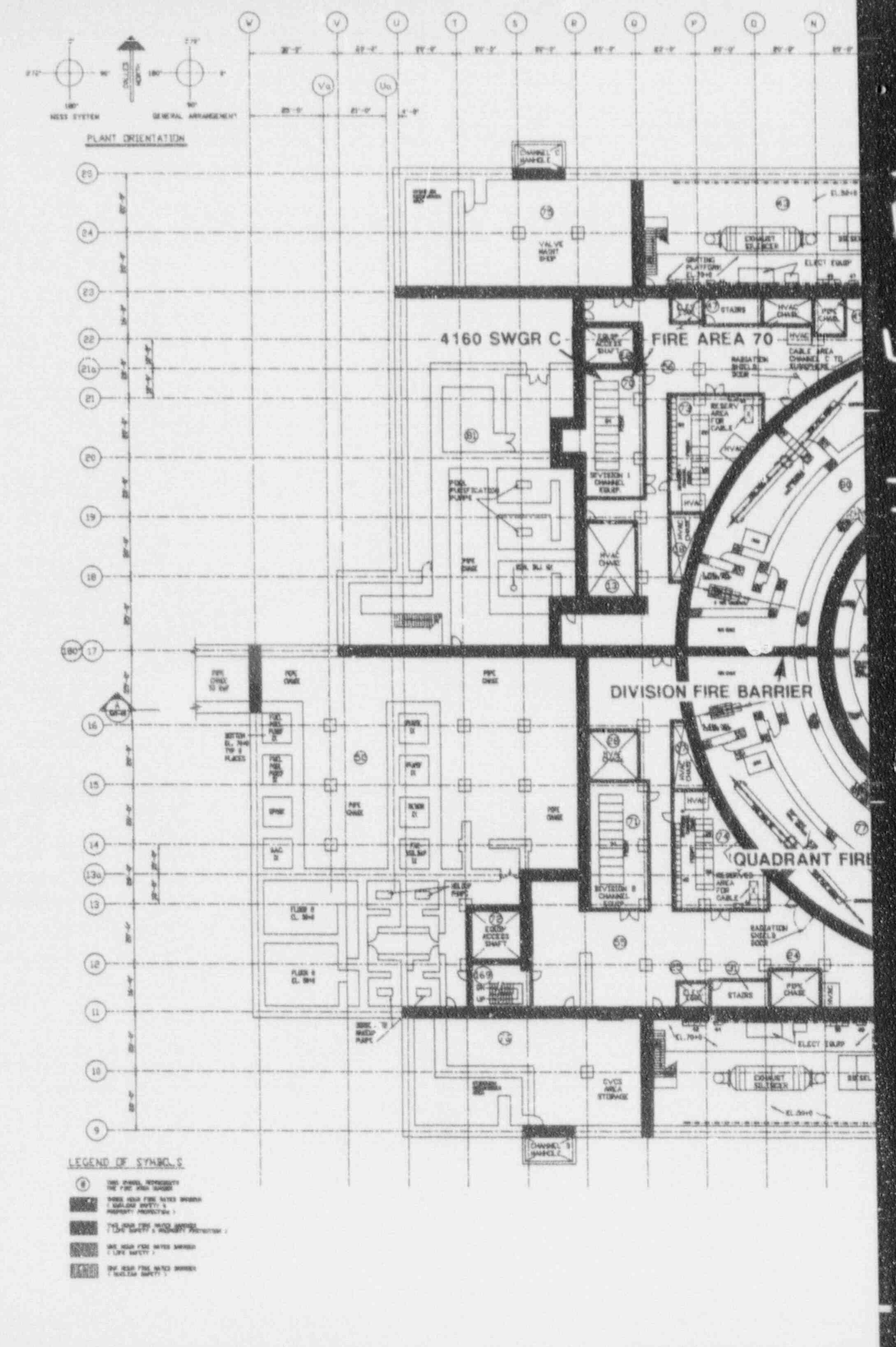
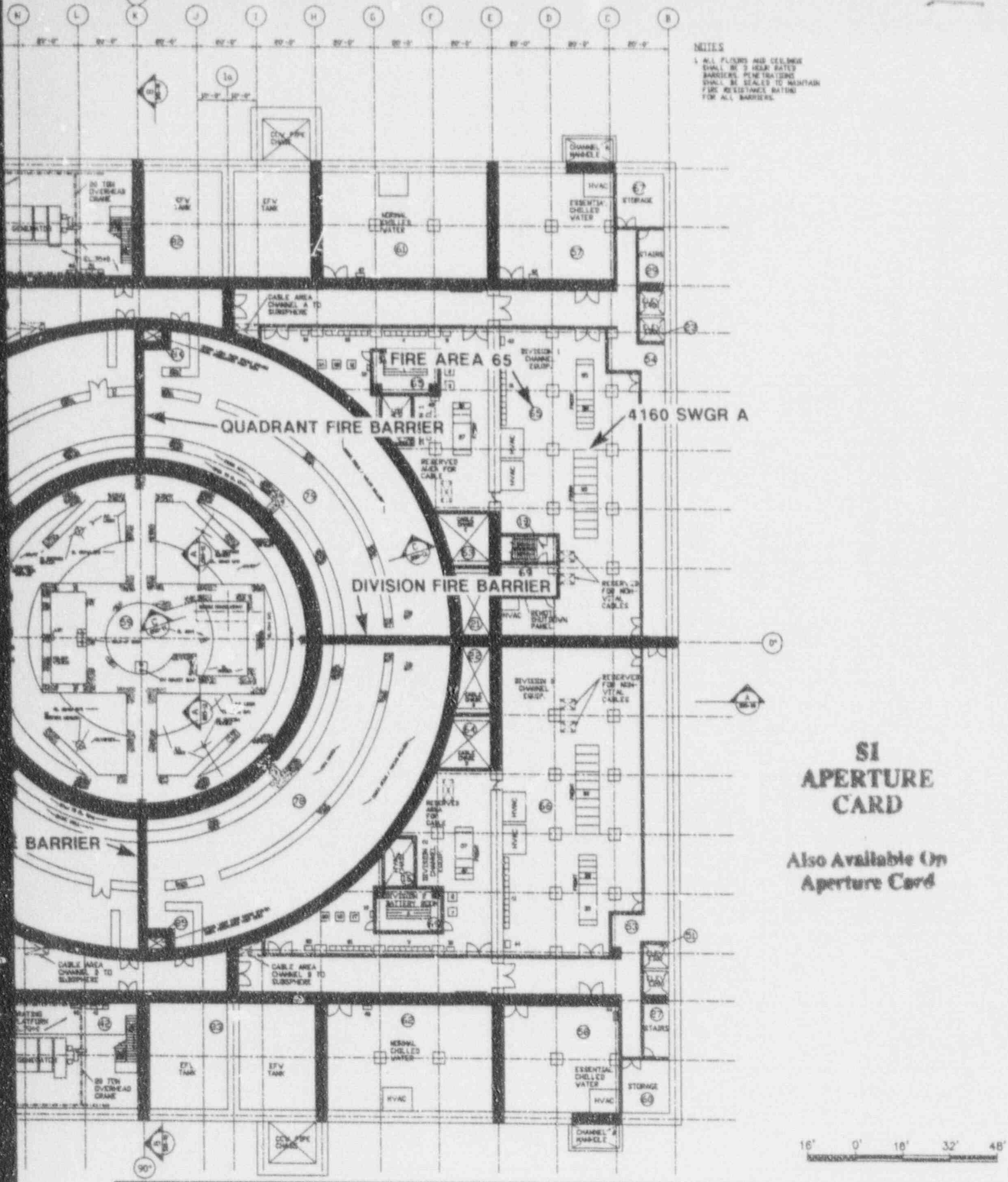


PLANT ORIENTATION
4160 SWGR C
FIRE AREA 70
DIVISION FIRE BARRIER
QUADRANT FIRE
LEGEND OF SYMBOLS



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SYSTEM 80+™	NUCLEAR ISLAND FIRE BARRIER LOCATIONS PLAN AT ELEVATION 70+0	Figure 2.7-2

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2.8 INSTRUMENTATION

2.8.1 ISSUE

Over the past several years, industry and regulatory concern with a loss of shutdown cooling has increased. Despite an emphasis on improved shutdown procedures, the frequency of some incidents has not been reduced, particularly for losses of shutdown cooling during mid-loop operations. Furthermore, the effects of a loss of shutdown cooling are more serious than originally realized. The Nuclear Regulatory Commission (NRC) has requested responses to several design issues related to Nuclear Steam Supply Systems (NSSS) operations while on shutdown cooling; specifically during reduced inventory operations.

Operators have, in many cases, had difficulty in determining plant parameters and equipment status during depressurized, shutdown conditions. This is due to the amount and quality of information available being marginally adequate or inadequate for prevention, recognition and mitigation of abnormal conditions in a timely manner. In particular, this information includes the reactor coolant system water level, reactor core exit temperature, and performance of decay heat removal systems.

Losses of shutdown cooling can be partially attributed to misleading, inaccurate, or erroneous vessel level indication, particularly when vessel coolant level is lowered to within the hot leg between the level required for steam generator nozzle dam installation and the level required to prevent vortexing in the shutdown cooling suction line. Refer to Figure 2.8-1. Providing an adequate fluid level in the hot leg above the level at which vortexing occurs will ensure that the shutdown cooling fluid will not entrain air. This scenario has been a contributor to the loss of shutdown cooling due to pump cavitation.

The NRC has recommended that advanced reactor designs include an enhanced instrumentation package which assures:

1. that reduced inventory operations can be accurately and continuously measured. For example, accurate instrumentation can establish reactor coolant level anytime during the draindown process. Accurate level measurement can assist in differentiating between the anticipated dynamic effects of the draindown process and additional, unintended inventory losses; and

2. that a loss of decay heat removal event during reduced inventory operations can be readily detected. This ensures a timely response to a loss of shutdown cooling event. The instrumentation should "provide reliable indication of parameters that describe the state of the

Reactor Coolant System (RCS) and the performance of systems normally used to cool the RCS for both normal and accident conditions" (Reference 4).

The NRC has specified that instrumentation for reduced inventory conditions should provide both visible and audible indications of abnormal conditions in reactor vessel temperature and level, and decay heat removal system performance.

2.8.2 ACCEPTANCE CRITERIA

The instrumentation provided for reduced inventory operations in the System 80+ design will reduce the safety risks associated with shutdown modes of operation. Instrumentation will be provided to avoid causing or contributing to a loss of shutdown cooling at reduced inventory conditions, and to aid in correctly interpreting a loss of shutdown cooling, should one occur.

The following recommendations are taken from Enclosure 2 to Reference 4:

"At a minimum, provide the following in the Control Room (CR):

1. two independent RCS level indications when the reactor vessel (RV) head is on the vessel

2. at least two independent temperature measurements representative of the core exit whenever the RV head is located on the top of the RV (we [NRC] suggest that temperature indications be provided at all times)

3. the capability of continuously monitoring decay heat removal (DHR) system performance whenever a DHR system is being used for cooling the RCS

4. visible and audible indications of abnormal conditions in temperature, level and DHR system performance."

Also, Enclosure 2 of Reference 4 includes NRC concerns and suggestions on meeting these recommendations. These include, for example:

"1. We suggest that licensees investigate ways to provide [accurate] temperature [measurements] even if the head is removed, particularly if a lowered RCS inventory condition exists.

2. We expect sufficient information [be provided] to the operators that an approaching [DHR system] malfunction is clearly indicated.

3. We expect both audible alarms and a panel indication when conditions exist which jeopardize continued operation of a DHR system, as well as when DHR is lost.

4. The low limit of level indication must be below the level necessary for operation of the DHR system. Level information is necessary under loss of DHR conditions since it provides an indication of core coverage and ... of the time to core uncovery. It is also useful in mitigating the loss of DHR accident."

Section 2.8.3.2 of this report contains the description of the System 80+ instrumentation package for reduced inventory operations, including:

- the monitored parameters,
- instrumentation ranges and accuracies,
- alarm setpoints,
- instrument availability,
- display and monitoring capability, and
- quality assurance.

A summary of the System 80+ design features which meet each of the above mentioned NRC recommendations for instrumentation will be included in the final submittal of this report.

2.8.3 DISCUSSION

2.8.3.1 Instrumentation Design Basis

To effectively monitor the draindown process to mid-loop via System 80+ enhanced instrumentation, information obtained from plant analyses forms the basis for the instrument's design requirements.

Instrumentation specified for reduced inventory operations is based on analyses in the following areas:

- operations from a solid plant to mid-loop conditions (which define dynamic draindown characteristics);
- instrumentation features which will reduce the likelihood of operator error during shutdown operation;
- possible ways in which shutdown cooling can be lost while the plant is in a reduced inventory condition;

- flow dynamics of the shutdown cooling system (SCS), including those which contribute to vortexing;

- the plant response to losses of shutdown cooling, due to various initiators, including RCS thermal hydraulic effects and manometric effects; and

- mitigation planning aimed at the reinitiation of shutdown cooling, delaying the onset of boiling, and delaying core uncovery.

The design goals of the instrumentation package are to provide:

prevention	-	enhanced monitoring capabilities for prevention of a complete loss of SCS operation, and
mitigation	-	the timely response to a loss of SCS.

These goals cannot be achieved without a complete understanding of plant behavior during reduced inventory operations.

2.8.3.2 Instrumentation Description

Table 2.8-1 describes the instrumentation package for reduced inventory operations included in the System 80+ design. Additional detail will be provided in the June 15, 1992 updated submittal of this report.

2.8.3.2.1 Level

Two diverse instruments types will be provided for the measurement of level during reduced inventory operations. These instruments will make up the refueling water level indication system. A wide range dP based level sensor will be provided which functions to measure level between the pressurizer and the junction of the shutdown cooling suction line on the hot leg. A narrow range system will be provided which measures RCS coolant level during reduced inventory operations.

Two instrument types are being considered for level indication during reduced inventory operations. Although complete details of the final instrument design will be presented in the June 15, 1992 submittal, brief descriptions of each of the instrument types are provided below for information. The choice of instrument type does not impact the evaluations presented in this report, given that the measurement function will be provided in the plant.

A narrow range heated junction thermocouple based system is being considered which uses input from four separate heated junction thermocouple probes for accurate measurement of level in the reactor vessel and hot leg region. The heated junction thermocouple system combines the output from two inadequate core cooling heated junction thermocouples and two heated junction thermocouple probes designed specifically for the hot leg region of measurement (see Figure 2.8-2). The range of the inadequate core cooling heated junction thermocouple probe extends from the reactor vessel head to the fuel alignment plate. The range of the clustered heated junction thermocouple probe emphasizes the hot leg region. Thus, two control room displays are obtained from combining the output from one inadequate core cooling HJTC and one clustered HJTC probe. Thus, the operator is provided with two continuous, independent RCS level indications in the control room for reactor vessel level.

The heated junction thermocouple system compensates for the flow gradient across the core associated with the operation of only one shutdown cooling suction line. The instruments are located in areas of the core which minimize the effect of core outlet nozzle exit effects. The instrument sensors have an accuracy and response time consistent with the maximum drain down rate of the RCS. The instruments are designed so that the signal and power are transmitted on individual electrical conductors. Failure of one sensor will not result in the loss of signal from the remaining sensors. The measurements of RCS water level obtained via these probes are limited to those periods when the reactor vessel head is installed.

A narrow range dP based level sensor system is being considered which functions to measure level in the hot leg region. The lower tap is located at the hot leg/SCS suction line interface. The reference leg for this system will be located either at a direct vessel injection nozzle or will be opened to atmosphere. The measurement of RCS water level obtained via the dP level sensor is available when the head is on or off the vessel.

The use of both a wide range pressure differential (dP) instrument and either two pairs of heated junction thermocouple (HJTC) based instruments or a narrow range dP based instrument for refueling water level monitoring provides highly reliable, redundant, and independent indications of vessel water level. Instrument ranges provide a continual draindown measurement from the pressurizer to a level below that required for SCS operation. Since the level instrumentation is independent, common mode misoperation or failures due to dynamic effects will not be masked.

Each independent level instrument provides a suitable measurement and is accurate for it's intended range of use. For mid-loop operations, the HJTC based level probe provides accurate level measurements to within less than 1 inch of vessel level. This is key, since there is a very narrow margin between manway flooding (or nozzle dam installation) and SCS pump cavitation. The level instrumentation is alarmed in the Control Room to reflect this tight operational tolerance in the hot leg.

2.8.3.2.2 Temperature

Several instruments are available for continuous temperature measurements during reduced inventory operations with the reactor vessel head on. These include:

- core exit thermocouples (CETs),
- shutdown cooling heat exchanger inlet and return line temperature sensors,
- hot leg resistance temperature detectors (RTDs), and
- refueling water level instrument temperature sensor (HJTC system only).

All provide representative indications of the core exit temperature when the shutdown cooling system is operational. If the shutdown cooling system is lost, the CETs, hot leg RTDs, and reduced inventory water level thermocouple sensor input are available to track the response to the loss of shutdown cooling or the approach to boiling.

Per Enclosure 2 to Reference 4, temperature measurement is provided with the reactor vessel head off. The temperature instruments operable during this mode are the hot leg resistance temperature detectors and, while fuel is not being shuffled, the CETs. The core exit fluid temperature can be measured through the use of hot leg RTDs as long as the SCS is operable. Each RCS hot leg has a total of five RTDs which are located in the hot leg at the junction of the SCS suction nozzle. In relation to the hot leg horizontal centerline, two RTDs are located above the centerline, one is at the centerline, and two are below the centerline. Only the lowermost two in each hot leg will provide input to the temperature reading for mid-loop operations, since they will be the only ones in full contact with reactor coolant. The lowest probes penetrate the internal diameter of the hot leg pipe at approximately 10" below the midloop fluid level, thus assuring accurate readings are provided.

All temperature sensors will have associated alarms in the control room to be used as aids in determining the response to a loss of shutdown cooling and tracking the approach to boiling. Awareness of time constraints provides information that is useful for deciding how to allocate effort.

2.8.3.2.3 Shutdown Cooling System Performance

As stated in Enclosure 2 to Reference 4, sufficient information will be available to the control room operator to indicate an approaching shutdown cooling system malfunction. Indications of sufficient pump suction pressure and possible vortexing include unsteady pump current (as indicated by SCS/containment spray system (CSS) pump motor current), loss or reduction in shutdown cooling flow (as indicated by the shutdown cooling system flowrate), insufficient pump NPSH (as indicated by the pump suction pressure sensor), or indication or rising RCS level (as water is displaced by the air and vapor in the shutdown cooling system). If a pump gives indications of air ingestion or cavitation, alarms will prompt the operator to stop the pump immediately. As detailed in Section 2.8.3.2.5, shutdown cooling panel displays will include valve lineup information for critical shutdown cooling flowpaths.

2.8.3.2.4 Quality Assurance

The following instruments are designated as safety related and therefore within the scope of environmental qualification and quality assurance.

- core exit thermocouples
- hot leg resistance temperature detectors
- refueling water level temperature sensor (unheated thermocouple)
- refueling water level indicator (ICCI heated junction thermocouple based design)
- shutdown cooling flowmeter
- shutdown cooling heat exchanger inlet and return line temperature sensors
- shutdown cooling valve position indicators

The safety related designation of these instruments is a consequence of their required functions in other plant modes of operation, including for some, inadequate core cooling. The CENP Quality Assurance Program designates items which are safety-related as Quality Class 1 equipment, and therefore, are subject to the highest level of quality activity. The CENP Quality Assurance Program designates items which are not safety-related but nevertheless require a high level of quality activity, as Quality Class 2 equipment. In this case, where reliable and accurate instrumentation is required for reduced RCS inventory conditions, designating the instrument as Quality Class 2 requires that a quality program be implemented that assures that quality is commensurate with intended use. In the procurement of the instrumentation, appropriate technical requirements and quality requirements are specified in the purchase order to this end.

Enclosure 2 to Reference 4 states: "...we will accept the following for resolving the items identified in the letter: ..... (2) reliable equipment in lieu of the comparable safety grade classification ...." Therefore, the following list of Quality Class 2 instruments identified on Table 2.8-1 are classified as non safety-related:

- refueling water level indicator (wide and narrow range dP design),
- refueling water level indicator (clustered heated junction thermocouple based design),
- shutdown cooling pump suction and discharge pressure sensors, and
- SCS pump/CS pump ammeter.

2.8.3.2.5 Display and Monitoring Capability

Details of the NUPLEX 80+ Advanced Control Complex Information presentation and panel layout evaluation are described in CESSAR-DC Section 18.7. In addition to the following summary, refer to Section 18.7 for detailed or supplementary explanation of control room information presentation.

The operator obtains plant information from a number of sources in the NUPLEX 80+ control room, which include:

1. A large plant overview status board known as the Integrated Process Status Overview (IPSO),
2. Alarm tiles and associated alarm messages,
3. Discrete indicators which provide frequently used and important information,
4. CRT display formats containing essentially all power plant information, and
5. Component and process control indicators.

There are a number of NUPLEX 80+ design features in 1 through 5 above that specifically implement indications, alarms, and displays applicable to depressurized, shutdown conditions. They are described in the following sections.

2.8.3.2.5.1 Integrated Process Status Overview (IPSO)

IPSO is used for quickly assessing overall plant status, organizing operational concerns, and establishing priorities for operator action. Information provided on the IPSO display includes:

1. Major system and component statuses shown on an overview schematic which are representative of the current operating heat transport systems,

2. Alarms to aid the operator in quickly identifying the location of important status information,

3. Deviations from control setpoints and identification of improving or degrading trends to improve the operator's awareness of plant conditions, and

4. Key representative parameters (e.g., RCS temperature and reactor vessel level).

Alarm windows are provided for plant critical functions:

- Reactivity Control
- Core Heat Removal
- RCS Heat Removal
- RCS Inventory Cont.
- RCS Pressure Cont.
- Steam/Feedwater Conversion*
- Electrical Generation*
- Heat Rejection*
- Containment Environment Control
- Containment Isolation
- Radiological Emissions Control

*For power production only

Nuplex 80+ alarms are mode-dependent and equipment dependent to ensure their validity for different operational conditions. For all modes, including shutdown and refueling conditions, individual sensed process parameter values and alarm states are used to determine critical function alarms, either directly or as processed by an algorithm that uses more than one (1) process parameter input. In either case, the operator quickly is made aware of the affected critical function(s). For example, a high core exit temperature alarm state would be used as an input to the Core Heat Removal critical safety function alarm during a loss of shutdown cooling.

The systems represented on IPSO are the major heat transport pathways and systems that are required to support the heat transport process. These systems include those that require availability monitoring per Regulatory Guide 1.47, and all major success paths that support the Plant Critical Functions.

The following systems have dynamic operating status representations on IPSO. Their identifying descriptors on the IPSO display are shown below:

CC - Component cooling water
CD - Condensate
CI - Containment isolation
CS - Containment spray
CW - Circulating water
EF - Emergency feedwater
FW - Feedwater
IA - Instrument air
SC - Shutdown cooling
RC - Reactor coolant
SI - Safety injection
SW - Service water
TB - Turbine bypass
SD - Safety Depressurization

System information presented on IPSO includes system operational status, any change in operational status (i.e., active to inactive, or inactive to active) and the existence of alarms associated with the system. Alarm information on systems helps to directly inform an operator about possible underlying causes of critical function alarms. The IPSO display, as well as all display pages, is also available at any data processing system CRT, which includes control room panels, the control room supervisor's desk, assistant operator workstations, and the technical support station.

2.8.3.2.5.2 Alarm Tiles and Associated Alarm Messages

Alarm tiles are displayed on electroluminescent flat panel displays in the Discrete Indication Alarm System (DIAS). These tiles are functionally grouped and located on the appropriate control room panel. Shutdown cooling system alarm tiles are located on the Engineered Safety Features panel. This panel includes the controls for Safety Injection, the Safety Injection Tanks, Shutdown Cooling, Reactor Cavity Flood, Safety Depressurization, Emergency Feedwater, Containment Spray, IRWST, and Containment Isolation. Individual alarm inputs to the shutdown cooling alarm tiles include (for each train):

- low shutdown cooling pump header pressure
- low shutdown cooling flow
- high shutdown cooling heat exchanger outlet temperature
- shutdown cooling pump motor current deviation

In addition, this panel will have a tile for RCS conditions, with individual inputs for shutdown, depressurized conditions:

- low RCS water level
- high core exit temperature
- low refueling cavity level

To ensure alarm validity, all NUPLEX 80+ alarms are mode and equipment status dependent, and signal validation of inputs is done where multiple signals of the same process parameter exist. These features eliminate nuisance alarms and help ensure a true "dark board" when alarms do not exist. These features enhance operator diagnosis of alarms when they do exist.

When alarm tiles in DIAS are acknowledged, the operator is presented with a DIAS display with alarm messages showing which of the alarm tile inputs caused the alarm.

2.8.3.2.5.3 Discrete Indicators

Discrete indicators are provided on the NUPLEX 80+ control room workstations to provide the operator with information that (1) is frequently used to assess system level performance, and (2) allows continued operation if the Data Processing System should become unavailable. Discrete indicators use validated process parameter inputs where multiple process parameter measurements exist, and include trend information for routine monitoring, and diagnosis of abnormal conditions. Where analog data [illegible] posed of different ranges of information, DIAS automatically [illegible]s to the appropriate range, and indicates to the operator [illegible] a range change has occurred.

Discrete Indicator displays to support shutdown cooling for key parameters are on the Engineered Safety Features panel. These include:

<u>Shutdown Cooling System (per train)</u>

- inlet temperature
- outlet temperature
- pump header pressure
- flow
- heat exchanger inlet temperature
- heat exchanger outlet temperature

- pump motor current

Reactor Coolant System

- pressurizer level
- reactor coolant system level
- pressure
- core exit temperature
- refueling cavity level

2.8.3.2.5.4 CRT Display Pages

CRT display pages contain, in a structured hierarchy, all the System 80+ plant information that is available to the operator. The CRT pages are useful for information presentation because they allow graphic layouts of plant processes in formats that are consistent with the operator's visualization of the plant. In addition, CRT formats are designed to aid operational activities of the plant by providing trends, categorized listings, messages, operational prompts, as well as alert the operator to abnormal processes.

The IPSO display page forms the apex of the NUPLEX 80+ CRT display page hierarchy. Three levels exist below IPSO: general monitoring, system/component control, detail/diagnostic. Each level of the hierarchy provides an information content designed to satisfy particular operational needs.

The CRT displays are provided by the Data Processing System (DPS). Any display page is available at any CRT. Operator acknowledgement of CRT alarms also acknowledges the same alarm in DIAS (and vice versa). The CRT alarm actuation message indicates the cause of the alarm, similar to DIAS

The shutdown cooling system will be shown on a Level 2 display, with more detailed information on two Level 3 displays, one per shutdown cooling train. These displays will include all necessary information to clearly describe the status and performance of the system. This includes system mimic, component activity (e.g., on/off or open/closed) component controllability (e.g., key valves locked open or closed), system parameters (e.g., temperature, level), and system/component alarms. The Level 2 display will include reactor coolant system level and core exit temperature to integrate the shutdown cooling and RCS status for this display. The RCS is also presented on a separate Level 2 display.

2.8.3.2.5.5 Component and Process Control Indicators

NUPLEX 80+ component control features (e.g., actuation/switches/controls) provide the primary method by which the operator actuates equipment and systems. The shutdown cooling system controls are functionally grouped within a system mimic on the Engineered Safety Features panel. At that panel, shutdown cooling system control is integrated with DIAS alarm tiles important to shutdown cooling and with CRT display of the shutdown cooling system. Controls, alarms and CRT displays for other systems applicable to shutdown operations, such as component cooling water and safety injection, are available at that panel as well.

2.8.3.2.5.6 NUPLEX 80+ Alarm Characteristics

There are a number of special features in the design of the NUPLEX 80+ alarm system that support operator diagnosis of alarm conditions and that would be particularly supportive of depressurized, shutdown operations. These are:

1. Mode and Equipment Status Dependency

2. Audible Alarm Information

3. Stop Flash Feature

4. Operator Established Alarms

5. Operator Aids

In addition, the categorization of all alarms is considered in the bases for alarm display location. For instance, alarms that indicate approach to potential equipment damage, but do not affect critical function or success path status, are presented only on alarm tiles. These would not be included as input to alarms shown on IPSO.

A key feature to aid operator navigation in the CRT display page hierarchy also includes alarm categorization to assist the operator. This feature, the "display page menu", is on each CRT display page. The menu indicates alarms exist in various sectors of the hierarchy, and depending on the sector, the operator can distinguish between lower level alarms that are critical function or success path related, and those that are not (e.g., personnel hazard, or equipment damage).

2.8.4 RESOLUTION

The issue of instrumentation for shutdown operation is resolved on System 80+ by the instrumentation and control room displays described in the previous sections of this report. This instrumentation will meet or exceed the recommendations of Generic Letter 88-17, and will significantly reduce risk associated with operations during shutdown, particularly when the reactor is in a reduced inventory condition, as long as prior to the start of draining the reduced inventory instrumentation is placed into operation.

The NUPLEX 80+ Advanced Control Room Complex provides an overview display, indicators, CRT displays, and alarms that meet the acceptance criteria in Section 2.8.2. Indication and alarms are provided on discrete indicators, alarm tile windows and CRTs for RCS level and temperature. In addition, shutdown cooling system status and performance is monitored on CRTs. Shutdown cooling system performance is alarmed on IPSO, alarm tile windows and on CRTs. Also, all alarms are processed for their individual effect on plant critical functions such as reactivity control, core heat removal and RCS heat removal.

Table 2.8-1

Reduced Inventory
Instrumentation Package

Monitored Parameter	Instrument Type	Instrument Function	Range	Indication and Alarm Location	Comments
RCS Water level	Refueling Water Level Indication System (dP based design)	Continuous wide range RCS water level indication during draindown operations.	Wide Range; tap at hot leg/ SCS suction line interface, reference leg at top of pressurizer.	Control Room, with low level alarm.	Highly reliable. Meets NRC requirement for water level measurement to a point lower than that required for SCS operation.
RCS Water level	Refueling Water Level Indic[illegible]ion System (heated junction thermocouple based design)	Independent, continuous narrow range level indication in the hot leg region.	Top of the vessel to the fuel alignment plate.	Control Room with low and high level alarms.	1 located near each hot leg. Effective when SCS suction is aligned to either hot leg. 2 Axi[illegible] [illegible]ings of the.mo[illegible] couples. One spans from the vessel head to the fuel alignment plate. The other contains thermocouples clustered over hot leg region. System provides excellent accuracy and continuous measurement.
RCS Water Level	Refueling Water Level Indication (dP based design)	Continuous narrow range level indication during reduced inventory operations.	Narrow range: tap at SCS suction line/ hot leg interface, reference leg at DVI nozzle or open to atmosphere.	Control Room with low and high level alarms.	Highly reliable for mid loop operations. Meets NRC requirement for water level measurement to a point lower than required for SCS operation.
RCS Temperature	CETs (thermocouple design)	Measures temperature of coolant exiting core.	Optimized for SCS and refueling modes. Will measure boiling. Approximate range 100 - 250 deg F.	Control Room with alarms at high and high-high temperature	Tracks approach to boiling. Temperature indication provided even when head is off vessel. Not available during fuel shuffling. Availability will be maximized.

Table 2.8-1 (Continued)

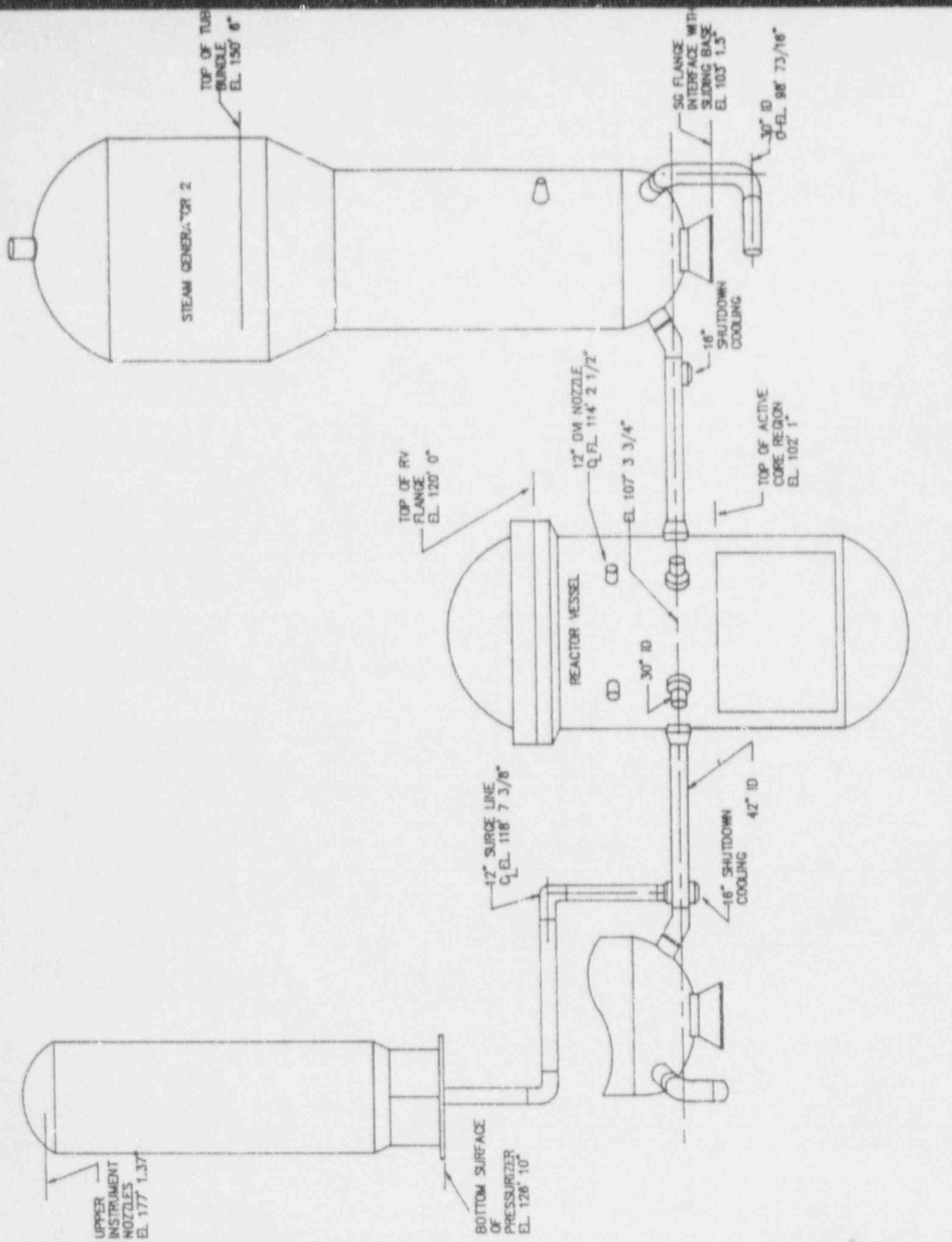
Reduced Inventory
Instrumentation Package

Monitored Parameter	Instrument Type	Instrument Function	Range	Indication and Alarm Location	Comments
RCS Temperature	Refueling water level probe (heated junction thermocouple based design)	Continuous, independent temperature measurement inside the vessel.	Optimized for SCS and refueling modes. Will measure boiling. Approximate range 100 - 250 deg F.	Control Room with alarms at high and high-high temperatures.	Indicates actual vessel water temperature. Tracks approach to boiling.
Hot Leg Temperature	Resistance Temperature Detectors (RTDs)	Measures core exit temperature in the hot leg at both SCS suction line regions. Redundant RTDs provided in each hot leg.	Optimized for shutdown operations. Approximate range 50 - 250 deg F.	Control Room, alarms at high temperature.	Temperature indication is affected by loss of shutdown cooling flow, since flow by the RTDs will not occur.
SCS Flowrate	Flowmeter	Decay heat removal system performance.	Bounds SCS pump flow range.	Control Room, includes low flow alarm.	One located in each SCS return line to the RCS. Can be used to measure CSP flow if CSPs are used for SCS.
SCS Pump/CS Pump Discharge Pressure	Pressure sensor	Measures individual pump discharge pressures.	0 to system design pressure.	Control Room with low pressure.	One instrument located at the discharge to each pump. Identifies individual pump status.
SCS Pump/CS Pump Motor Current	Ammeter	Measure current drawn by pump motor. Fluctuations show air entrainment.	0 to design	Control Room, alarms with preset drop in current.	Confirms pump status (individual pump air entrainment) independent of pressure and flow indicators.
SCS Pump/CS Pump suction pressure	Pressure sensor	Measure pump suction pressure in each pump.	0 to system design pressure.	Control Room with low pressure alarm.	One instrument located at the suction of each pump. Identifies individual pump status.
SCHX Inlet and Return Line Temperature	Temperature sensor	Measures temperature in the suction and discharge lines of the shutdown cooling heat exchanger.	40 - 392 deg. F.	Control Room alarms at high temperature	Temperature indication only available when SCS is operational.

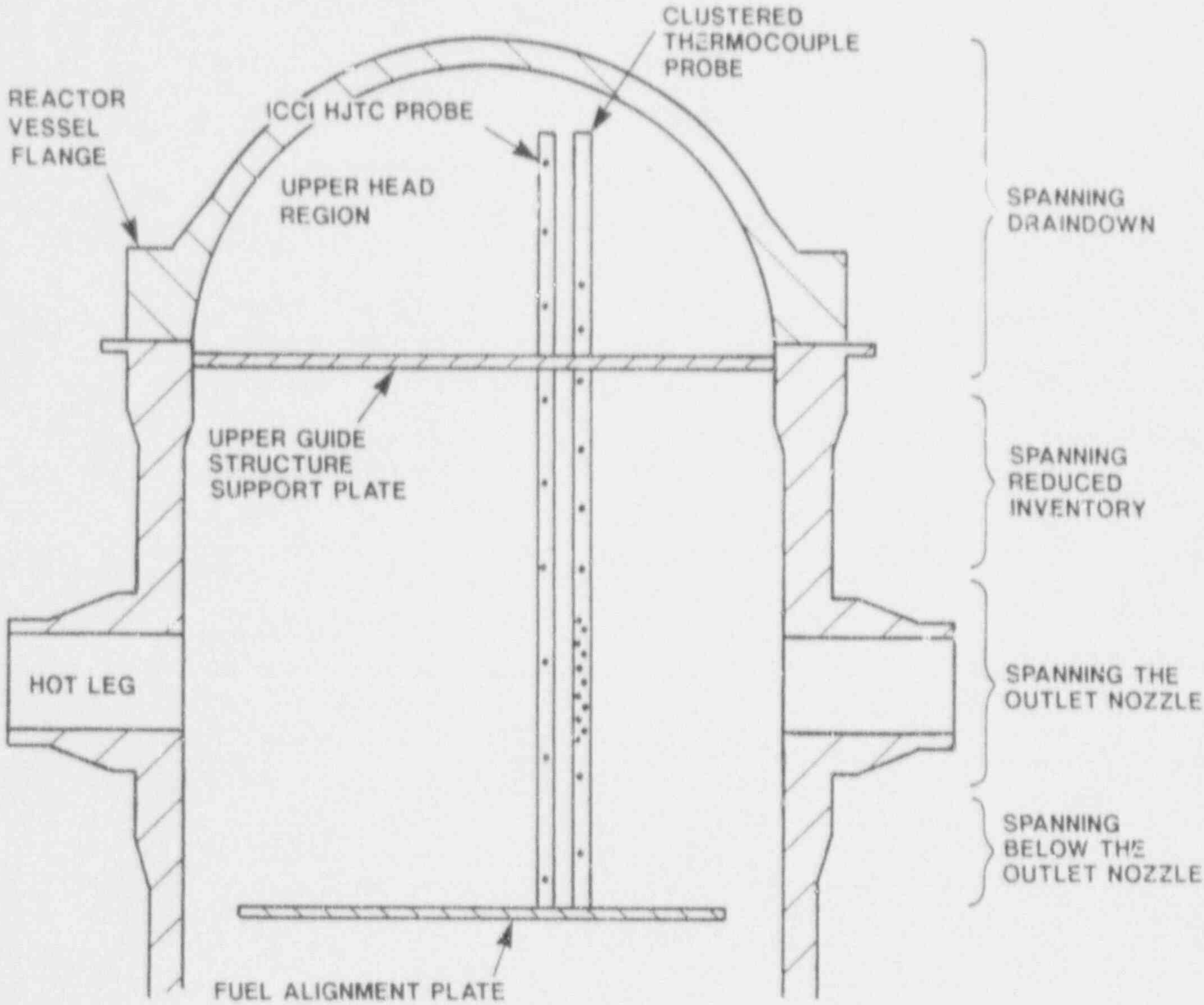
Table 2.8-1 (Continued)

Reduced Inventory
Instrumentation Package

Monitored Paramete:	Instrument Type	Instrument Function	Range	Indication and Alarm Location	Comments
SCS Valve Position Indic:tion	Valve position indication open/closed or throttled.	Status of valve positions in the SCS.	Open/:losed/thr ottled position indication.	Control Room	Will provide information of system lineup status and a:ailable flowpaths.



SYSTEM 80+™	REACTOR COOLANT SYSTEM ELEVATIONS RELATED TO SHUTDOWN COOLING OPERATIONS	Figure 2.8-1



*EXAMPLE OF SENSOR POSITIONING. NOT INTENDED TO BE TO SCALE
(ONE OF TWO INSTRUMENT PAIRS SHOWN)

SYSTEM 80+™	SCHEMATIC REPRESENTATION OF THE NARROW RANGE HEATED JUNCTION THERMOCOUPLE BASED REFUELING WATER LEVEL INDICATION SCHEME	Figure 2.8-2

2.9 ECCS RECIRCULATION CAPABILITY

2.9.1 ISSUE

The issue is the potential for loss of flow to the Containment Spray (CS) and Safety Injection (SI) pumps during accident conditions. System flow could be inhibited by a number of factors. These factors include:

1. Hydraulic effects, such as air ingestion and vortex formation.

2. Debris in the IRWST resulting from maintenance activities or deterioration of insulation from actuation of containment sprays or from LOCA consequences, or

3. The combined effects of items (1) and (2).

2.9.2 ACCEPTANCE CRITERIA

The design of the System 80+ Incontainment Refueling Water Storage Tank (IRWST) and Holdup Volume Tank (HVT) and their associated debris-blocking devices shall comply with the requirements of General Design Criterion 35 of Title 10, Code of Federal Regulations, Part 50, Appendix A. Design Criterion 35 requires that "...suitable containment capabilities shall be provided to assure that ... the system's safety function can be accomplished." To satisfy this requirement, the IRWST and HVT are designed to provide a clean and reliable source of water to the SI pumps for long-term recirculation. The containment is designed to direct containment spray water and emergency core cooling water to the HVT and then to the IRWST.

The SIS shall meet the acceptance criteria specified in USNRC Standard Review Plan section 6.3, Emergency Core Cooling System, Revision 2. In particular, Section 6.3 of the SRP addresses the availability of an adequate source of water for the SIS. Acceptance criteria pertaining to the design of the containment emergency sumps are provided in SRP section 6.2.2, Containment Heat Removal Systems, Revision 4. These criteria address the drainage of containment spray water and emergency core cooling water to the recirculation suction points (sumps) and the screen assemblies surrounding these suction points. Regulatory Guide 1.82. Water Sources for Long-term Recirculation Cooling Following a Loss-of-Coolant Accident, Rev. 1, provides the guidelines for the design of the IRWST and the HVT, and the design of the screens associated with these tanks. Technical considerations related to this issue are detailed in NUREG-0897, Containment Emergency Sump Performance, Revision 1.

2.9.3 DISCUSSION

Water introduced into the System 80+ containment from a RCS break or from containment sprays drains into the Holdup Volume Tank (HVT). This tank serves the purpose of the "containment sump." The Holdup Volume Tank is therefore the low collection point in containment. The contents of this tank are directed to the IRWST through the two IRWST spillways (see Figure 2.9-3). The IRWST serves as the single water source of long-term recirculation for emergency core cooling and containment heat removal.

With the System 80+ design, it should be noted that the IRWST does not serve as the containment sump; this tank specifically serves as a storage tank for refueling water, a clean and reliable source of water for Safety Injection, and a heat sink for condensing steam discharged from the pressurizer.

The arrangement of the IRWST within containment meets the multi-sump requirement of Reg. Guide 1.82, Water Sources for Long-term Recirculation Cooling Following a LOCA. The general plant arrangement separates redundant trains of the SIS and the CSS. The divisional boundary provides complete separation between divisions and effectively creates two identical support buildings. The result is a plant arrangement with two SI pumps and one CS pump in each division. Within each division, the two SI trains (and each CS train) are separated by a quadrant wall to isolate the trains from each other to the maximum extent practical. Each of the four SI pumps has its own suction connection to the IRWST (see Figure 2.9-1) and each of the two CS pumps shares one of these four connections.

Following an accident, water introduced into containment drains to the Holdup Volume Tank. Debris that may exist in containment may be transported to the HVT with this fluid. Debris greater than 1.5 inches diameter is prevented from entering the HVT by a vertical trash rack, which is located at the entrance to the HVT (see Figure 2.9-3). The vertical trash rack is greater than six feet high and more than forty feet long. A debris curb exists at the base of this trash rack to prevent high density debris that may be swept along the floor by fluid flow toward the HVT from reaching the trash rack. The vertical orientation of the trash rack will help impede the deposition of debris buildup on the screen surface. Particles that are smaller than the trash rack mesh will enter the Holdup Volume Tank.

The Holdup Volume Tank is designed to function as a solids trap to help prevent debris from entering the IRWST. High density debris that makes its way through the trash rack will accumulate in the bottom of this tank. The IRWST spillways are located at a high enough elevation to assure that much of the higher-density debris (and debris that tends to sink slowly) will settle to the bottom of the HVT before spilling over into the IRWST. Debris that remains in suspension will make its way to the IRWST spillways. The spillways are shown in Figure 2.9-3. Screens are not present in these spillways to assure uninterrupted flow to the IRWST.

The fine debris that is introduced into the IRWST is prevented from entering the SIS suction piping by a debris screen. These screens are located at each end of the four wing walls that serve as supports for the reactor coolant pumps (see Figures 2.9-1 and 2.9-2). These wing wall assemblies extend from the IRWST floor to the maximum IRWST water level, assuring that all debris will be filtered before reaching the SIS suction lines. The screen assemblies completely enclose the suction lines by running from the end of each wing wall to the side walls of the Holdup Volume Tank or the primary shield walls, as applicable. The wing-wall screens have the capability of removing particles greater than 0.09 inches diameter. This screen size is consistent with the screens used on currently operating units. The wing-wall screens are the final barrier to debris before the SIS suction lines.

Blockage of the debris screens is a major concern with respect to recirculation. The System 80+ screens have a vertical orientation to prevent debris from settling on the screen surfaces. This helps in keeping the screens clear. The design considered the types and quantities of insulation used for the System 80+ components, since post-LOCA deterioration of this insulation is the major potential source of debris in containment. The location of insulation with respect to the HVT and IRWST as well as the possible location of breaks have also been considered. The effective areas of the screens have been determined according to the guidelines provided in Appendix A to Regulatory Guide 1.82, Guidelines for Review of Sump Design and Water Sources for Emergency Core Cooling.

The debris screens have been designed to withstand the vibratory motion of a seismic event without loss of structural integrity. Each screen is capable of withstanding loads imposed by postulated missiles as well as loads due to pressure head differentials.

Consideration has also been given to the materials used for the debris screens. Materials have been selected to avoid degradation during periods of inactivity (i.e., no submergence), and during periods in which the screens are partially or fully submerged.

Each screen used in the System 80+ design is provided with an access opening to allow for inspection of the racks or screens. The screens will be visually examined periodically to detect any corrosion or structural degradation during refueling outage periods. As seen in figure 2.9-1, the suction lines are located within the confines of the wing wall away from the IRWST spargers. This wall design helps isolate the suction lines from the open sections of the IRWST, where most of the maintenance activities will be performed. The fine wing-wall screen will filter any trash generated from this type of activity. In the event that maintenance is needed within the wing walls and near the ECCS suction inlets, permanent box-like screens over the suction piping will protect these lines.

Long-term return of spray water from upper level elevations is not dependent on individual piping runs or spillways. Multiple passive spillways are provided to route water back to the Holdup Volume Tank. Major openings such as hatches and stairwells are also available to return water to the screened entrance to the HVT.

Protection against air ingestion by SIS pumps is also a major concern with respect to recirculation and has been considered in the System 80+ design. The location and size of the suction lines in the IRWST have been chosen such that air entrainment is minimized. Pump air ingestion analysis is based on minimum submergence, maximum Froude number, and maximum pipe velocities. The available surface area used in determining the design coolant velocity has been calculated conservatively to account for blockage that may result as per Reg. Guide 1.82, Appendix A, Guidelines for Review of Sump Design and Water Sources for Emergency Core Cooling. The minimum water level in the IRWST has been conservatively calculated to be 75+6 (elevation). This water level allows for sufficient NPSH for the Containment Spray pumps and Safety Injection pumps operating at runout flow. A conservative margin has been provided between the elevation of the suction piping opening and this minimum water level to minimize the possibility of air ingestion. Applying the parameters of the IRWST to the equations in Reg. Guide 1.82, Appendix A, yields zero air ingestion at normal pump flowrates and less than 2% air ingestion at pump runout flowrates. The IRWST suction lines are also provided with vortex suppressors to aid in minimizing air ingestion by the SIS pumps. The guidelines in Appendix A of Reg. Guide 1.82 regarding the design of these vortex suppressors have been considered.

During normal full power operation, it is possible to perform a full flow test of the SIS and CSS pumps while taking suction from the IRWST and returning to the IRWST via a recirculation line (see SIS P&IDs in CESSAR-DC, Figures 6.3.2-1 A, B, C). This testing can verify the satisfactory hydraulic performance of the IRWST by running the pumps at runout flow with the minimum IRWST water level.

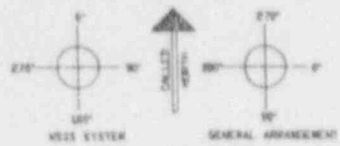
2.9.4 RESOLUTION

The design of the System 80+ IRWST and HVT assures that a clean and reliable source of borated water is available for ECCS recirculation. The arrangement of the IRWST within the System 80+ containment offers advantages over conventional sumps. Like sumps, the tank serves as the single source of water for SIS and CSS pump recirculation, but the protection afforded the SIS pumps against debris ingestion or blockage is significantly greater than in current designs. First, water in containment draining back to the IRWST must pass through a large trash rack before entering the HVT. The HVT serves as an effective solids trap for high density debris. Lower density debris that makes its way into the IRWST via the IRWST spillways encounters debris screens that filter fine particles from the SIS suction inlets. Each of the four SIS pumps have separate IRWST suction lines and each of the two CSS pumps takes suction from one of these four lines. Box screens at all four suction lines provide a final trap.

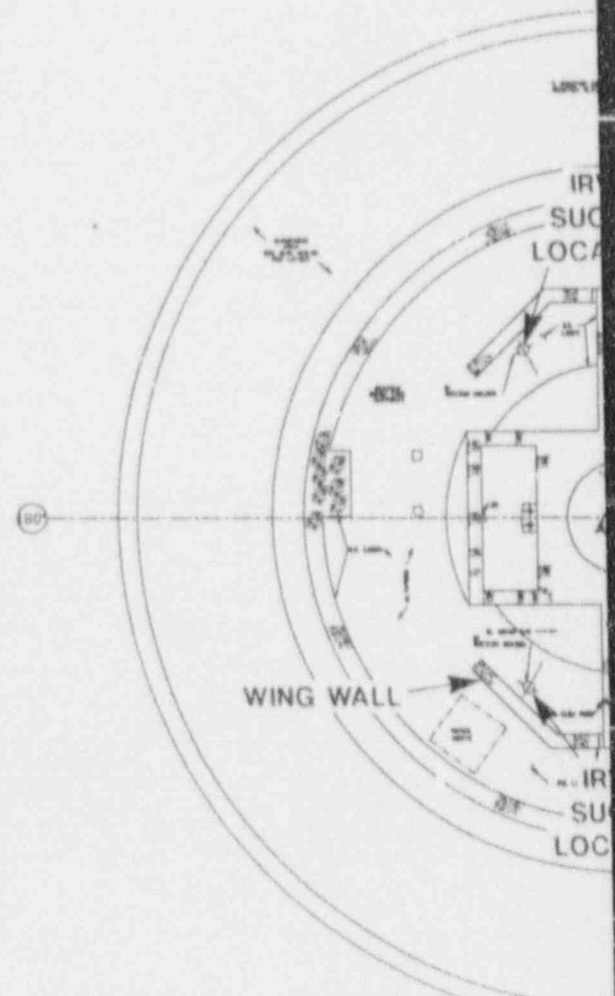
Multiple spillways are available to return water from the upper containment elevations to the IRWST. The drain pathways are fully redundant to assure recirculation capability. The location of the suction inlets within the IRWST provide additional protection against suction inlet damage and/or blockage.

Consideration has been given to IRWST hydraulic performance, the generation of potential debris and associated effects (including debris screen blockage), and the preservation of SIS pump NPSH during post-LOCA conditions in the overall design. The performance of the design is deemed acceptable with respect to these considerations.

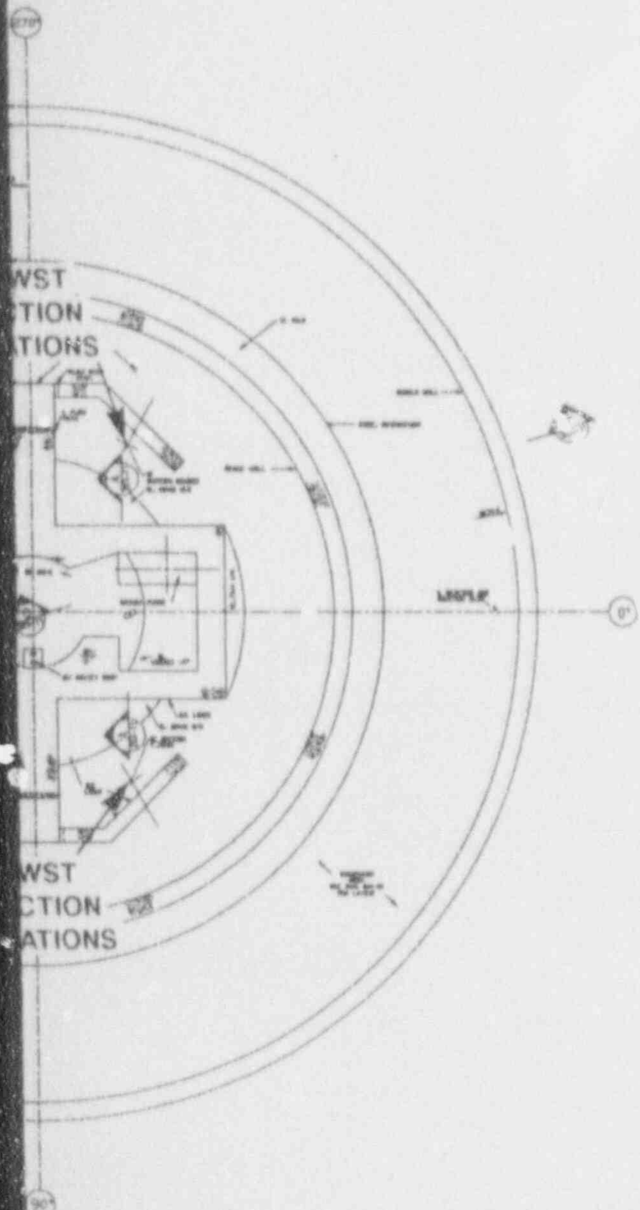
This issue has been resolved by design features of System 80+.



PLANT ORIENTATION



PLAN



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SYSTEM 80+™	LOCATIONS OF SAFETY INJECTION SYSTEM SUCTION IN IRWST	Figure 2.9-1

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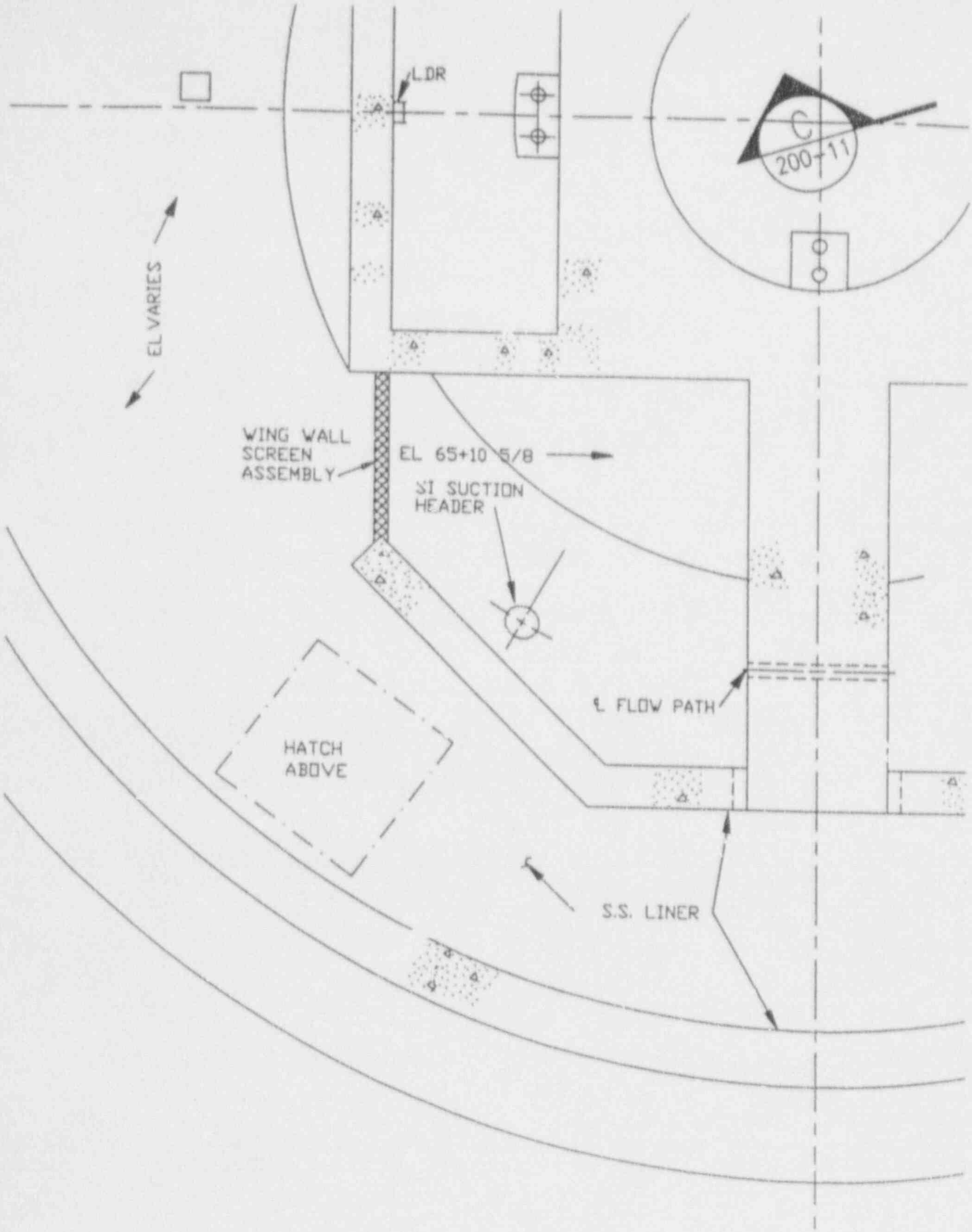
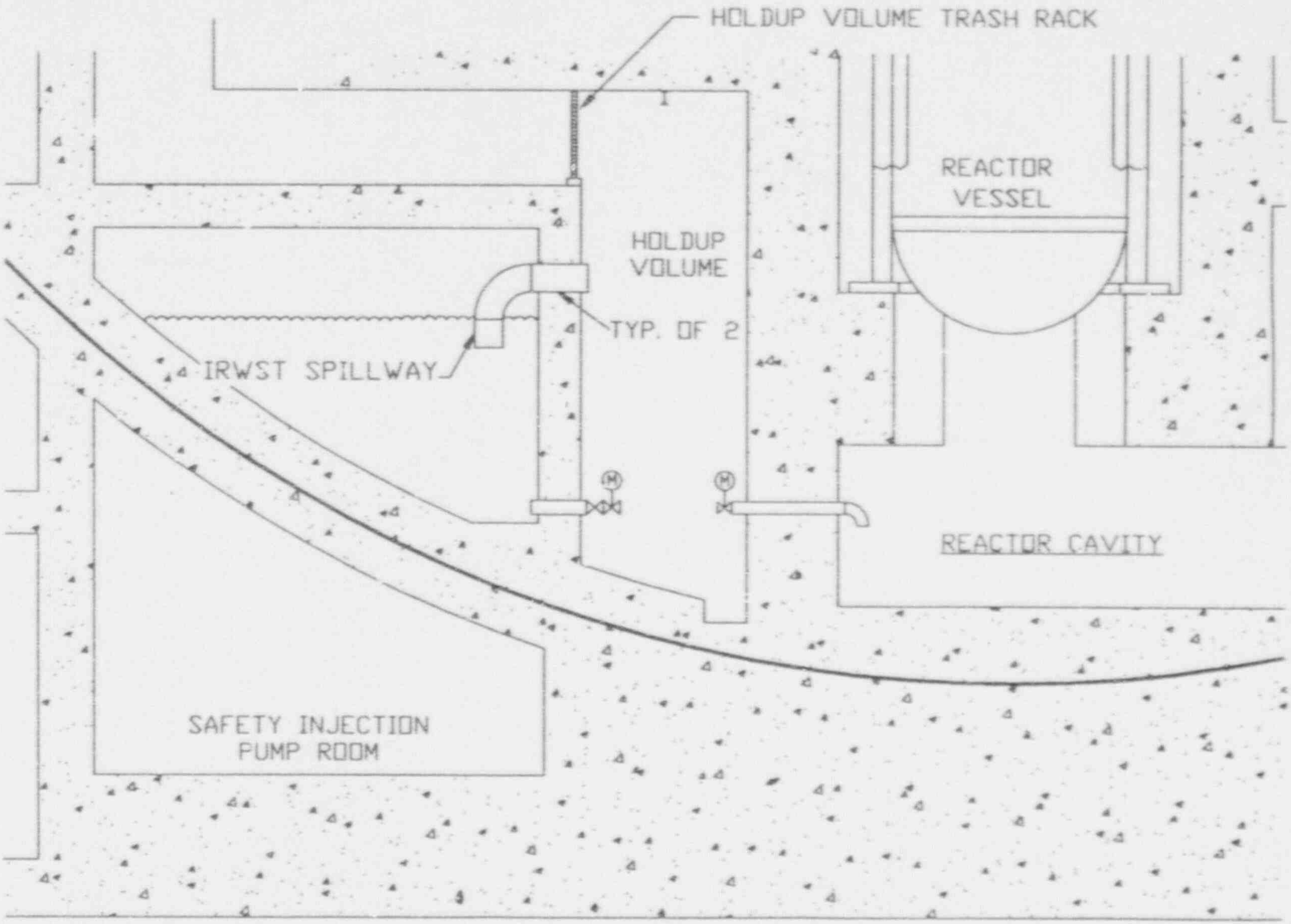


SYSTEM 80+™

LOCATION OF WING WALL DEBRIS SCREEN ASSEMBLIES

Figure 2.9-2



System 80+™

LOCATIONS OF TRASH RACK, DEBRIS SCREEN AND SPILLWAY FOR IRWST AND HVT

Figure 2.9-3

2.10 EFFECTS OF PWR UPPER INTERNALS

2.10.1 ISSUE

Events with the potential for loss of Decay Heat Removal (DHR) have initiated from plant configurations with the reactor vessel head removed, the refueling pool filled with water and the reactor upper internals still in place. Under these conditions, the reactor vessel upper internals may provide sufficient hydraulic resistance to natural circulation flow between the refueling pool and the reactor core to inhibit, or even prevent, the refueling pool water from cooling the core under circumstances when forced convection DHR has been lost.

2.10.2 ACCEPTANCE CRITERIA

When the reactor vessel head is off and the core and upper internals are in the vessel, any one or more of the following conditions shall be satisfied:

1. Demonstration by analysis that the time to boil exceeds the time required to evacuate and establish containment integrity; and

2. Demonstration by analysis that either a natural or forced circulation flow path, with or without heat exchangers, can be established to perform DHR for a sufficiently long period of time to allow plant operators to terminate the event.

2.10.3 DISCUSSION

An evaluation of the hydraulic flow resistance through the upper internals, of the time to boil and of the characteristics of the SDC System will be presented in the June 15, 1992 updated submittal of this report.

2.10.4 RESOLUTION

resolution of loss of DHR while the upper internals are in the sel will be presented in the June 15, 1992 updated submittal of s report.

2.11 FUEL HANDLING AND HEAVY LOADS

2.11.1 ISSUE

Questions have been raised regarding the potential for damage to fuel and safety related equipment due to dropping of heavy loads during plant shutdown. Related issues involve the transport of heavy loads within the reactor containment building and the spent fuel building. These include dropping the reactor vessel closure head and internals, dropping the head area cable trays [HACTS], accidental release of a fuel assembly, and movement of the spent fuel storage cask. Drop accidents involving primary NSSS piping are not considered, since by design piping is routed beneath the reactor refueling pool.

2.11.2 ACCEPTANCE CRITERIA

Fuel and safety related equipment shall not be subject to damage that may adversely effect public health. Also, fuel assemblies located within the reactor or within storage racks shall remain subcritical during and following postulated load drop accidents.

2.11.3 DISCUSSION

The transport of heavy loads within the containment building and the spent fuel building is controlled by integrating relevant design characteristics for the building and the handling equipment. Plant layout, equipment design and handling procedures are chosen to insure that heavy loads are restricted to preassigned travel zones. Equipment interlocks and procedures are also used to insure that load transport is accomplished in a predictable manner.

Specific issues associated with the transport of heavy loads within the containment building include movement of the reactor vessel closure head, the reactor internals, the HACTS, and individual fuel assemblies. Special measures are taken to safeguard these operations and mitigate the consequences of postulated load drop accidents.

Procedural guidance for raising the reactor closure head, as provided in CESSAR-DC Section 9.1.4.2.3.3, specifies that the fuel transfer tube valve be closed and that the pool water level follow the vertical movement of the closure head as it is raised from the reactor. This insures that the containment building remains isolated from the spent fuel pool building during transport of the closure head. Also, by isolating the containment building from the spent fuel building, the spent fuel pool is protected against drain down that might occur as a result of a postulated drop accident.

Evaluations have been performed which demonstrate that a postulated head drop, from its specified maximum lift height, onto the reactor vessel will not result in a significant risk to public safety. Though the reactor vessel and internals may sustain damage, the reactor vessel will remain filled and the fuel will remain covered and in a subcritical configuration. Evaluations of the reactor internals demonstrate that a drop accident involving the internals would be less severe than the postulated head drop accident.

Travel paths for the closure head and the internals, leading from the reactor vessel to the respective storage stands, are arranged so that the transported loads do not pass directly over the ICI seal table (Refer to Figure 2.11-1). If it is postulated that portions of these structures do impact the seal table, seal housings and guide tubing above the seal table, the resulting damage would be localized to these components. Under these conditions, the water level within the vessel will remain at the flange level. In addition should the reactor cavity pool seal be damaged to the extent that there will be significant pool drainage, the reactor vessel will remain filled.

The refueling machine is structurally designed to withstand the affects of design basis seismic motions. In addition, this machine is provided with interlocks which restrict machine movements to permissible zones as well as lock the fuel grapple in place. The refueling machine is designed to transport one fuel assembly at a time between the reactor core and the fuel transfer system. It is also designed to transport CEA rod and ICI disposal containers between an intermediate storage rack and the fuel transfer system. The grapple for the disposal containers is the same design as the one used for fuel assemblies.

The refueling machine is designed so that it can not pass over the top of the ICI seal table. This precludes the possibility of load drop accident involving a fuel assembly falling onto the ICI seal table. Also, during normal refueling operations the travel path is restricted so that it passes over the reactor cavity pool seal at predetermined locations. As a minimum, the pool seal is designed so that it will withstand without leakage a postulated fuel drop accident in these zones. If, for other postulated reasons, there is significant drainage of the pool, it is possible to rapidly lower a fuel assembly on the refueling machine grapple to an elevation which insures that it remains submersed in water. The assembly may be inserted into the reactor vessel or lowered into the deep end of the refueling pool, adjacent to the fuel transfer system.

The head area cable tray assembly [HACTS], which is used to route power and signal lines away from the reactor vessel closure head, is designed to be handled by the auxiliary hoist on the polar crane. The cable trays are raised vertically by this hoist from the installed position over the reactor vessel and moved to a storage position on top of either of the two steam generator walls. The trays are handled by four separate slings that are fastened to specially designed lift fixtures on the structural frame of the HACTS. All lifting components are designed in accordance with the criteria of NUREG-0612, Control of Heavy Loads at Nuclear Power plants. Prior to movement of the cable trays, the reactor must be in the shutdown mode and depressurized.

During a postulated load drop accident involving the impact of the HACTS onto the reactor vessel, the maximum impact energy is estimated to be about twenty percent of that associated with a of the reactor vessel closure head drop.

Though the reactor vessel may be damaged, the level of damage to the vessel and its support would be less severe than that associated with dropping the reactor vessel head. Furthermore, since the cable tray assembly is not a rigid structure, an appreciable fraction of the impact energy would be dissipated during plastic deformation of the HACTS itself.

For the postulated dropped cable tray accident, it is most probable that the HACTS would impact the closure head lift rig and the CEDM pressure housings. These structures are likely to be permanently deformed by bending and/or buckling. In some instances the pressure housings may also leak. The extent of damage, however, would not be sufficient to cause the reactor vessel to drain down, nor to adversely affect core criticality.

As with the containment building, special consideration is given to the transport of heavy loads and fuel assemblies in the fuel building. Also restrictions regarding the transport of heavy loads over fuel storage racks, and movement of the fuel shipping cask apply (Refer to Figure 2.11-2).

Transport of the fuel shipping cask within the spent fuel building is accomplished using a special high capacity hoist. The cask is transported using a staggered lift from the wash down area to the laydown area, where fuel loading takes place. This is done to limit the maximum drop height for the respective regions. In each case the floors and walls have been designed to withstand a postulated cask drop accident.

The spent fuel pool is connected to the cask laydown area by a gate, which will be closed to isolate the two zones during cask movement. The elevation of the gate is specified so that fuel located in the spent fuel storage racks would remain submerged following a postulated pool drain down through the gate.

The hoist used for transport of the fuel shipping cask is mechanically interlocked to prevent travel over the spent fuel pool. This interlock prevents the possibility of inadvertent movement of heavy loads ove· ᵗhe spent fuel storage racks.

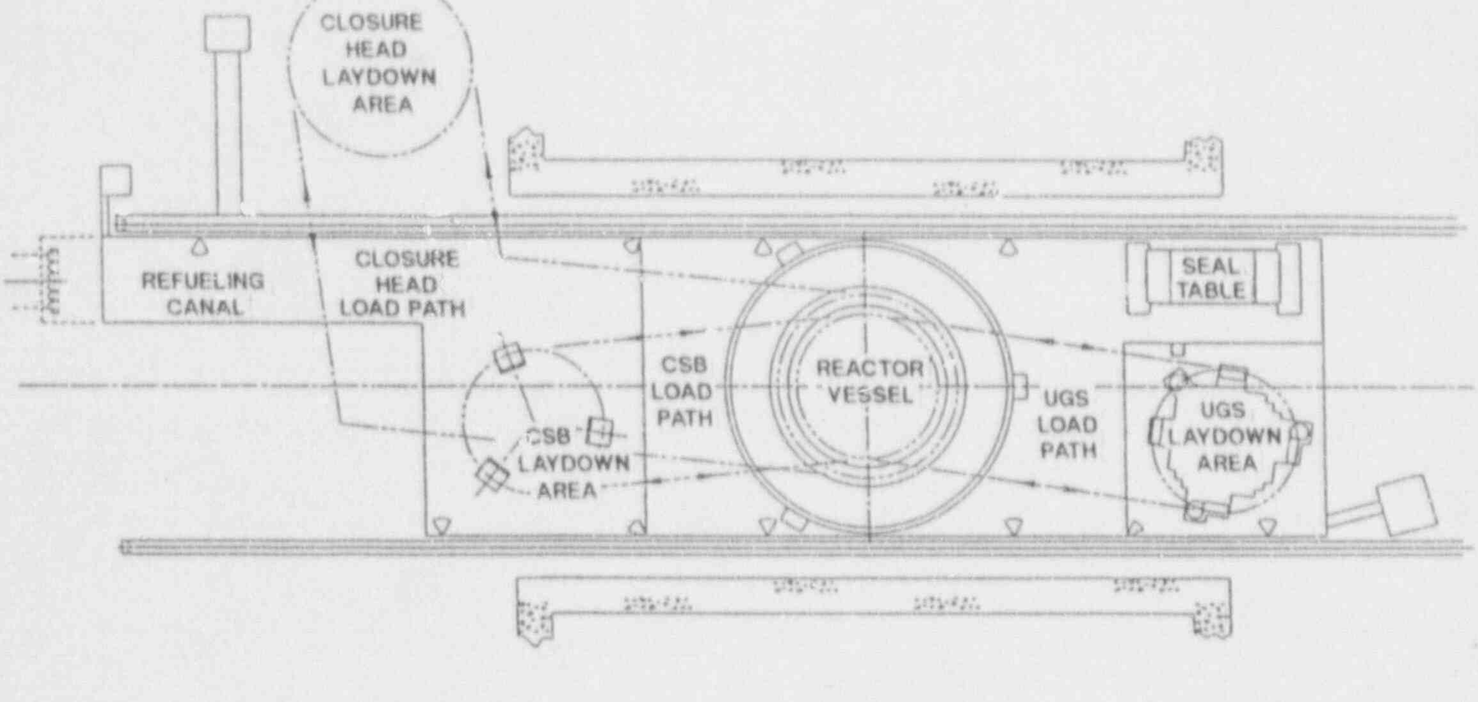
New fuel enters the fuel bui ng through a designated unloading area. It is handled and trans, ɔrted to new fuel storage racks by an intermediate capacity hoist. The lift height of the hoist is restricted to limit the maximum drop height of the fuel and tool onto the new fuel storage racks. Like the fuel storage cask hoist, this hoist is also mechanically interlocked to prevent travel over the spent fuel pool.

The fuel handling machine used in the fuel building is similar in design to the refueling machine. It is structurally designed to withstand seismic excitations. Also, it is provided with interlocks to control the movement of fuel within the pool.

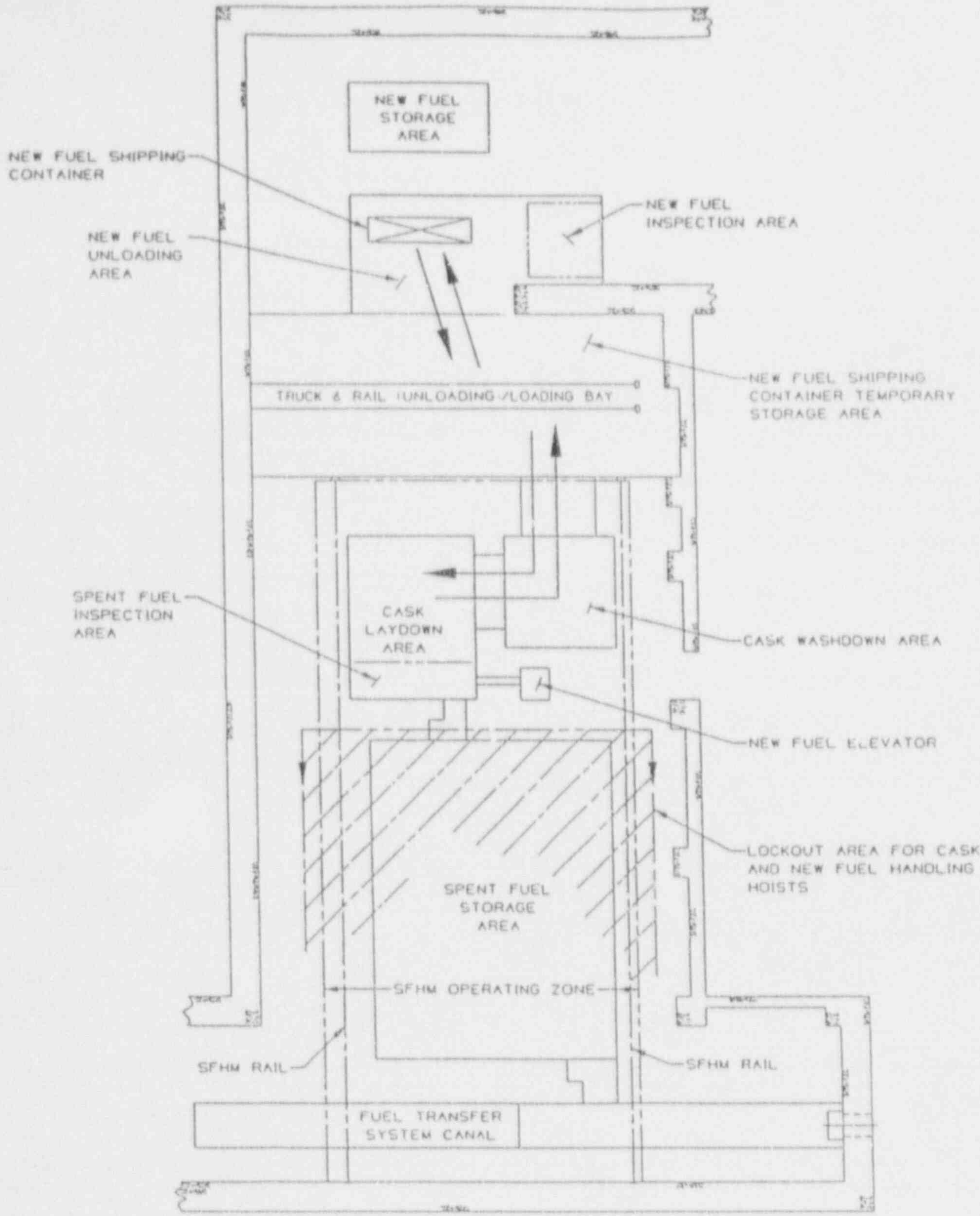
Both the new fuel and spent fuel storage racks are designed to withstand impact energies associated with postulated fuel drop accidents. They are designed to limit damage to the stored fuel and to maintain it in a subcritical configuration. Plant operating procedures also restrict the transport of loads over the fuel storage areas so that they do not exceed the design requirements for the storage racks. The consequences of dropping a spent fuel assembly in the spent fuel pool have been evaluated and the results presented in CESSAR-DC, Section 15.7.4. It has been shown that a postulated accident of this type would not present a risk to public health.

2.11.4 RESOLUTION

The issue of fuel handling and heavy loads is resolved for System - 80+ by the equipment design and building layout which satisfy applicable criteria and provide physical limitations to movement and by administrative limitations in Chapter 9 of CESSAR-DC.



SYSTEM 80+™	CONTAINMENT BUILDING LOAD HANDLING PATHS	Figure 2.11-1



SYSTEM 80+™	FUEL HANDLING BUILDING LOAD HANDLING PATHS	Figure 2.11-2

2.12 POTENTIAL FOR DRAINING THE REACTOR VESSEL

2.12.1 ISSUE

The issue is the risk of losing primary coolant from the reactor vessel during modes 2 through 6 (startup/shutdown, hot standby, hot shutdown, cold shutdown, and refueling). The safety significance of draining the coolant from the reactor coolant system during shutdown is that such an event can directly lead to voiding in the core and eventual core damage. The amount of damage would, among other factors, depend on the degree to which the core is uncovered and the decay heat level of the core at the time of uncovery. The draining of the reactor coolant system could also lead to a loss of decay heat removal cooling capability which in turn could lead to core uncovery.

2.12.2 ACCEPTANCE CRITERIA

The criteria utilized to evaluate the adequacy of the System 80+ design with respect to the potential for draining the reactor vessel are prevention, detection and mitigation. Prevention is the preferred criteria but in some instances (when draining potential cannot be eliminated) detection and mitigation are to be provided.

2.12.2.1 Prevention Criteria

1. The design shall prevent or inhibit the draining through the use of isolation valves, interlocks, and system alignment restrictions during the various modes of plant operation.

2. The design shall minimize the potential for component failure, inadvertent action, or human/operator error to result in the draining of the reactor vessel. Redundant components shall be provided as appropriate. The design shall provide instrumentation, overview displays, and alarms to clearly supply the operator with equipment status specific to shutdown modes.

3. Technical specifications and procedural guidance shall be provided to the plant owner/operator to assist in identifying plant conditions and configurations in modes 2-6 that could result in a potential primary coolant drainage event.

2.12.2.2 Detection Criteria

The design shall have the capability to detect, monitor and locate identified drainage paths that would occur in a time frame required to prevent a loss of decay heat removal or core uncovery. Appropriate instrumentation, displays and alarms shall be provided.

2.12.2.3 Mitigation Criteria

1. The design shall have the capability to mitigate the loss of primary coolant from the reactor including shutting-off of a drain path and the ability to provide the source and path for sufficient make-up.

2. Technical specifications and procedural guidance shall be provided to identify potential make-up water injection sources and paths in the event a drainage path does occur. Recovery actions shall be specified.

2 3 DISCUSSION

The primary coolant can potentially drain from the reactor via paths directly from the Reactor Coolant System (RCS) or via paths through systems interfacing the RCS. The plant mode of operation affects the potential for draining the reactor coolant system. This task is focused on shutdown risk. Shutdown risk encompasses operation when the reactor is subcritical or in transition between subcriticality and power operation up to five percent rated thermal power; i.e., for the System 80+ design between Mode 6 and Mode 2 where the modes of operation are defined in CESSAR-DC Chapter 16, Table 1.1.1. The plant mode prescribes and/or allows certain alignments and conditions (pressure, temperature and flow) within and between the RCS and interfacing systems.

The causes (initiators) of reactor vessel draining events are grouped into the following categories:

1. Components/equipment that fail to operate as intended. This could result from equipment malfunction (e.g., stuck open relief valve), or overpressurization (e.g., seal rupture or safety valve lift).

2. Human (operator) error such as misoperation of valves or pumps leading to a loss of coolant directly from the RCS or through systems connected to the RCS.

There are several factors that affect the likelihood and/or the ultimate consequence of a loss of primary coolant during shutdown. The plant configuration is one such factor. Plant configuration includes the following:

1. The alignment, during various modes of shutdown operation, of systems with the RCS and other systems. This affects possible paths for primary coolant flow from the reactor vessel.

2. The use of temporary seals in the RCS and interfacing systems during maintenance and refueling activities. The temporary seals include for example, nozzle dams. The failure of such

seals can open a path for leakage or close-off a mitigation path.

3. The elevations of water within the reactor vessel. Midloop operation reduces margin for recovery after initiation of a loss of primary coolant event relative to a more conservative water inventory.

4. The availability of mitigating systems. Assuming some drain path is opened, allowing primary coolant to escape, there must be sufficient make-up available and paths available to transport this make-up to replenish the water in the reactor vessel. The means to close the drainage path is also needed. It must be assured that maintenance activities do not take required mitigating systems out of service.

Another factor that can affect the potential for and consequence of a loss of primary coolant event during shutdown is the ability to quickly determine that a loss is occurring and the source of the loss. This would include temperature, pressure and water level monitoring. Such monitoring can also provide operators with information that could reduce the likelihood of failure.

An evaluation of the System 80+ plant arrangements and proposed operating configurations will complete this discussion and will be presented in the June 15, 1992 updated submittal of this report.

2.12.4 RESOLUTION

The resolution of the reactor vessel draining issue will be presented in the June 15, 1992 updated submittal of this report.

2.13 FLOODING AND SPILLS

2.13.1 ISSUE

Essential systems may be at higher risk for failure due to flooding and spills during shutdown because of the varied and interrelated maintenance activities that may be in progress simultaneously. Past events have involved, for example, spills from the component cooling water system, service water system, condensers, and refueling pool seals. The issue addressed here is the potential for loss of decay heat removal as a consequence of spills and internal flooding that may disable components of the shutdown cooling system.

2.13.2 ACCEPTANCE CRITERIA

The flood protection design will provide separation of redundant equipment to ensure decay heat removal (DHR) systems availability and capability are not precluded due to flooding and spills.

2.13.3 DISCUSSION

The flood protection provided insures a boundary of separation between redundant DHR systems. The separation includes components and structures to prevent the migration of water. Preventing the migration of water eliminates the potential for rendering redundant DHR equipment inoperable.

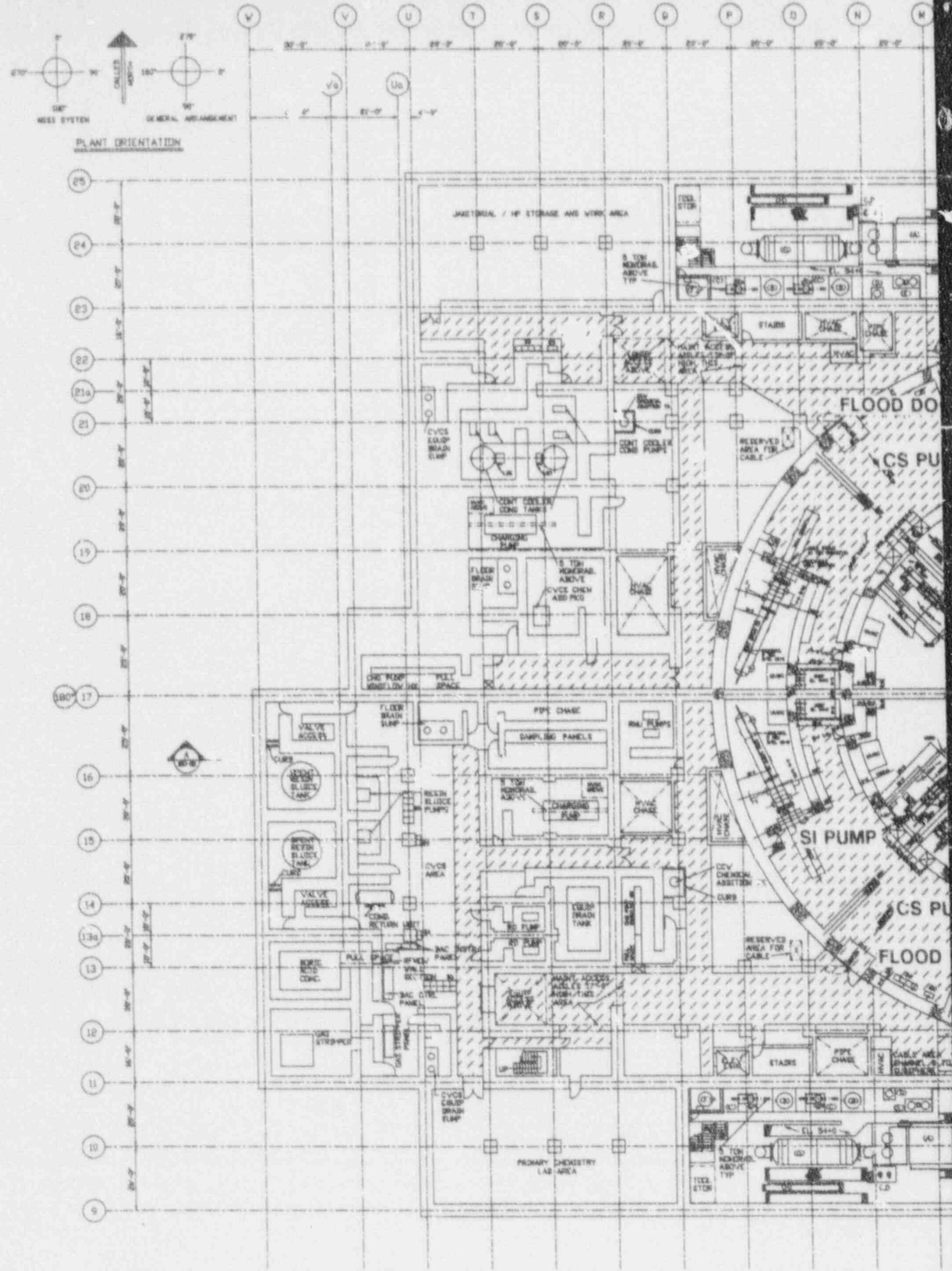
The System 80+ design provides separation and flood barriers to prevent the flood of redundant equipment. The design features a divisional separation. This divisional separation is a wall in the Nuclear Annex and the Reactor Building Sub-Sphere. The wall forms a barrier between the Division 1 and the Division 2 mechanical and electrical equipment. This wall contains no unsealed penetrations below the 70' elevation level. This wall is along column line 17 (see CESSAR-DC Figures 1.2-4 and 1.2-5, reproduced here as Figures 2.13-1 and 2.13-2). Additional separation of the divisions is provided by the floor drain systems. The sumps and floor drains located in the Nuclear Annex and the Reactor Building Sub-Sphere are divisionally separated. This design feature prevents the migration of floodwater from one division to the other through the floor drains.

The Systems 80+ design utilizes flood doors to provide separation within the same division. In the Reactor Building Sub-Sphere, the flood doors provide quadrant separation, therefore equipment is protected from floods within the same division. Flood doors also provide protection for Reactor Building Sub-Sphere Quadrants A and B from flooding outside the sub-sphere. This protects the Shutdown Cooling Systems from floods that could occur in the Nuclear Annex and migrate into the sub-sphere. Flood doors also provide protection for the Vital Electrical Equipment located in the Nuclear Annex on elevation 50′ (see Figure 2.13-1).

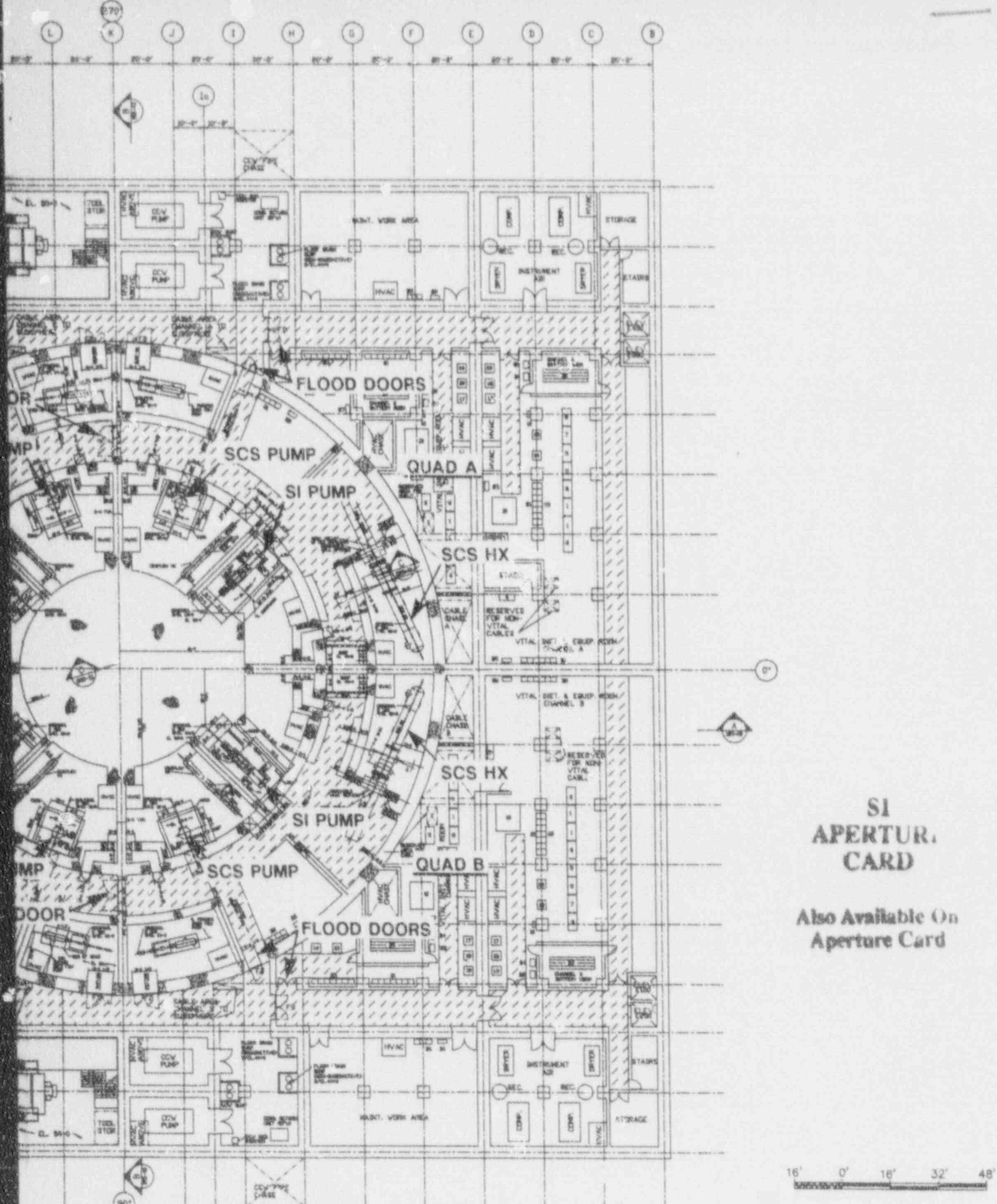
The System 80+ does not have any raw water systems inside the Nuclear Annex or Reactor Building. This design provides a significant contribution to flood protection because the flood sources are finite. Two significant sources of water are the Component Cooling Water System and the Emergency Feedwater Storage Tanks. Emptying the entire volume of water contained in a division of either of these systems will not flood above the 70′ elevation. Therefore, no migration of water to the other division or to other protected areas (e.g., electrical equipment) will occur due to the flood. This ensures the redundant systems and equipment located in the other division are available for decay heat removal.

2.13.4 RESOLUTION

The System 80+ flood protection design features are consistent with acceptance criteria outlined above in Section 2.13.2. These features resolve the issue of flooding and spills during shutdown operations on System 80+ by providing separation of redundant equipment required for decay heat removal. This separation provides the availability of DHR when a flood has occurred within the Nuclear Annex or Reactor Building Sub-Sphere.



PLANT ORIENTATION
NSSS SYSTEM
GENERAL ARRANGEMENT
JANITORIAL / HP STORAGE AND WORK AREA
FLOOD DO
CS PU
SI PUMP
CS PU
FLOOD
STAIRS
CVCS EQUIP DRAIN SUMP
CHARGING PUMP
PIPE CHASE
SAMPLING PANELS
CHARGING PUMP
VALVE ACCESS
RESIN SLUICE PUMPS
CVCS AREA
EQUIP DRAIN TANK
RESERVED AREA FOR CABLE
PRIMARY CHEMISTRY LAB AREA



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SYSTEM 80+™	NUCLEAR ISLAND DETAILED ARRANGEMENT PLAN AT EL. 50+0	Figure 2.13-1

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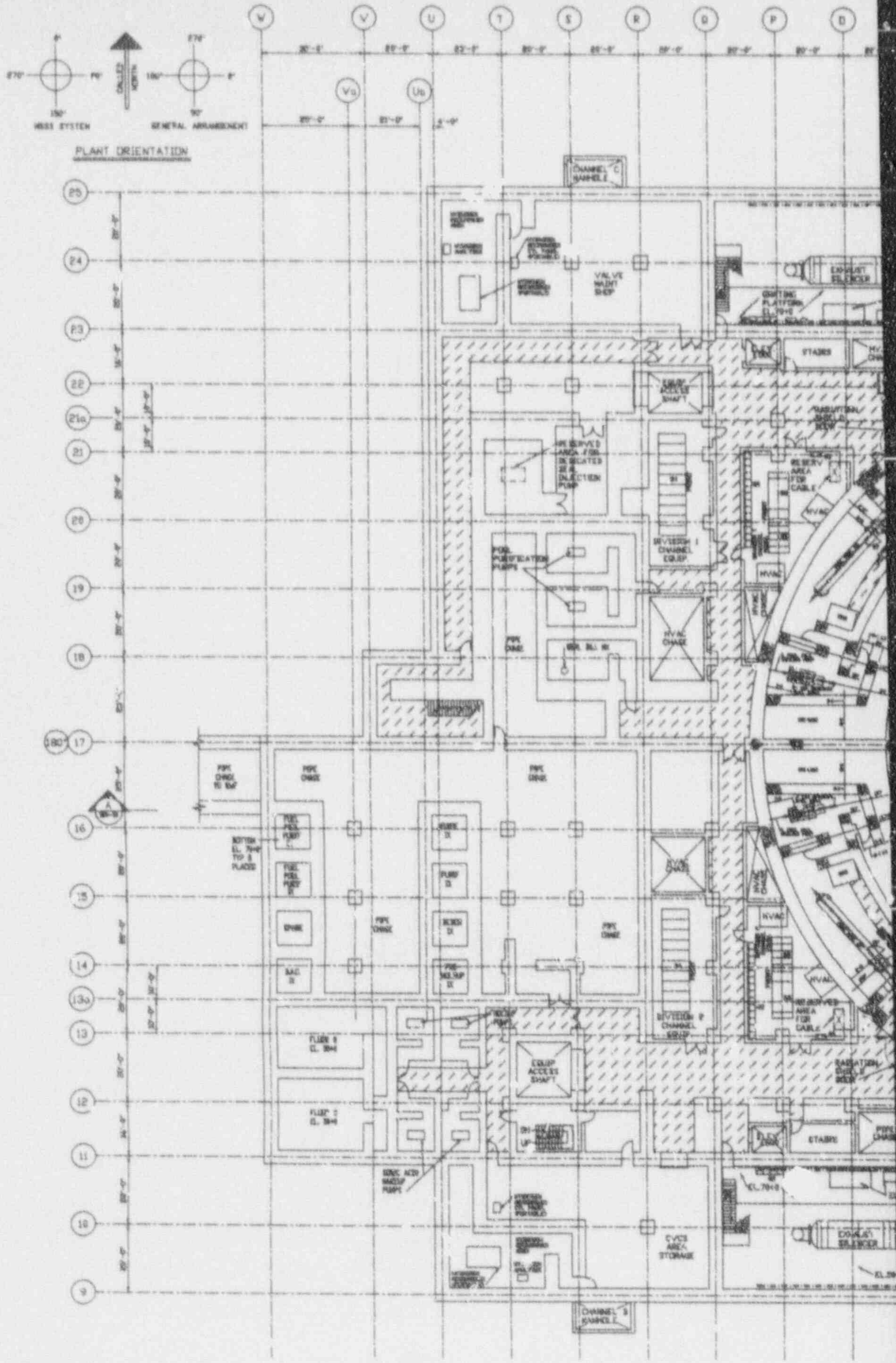
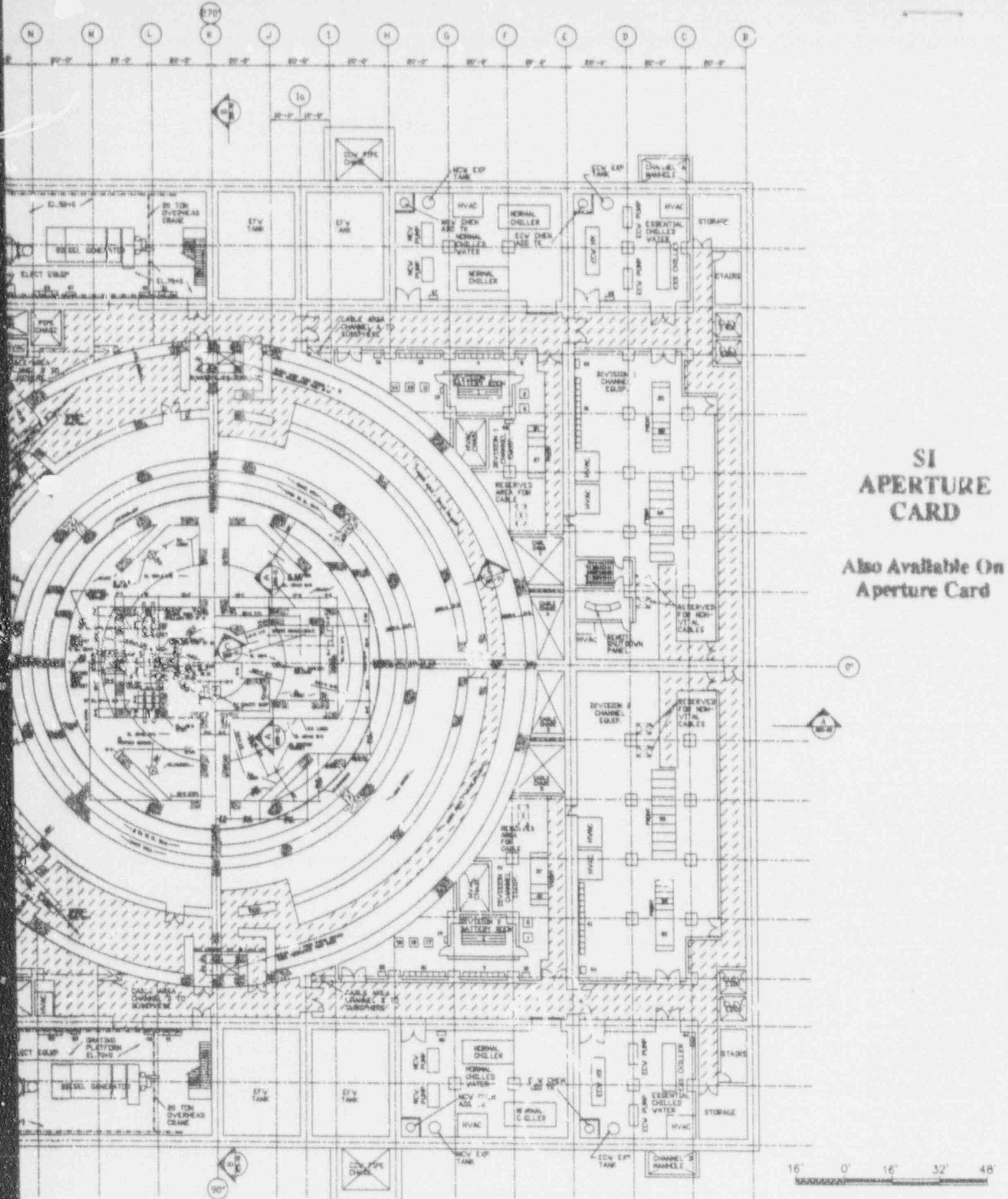


PLANT ORIENTATION
GENERAL ARRANGEMENT
CHANNEL 'C' MANHOLE
VALVE MAINT SHOP
EXHAUST SILENCER
STAIRS
EQUIP ACCESS SHAFT
DIVISION 1 CHANNEL EQUIP
DIVISION 2 CHANNEL EQUIP
HVAC CHASE
PIPE CHASE
HVAC
CVCS AREA STORAGE
CHANNEL 'B' MANHOLE



SYSTEM 80+ ™

NUCLEAR ISLAND DETAILED ARRANGEMENT
PLAN AT EL. 70+0

Figure
2.13-2

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3.0 PROBABILISTIC RISK ASSESSMENTS

3.1 INTRODUCTION

Emphasis has previously been on the safety of power plants during power operation. This was due to the fact that the plant is in this configuration most of the time and the core power, decay heat rate, and fission product inventory are highest at this time. The System 80+ PRA (Reference 7) documents the risk associated with the operation of this ALWR at normal power. More recently, people have been investigating the risk of plants during shutdown and refueling. During these modes of operation, the plant has lower decay heat rates and fission product inventory. The plant configuration is not as well defined as in full power operation because of the maintenance and testing that is being performed. This report documents the risk of operation of this plant in Modes 3 through 6. Modes 1 and 2 are covered in Reference 7.

The awareness of the risks of plant operation during refueling and maintenance outage developed slowly. During refueling the plants have low decay heat and also have a large primary coolant inventory with the refueling cavity flooded. The emphasis in the SARs was on operational events and operator training also emphasis power operation modes. Although no core damage has occurred during reactor outages, a few events have occurred which were precursors to more severe accidents. One of the first events to increase the awareness of risks in outages was the loss of the refueling cavity seal at Connecticut Yankee on August 21, 1984. In this event, 200,000 gallons of water quickly spilled into the containment, draining the refueling cavity. This event would have been more difficult to handle if the refueling had actually started. The seal failure had not been considered in the safety analysis.

Another event which increased the awareness of the risks during outages was the Vogtle 1 event on March 20, 1990. The event started with a truck backing into equipment in a switch yard causing a loss of power to the first auxiliary transformer. The second auxiliary transformer was out for maintenance. One of the diesels was also out for maintenance and the second failed to start. This combination of maintenance activities and failures let to a station blackout and loss of Residual Heat Removal (RHR). Under normal conditions, with the vessel filled or with the refueling cavity flooded, the operator would have many hours to restore RHR. In this incident, the plant was in mid-loop operation and the primary inventory was greatly reduced. When RHR was restored in 41 minutes, the primary coolant temperature had risen 46 deg F to 136 deg F. This incident demonstrates the unusual plant configurations that can exist during an outage and the risks associated with maintenance activities and mid-loop operation.

During 1991, there was a rash of incidents during shutdown. After four events occurred within six days in March, the NRC issued Information Notice No. 91-22 describing these events. One plant had two incidents during the same outage. There were at least seven events where loss of RHR occurred in 1991. All these events increased the awareness of the risks during outages.

The NRC requested additional information (Reference 8) from the ALWR participants pertaining to shutdown risk. The EPRI ALWR Utility Requirements Document thus was modified to include a risk assessment for shutdown modes (Reference 9). This analysis satisfies that requirement. This analysis uses a simplified event tree approach to estimate the core damage frequency (level 1 PRA) with only a simplified evaluation of the release frequencies and magnitudes.

The first step in the analysis was the identification of the initiating events of potential interest. This was done by first defining the plant conditions (in terms of physical parameters such as temperature, pressure, and inventory for the RCS) that will exist for different plant evolutions. For each of these operating conditions, general categories of initiating events were then defined. These initiating events were small LOCAs, Loss of RHR, Loss of AC, and Boron Dilution. The frequencies for these events were determined by operational history.

For each plant condition and initiating event, the plant and operator response was estimated based on the advanced instrumentation, procedures, technical specifications, and safety systems employed in the System 80+ design. The plant states and operator response were modeled and quantified using simplified event trees. The unavailability of each system was estimated using simplified assessments and adaptation of models developed for power operations. Care was taken in estimating the reliability of human actions. Earlier studies found that operator actions are one of the dominant factors in this analysis.

This Shutdown PRA was performed with the insight obtained from the previous PRAs. In 1981, NSAC-84 looked at the risk associated at Zion during outages. This study concluded that failures during reduced inventory operation accounted for 61% of the Core-Damage Frequency (CDF). Operator actions were required in almost all sequences. Operator failure to determine the proper actions to restore RHR accounted for 56% of the total CDF. Loss of RHR also accounted for 56% of the CDF. NUREG/CR-5015 tended to confirm the findings in NSAC-84. The Seabrook Shutdown PRA concluded that 82% of the CDF was due to loss of RHR and 71% was from reduced inventory operation. The study also showed that early health risks were dominated by LOCAs with the containment open. The NRC's Shutdown PRA for Surry (as summarized in NUREG-1449) showed the importance of plant specifics such as the controls of the ADVs, and the response to Generic Letter 88-17. The insights gained from reviewing these PRAs helped in analyzing the System 80+ plant.

Section 3.2 contains a discussion of the methodology. The initiating event evaluation is evaluated in Section 3.3. The accident sequence determination is given in Section 3.4. With the data in Section 3.5 and the system analysis in Section 3.6, the accident sequences are quantified in Section 3.7. The consequence analysis is given in Section 3.8. Sections 3.2 and 3.3 will be provided in the June 15, 1992 updated submittal of this report. The remaining sections will be provided in the final July 31, 1992 updated submittal of this report.

4.0 APPLICABILITY OF CHAPTER 15 ANALYSES

4.0.1 FORMAT AND CONTENT

The purpose of this section is to present evaluations which confirm the applicability of the analyses presented in Chapter 15 of CESSAR-DC to shutdown Modes (Mode 2 subcritical or Modes 3 through 6) for the System 80+ Standard Design.

The approach used in the documentation of the events in Chapter 15 of CESSAR-DC is to present the results for the events with the most adverse consequences. As a result, reference is most frequently to events postulated to occur in Mode 1 or Mode 2 critical. Only in certain cases which intrinsically involve shutdown Modes (e.g. startup of an inactive reactor coolant pump) are shutdown Modes stressed. The purpose of this section is to ensure that all possible Modes have been treated in the documentation for Chapter 15 events.

The following sections have been organized to parallel the sections of Chapter 15 of CESSAR-DC. For example, Section 4.1.1 treats the same group of initiating events that are documented in Section 15.1.1 of CESSAR-DC.

4.1 INCREASE IN HEAT REMOVAL BY THE SECONDARY SYSTEM

4.1.0 INTRODUCTION

The purpose of this section is to present evaluations which confirm that all increase in heat removal by the secondary system events postulated to be initiated in a shutdown Mode have acceptable consequences for the System 80+ Standard Design.

Section 15.1 of CESSAR-DC documents results which show that all increase in heat removal by the secondary system events have acceptable consequences if they are explicitly postulated to occur in Mode 1 or Mode 2 critical for the System 80+ Standard Design. This is demonstrated for steam system piping failures inside and outside containment (Section 15.1.5) by analyses for both Mode 1 and Mode 2 critical. The choice of initial conditions for the other analyses of Section 15.1 to minimize the transient DNBR for any operating condition ensures that the results presented bound those for all of Mode 1 and Mode 2 critical.

4.1.1 DECREASE IN FEEDWATER TEMPERATURE

Evaluation of decrease in feedwater temperature events initiated in a shutdown Mode is in progress. It is expected that the consequences of these events will be bounded by the most adverse consequences presented for the events in Section 15.1.4 of CESSAR-DC.

The results of this evaluation are to be submitted to the NRC in the June 15, 1992 updated submittal of this report.

4.1.2 INCREASE IN FEEDWATER FLOW

Evaluation of increase in feedwater flow events initiated in a shutdown Mode is in progress. It is expected that the consequences of these events will be bounded by the most adverse consequences presented for the events in Section 15.1.4 of CESSAR-DC. The results of this evaluation will be presented in the June 15, 1992, updated submittal of this report.

4.1.3 INCREASED MAIN STEAM FLOW

As noted in Section 15.1.3 of CESSAR-DC, the steam flow due to an increased main steam flow event is the same as (or less than) that due to an inadvertent opening of a steam generator relief or safety valve event. Further, there are no other differences between these events which affect their consequences. Therefore, the conclusions of Section 4.1.4 of this report apply also to an increased main steam flow event.

4.1.4 INADVERTENT OPENING OF A STEAM GENERATOR RELIEF OR SAFETY VALVE

Evaluation of inadvertent opening of a steam generator relief or safety valve events initiated in a shutdown Mode is in progress. It is expected that the consequences of these events will be bounded by the most adverse consequences presented for the events in Section 15.1.4 of CESSAR-DC. The results of this evaluation will be presented in the June 15, 1992 updated submittal of this report.

4.1.5 STEAM SYSTEM PIPING FAILURES INSIDE AND OUTSIDE CONTAINMENT

Evaluation of steam system piping failures inside and outside containment, initiated in a shutdown Mode, is in progress. It is expected that the consequences of these events will be bounded by the most adverse consequences presented for the events in Section 15.1.5 CESSAR-DC. The results of this evaluation will be presented in the June 15, 1992 updated submittal of this report.

4.2 DECREASE IN HEAT REMOVAL BY SECONDARY SYSTEM

4.2.0 INTRODUCTION

The purpose of this section is to present evaluations which confirm that all decrease in heat removal by the secondary system events postulated to be initiated in a shutdown Mode have acceptable consequences for the System 80+ Standard Design.

Section 15.2 of CESSAR-DC documents results which show that all decrease in heat removal by the secondary system events have acceptable consequences if they are explicitly postulated to occur in Mode 1 or, in general, in Mode 2 critical for the System 80+ Standard Design. If there is a question as to whether an event has already been considered in Mode 2 critical, however, this Mode is also included in the evaluation.

The focus of the evaluations presented in this section is on ensuring that the peak primary and secondary pressures are less than 110% of their design pressures, that the pressure-temperature limits for brittle fracture of the reactor coolant system (RCS) are not violated, and that fuel integrity is maintained for the events considered. Fuel performance, as measured by the departure from nucleate boiling ratio (DNBR), is used for verification of fuel integrity.

4.2.1 LOSS OF EXTERNAL LOAD

Since the turbine is not on line, a loss of external load is not possible in a shutdown Mode.

4.2.2 TURBINE TRIP

Since the turbine is not on line, a turbine trip is not possible in a shutdown Mode.

4.2.3 LOSS OF CONDENSER VACUUM (LOCV)

Evaluation of loss of condenser vacuum (LOCV) events initiated in a shutdown Mode is in progress. It is expected that the consequences of these events will be acceptable. The results of this evaluation will be presented in the June 15, 1992 updated submittal of this report.

4.2.4 MAIN STEAM ISOLATION VALVE CLOSURE

The comparison between main steam isolation valve (MSIV) closure and LOCV events made in Section 15.2.4 of CESSAR-DC applies for shutdown Modes, also. The evaluation of the LOCV event presented in Section 4.2.3 assumes a faster reduction in steam flow rate than would result from MSIV closure. The consequences of the MSIV closure event are, therefore, no more adverse in shutdown Modes than those for the LOCV presented in Section 4.2.3.

4.2.5 STEAM PRESSURE REGULATOR FAILURE

This event does not apply to the SYSTEM 80+ Standard Design and is, therefore, not evaluated here.

4.2.6 LOSS OF NON-EMERGENCY AC POWER TO THE STATION AUXILIARIES

The results of the loss of non-emergency AC power to the station auxiliaries (LOAC) event are the same as those for the loss of reactor coolant flow event presented in Section 4.3.1.

4.2.7 LOSS OF NORMAL FEEDWATER FLOW

A postulated loss of normal feedwater flow (startup feedwater flow) during a shutdown Mode would be less adverse than the LOCV event. The analysis assumptions for the LOCV event result in termination of steam flow, as well as termination of startup feedwater flow, causing a more severe decrease in heat removal by the secondary system. The consequences of the loss of normal feedwater flow are, therefore, bounded by the consequences of the LOCV event for shutdown Modes.

4.2.8 FEEDWATER SYSTEM PIPE BREAKS

Depending on the break size and location and the response of the feedwater system, the effect of a postulated feedwater system pipe break can vary from a heatup to a cooldown of the RCS. Based on the same arguments given in Section 15.2.8.1 of CESSAR-DC, the heatup event is considered in this section. The cooldown potential would be worse for a steam line break, which is discussed in Section 4.1.5. A heatup event is mitigated by the pressurizer safety valves or the shutdown cooling (SCS) relief valves when RCS temperatures are above or below, respectively, the LTOP enable/disable temperatures, the main steam safety valves, and the emergency feedwater (EFW) system.

A feedwater system pipe break postulated to occur in a shutdown Mode would result in less severe consequences than the event documented in Section 15.2.8 of CESSAR-DC due to the lower initial reactor power level. DNB is not of concern due to the low initial core power levels, in addition to the pressurization following event initiation. Further, the EFW system is capable of removing decay heat event with only one EFW pump in operation. The dominant factor which determines peak primary and secondary pressures is the magnitude of the energy mismatch between the primary and secondary systems. This mismatch is very much less for events postulated to occur in shutdown Modes than for the event of Section 15.2.8 of

CESSAR-DC. There is, therefore, no approach to 110% of steam generator design pressure, since the maximum pressure will be much less than that for the full power event. For the same reason there is no approach to the criterion of 110% of RCS design pressure when RCS temperatures are above the LTOP enable/disable temperatures. For RCS temperatures below the LTOP enable/disable temperatures the design of the SCS relief valves ensures that the pressure-temperature limits for brittle fracture of the RCS are not violated.

4.3 DECREASE IN REACTOR COOLANT FLOW RATE

4.3.0 INTRODUCTION

The purpose of this section is to present evaluations which confirm that all decrease in reactor coolant flow rate events postulated to be initiated in a shutdown Mode have acceptable consequences for the System 80+ Standard Design.

Section 15.3 of CESSAR-DC documents results which show that all decrease in reactor coolant flow rate events have acceptable consequences if they are explicitly postulated to occur in Mode 1 or Mode 2 critical for the System 80+ Standard Design. This is demonstrated for total loss of reactor coolant flow (Section 15.3.1) by explicit analyses. The choice of initial conditions for the other analyses of Section 15.3 to minimize the transient DNBR for any operating condition ensures that the results presented bound those for all of Mode 1 and Mode 2 critical.

Decrease in reactor coolant flow rate in Modes 4 through 6 when the shutdown cooling system (SCS) is being used for decay heat removal is addressed in Section 2.4 as integral to the evaluation of loss of decay heat removal capability. Therefore this section addresses events postulated to be initiated in Mode 2 subcritical, Mode 3 or Modes 4 and 5 when the SDS is not being used.

4.3.1 TOTAL LOSS OF REACTOR COOLANT FLOW

Evaluation of the factors affecting the consequences of the total loss of reactor coolant flow event shows that if this event is postulated to be initiated in shutdown Modes, the results are less adverse than those of the CESSAR-DC Chapter 15.3 full power event.

Loss of offsite power is the postulated initiating event for the total loss of reactor coolant flow event in Modes 1 or 2. All systems available to mitigate the Mode 1 transient are available in Mode 2 subcritical. The initial conditions for Mode 2 subcritical include four pumps operating, temperature and pressure identical to Mode 1, a low power level, and total energy stored in the reactor core much less than at full power. Therefore, a very large margin to DNB exists at the initiation of the event and the heat to be removed during the event is much less than for an event initiated at full power. The minimum DNBR for this event is substantially higher for Mode 2 than that for the full power case.

In addition, the initiating event could be postulated to be loss of power to any operating RCPs in Mode 3 or in Modes 4 or 5 when the SCS is not being used. Natural circulation is, however, sufficient for the removal of decay heat in these Modes. Thus no approach to DNB would occur. The consequences of the Chapter 15.3 full power event are therefore more adverse than an event in these Modes.

The full power four pump loss of flow transient produces RCS and steam generator pressures which are less than 110% of their design values. Transients postulated to be initiated in Modes 2, 3 or 4 when RCS temperatures are above the LTOP enable/disable temperatures, and for which the rate of heat production is orders of magnitude below full power values, would therefore yield even greater margins to the design pressure values. For transients postulated to be initiated in Modes 4 or 5 with RCS temperatures below the LTOP enable/disable temperatures, the design of the SDC relief valves ensures that the pressure-temperature limits for brittle fracture are not violated.

4.3.2 FLOW CONTROLLER MALFUNCTION CAUSING FLOW COASTDOWN

This event is categorized as a Boiling Water Reactor event in SRP 15.3.2 and is, therefore, not evaluated here.

4.3.3 SINGLE REACTOR COOLANT PUMP ROTOR SEIZURE WITH LOSS OF OFFSITE POWER

The major parameter of concern for the single reactor coolant pump rotor seizure with loss of offsite power event documented in Section 15.3.3 of CESSAR-DC is the minimum hot channel DNBR. This is minimized by higher power conditions. Lower power Modes would therefore, not produce any conditions which are more adverse than those presented in CESSAR-DC.

The second parameter of concern is the peak RCS pressure attained. The full power single reactor coolant pump rotor seizure with loss of offsite power event produces RCS pressures which are less than 110% of their design values. Transients postulated to be initiated in Modes 2, 3, or 4 when RCS temperatures are above the LTOP enable/disable temperatures, and for which the rate of heat production is orders of magnitude below full power values, would therefore yield even greater margins to the design pressure values. For transients postulated to be initiated in Modes 4 or 5 with RCS temperatures below the LTOP enable/disable temperatures the design of the SCS relief valves ensures that the pressure-temperature limits for brittle fracture of the RCS are not violated.

In addition, at these low power levels a concurrent turbine trip which results in a loss of offsite power is not an issue since the turbine is not in operation below 5% power.

4.3.4 REACTOR COOLANT PUMP SHAFT BREAK WITH LOSS OF OFFSITE POWER

Since a postulated reactor coolant pump shaft break (SB) transient results in a less rapid flow coastdown than a rotor seizure (RS) event, the results of the SB event are bounded by those of the evaluation of Section 4.3.3.

4.4 REACTIVITY AND POWER DISTRIBUTION ANOMALIES

4.4.0 INTRODUCTION

The purpose of this section is to present evaluations which confirm that all reactivity and power distribution anomaly events postulated to be initiated in a shutdown Mode have acceptable consequences for the System 80+ Standard Design.

4.4.1 UNCONTROLLED CONTROL ELEMENT ASSEMBLY WITHDRAWAL FROM SUBCRITICAL OR LOW POWER CONDITIONS

Analyses are in progress to demonstrate that the consequences of an uncontrolled CEA withdrawal initiated from a shutdown Mode is acceptable. It is expected that the consequences of these analyses will be acceptable. The results of these analyses will be presented in the June 15, 1992 updated submittal of this report.

4.4.2 UNCONTROLLED CONTROL ELEMENT ASSEMBLY WITHDRAWAL AT POWER

This event is not an issue for shutdown Modes since its intent is to examine high power operation only.

4.4.3 SINGLE CONTROL ELEMENT ASSEMBLY DROP

A postulated single control element assembly drop at power is analyzed for approach to the DNBR limit in CESSAR-DC, Section 15.4.3. For Mode 2 subcritical through the other subcritical Modes, a dropped rod only adds more negative reactivity to an already subcritical core and is, therefore, much less adverse than the full power event documented in CESSAR-DC.

4.4.4 STARTUP OF AN INACTIVE REACTOR COOLANT PUMP

Chapter 15.4.4 of CESSAR-DC presents analysis of startup of an inactive reactor coolant pump events which show that events postulated to be initiated in Modes 3 through 6 have acceptable consequences for the System 80+ Standard Design. (Operation with less than 4 RCPs is not permitted in Modes 1 or 2.)

4.4.5 FLOW CONTROLLER MALFUNCTION CAUSING AN INCREASE IN BWR CORE FLOW RATE

This event is categorized as a Boiling Water Reactor event in SRP 15.4.5 and is, therefore, not evaluated here.

4.4.6 INADVERTENT DEBORATION

Analyses of inadvertent deboration events in shutdown Modes have been presented in Chapter 15 of CESSAR-DC. An additional evaluation which considers rapid deboration is presented in Section 2.6 of this report.

4.4.7 INADVERTENT LOADING OF A FUEL ASSEMBLY INTO THE IMPROPER POSITION

This event has been evaluated in Section 15.4.7 of CESSAR-DC and is not mode dependent.

4.4.8 CONTROL ELEMENT ASSEMBLY (CEA) EJECTION

The core remains subcritical for CEA ejection events postulated to be initiated in a shutdown Mode. The technical specification on K_{N-1} (see Section 2.2) requires that the highest worth CEA be excluded from the subcriticality calculation for Modes 2 through 5. Thus, even if the highest worth CEA were to be assumed to be ejected from the core during shutdown, the core would remain subcritical. The full power event of Section 15.4.8 of CESSAR-DC is, therefore, the limiting CEA ejection event.

4.5 <u>INCREASE IN RCS INVENTORY</u>

4.5.0 INTRODUCTION

The purpose of this section is to present evaluations which demonstrate that all increase in RCS inventory events postulated to be initiated in a shutdown Mode have acceptable consequences for the System 80+ Standard Design.

Section 15.5 of CESSAR-DC documents results which show that all increase in RCS inventory events have acceptable consequences if they are explicitly postulated to occur in Mode 1 or, in general, in Mode 2 critical for the System 80+ Standard Design. If there is a question as to whether an event has already been considered in Mode 2 critical, however, this Mode is also included in the evaluation.

The focus of the evaluations presented in this section is on ensuring that the peak primary pressure is less than 110% of design pressure and that the pressure-temperature limits for brittle fracture of the RCS are not violated. Fuel performance, as measured by the departure from nucleate boiling ratio (DNBR), is used for verification of fuel integrity. Peak secondary pressure is also evaluated, as necessary, to ensure it remains less than 110% of its design pressure.

4.5.1 INADVERTENT OPERATION OF THE ECCS

The evaluation presented in Section 15.5.1 of CESSAR-DC establishes that a postulated inadvertent operation of the ECCS would have acceptable consequences for any Mode for the System 80+ Standard Design.

4.5.2 CVCS MALFUNCTION-PRESSURIZER LEVEL CONTROL SYSTEM MALFUNCTION WITH LOSS OF OFFSITE POWER

Peak RCS pressure due to a postulated malfunction/actuation of a charging pump in Modes 2 through 4 (before LTOP is active), is well within 110% of design pressure. Further, the pressure-temperature limits for brittle fracture of the RCS are not challenged for a postulated malfunction/actuation of a charging pump in Modes 4 through 6 (when LTOP is active or when the reactor vessel head is off).

In Modes 2 subcritical through 4, before shutdown cooling and LTOP are placed in service, the peak RCS pressure is limited by the pressurizer safety valves. Since only decay heat exists under these conditions, the peak pressures during the CVCS malfunction event would be substantially less severe than the Mode 1 case described in CESSAR-DC. It should be noted that the centrifugal charging pumps are protected from runout at low RCS pressures by design.

In Modes 4, 5, and 6 when LTOP is active (or in Mode 6 with the reactor vessel head off and LTOP not active), the CVCS malfunction event would be less limiting than the inadvertent SIS actuation described in Section 4.5.1 above, due to the much lower flow from one charging pump versus the four SI pumps.

4.6 <u>DECREASE IN REACTOR COOLANT SYSTEM INVENTORY</u>

4.6.0 INTRODUCTION

The purpose of this section is to present evaluations which demonstrate that all decrease in RCS inventory events postulated to be initiated in a shutdown Mode have acceptable consequences for the System 80+ Standard Design.

Section 15.6 of CESSAR-DC documents results which show that all decrease in RCS inventory events have acceptable consequences if they are explicitly postulated to occur in Mode 1 or, in general, in Mode 2 critical for the System 80+ Standard Design. If there is a question as to whether an event has already been considered in Mode 2 critical, however, this Mode is also included in the evaluation.

Fuel performance, as measured by the departure from nucleate boiling ratio (DNBR), is used for verification of fuel integrity.

4.6.1 INADVERTENT OPENING OF A PRESSURIZER SAFETY/RELIEF VALVE

The LOCA evaluations presented in Section 4.6.5 of this report show that the inadvertent opening of a pressure safety/relief valve is at a non-limiting location for LOCAs postulated to occur in shutdown Modes. The inadvertent opening of a pressurizer safety valve as described in the Standard Review Plan 15.6.1 is, therefore, a non-limiting event in the Safety Injection System evaluations.

4.6.2 DOUBLE-ENDED BREAK OF A LETDOWN LINE OUTSIDE CONTAINMENT

Evaluation of double-ended break of a letdown line outside containment event initiated in a shutdown Mode is in progress. It is expected that the consequences of this event will be bounded by the most adverse consequences presented for the events in Section 15.6.2 of CESSAR-DC. The results of this evaluation will be presented in the June 15, 1992 updated submittal of this report.

4.6.3 STEAM GENERATOR TUBE RUPTURE

Evaluation of steam generator tube rupture events initiated in a shutdown Mode is in progress. It is expected that the consequences of these events will be bounded by the most adverse consequences presented for the events in Section 15.6.3 of CESSAR-DC. The results of this evaluation will be presented in the June 15, 1992 updated submittal of this report.

4.6.4 RADIOLOGICAL CONSEQUENCES OF MAIN STEAM LINE FAILURE OUTSIDE CONTAINMENT (BWR)

The radiological consequences of main steam line failure outside containment (BWR) do not apply to the System 80+ Standard Design and are, therefore, not evaluated here.

4.6.5 LOSS-OF-COOLANT ACCIDENT

Consequences of a LOCA initiated from Modes 2 through 4 are bounded by the consequences reported for a LOCA from Mode 1 since the containment spray and annulus ventilation systems which are available in Mode 1 are also available in Modes 2 through 4.

Evaluation of the consequences of a LOCA initiated from Modes 5 and 6 is in progress and will be presented in the June 15, 1992 updated submittal of this report.

4.7 RADIOACTIVE MATERIAL RELEASE FROM A SUBSYSTEM OR COMPONENT

4.7.0 INTRODUCTION

The purpose of this section is to present evaluations which confirm that all radioactive material release from a subsystem or component events postulated to be initiated in a shutdown Mode have acceptable consequences for the System 80+ Standard Design.

4.7.1 RADIOACTIVE GAS WASTE SYSTEM FAILURE

This section of the Standard Review Plan has been deleted (Reference 26 of Section 15.0)

4.7.2 RADIOACTIVE LIQUID WASTE SYSTEM LEAK OR FAILURE

This section of the Standard Review Plan has been deleted (Reference 26 of Section 15.0)

4.7.3 POSTULATED RADIOACTIVE RELEASES DUE TO LIQUID CONTAINING TANK FAILURES

This event has been evaluated in Section 15.7.3 of CESSAR-DC and is not Mode dependent.

4.7.4 FUEL HANDLING ACCIDENT

This event has been evaluated in Section 15.7.4 of CESSAR-DC and is not Mode dependent.

4.7.5 SPENT FUEL CASK DROP ACCIDENTS

This event has been evaluated in Section 15.7.4 of CESSAR-DC and is not Mode dependent.

5.0 APPLICABILITY OF CHAPTER 6 LOCA ANALYSES TO SHUTDOWN MODES

5.1 ISSUE

A majority of the analyses of the Loss-of-Coolant Accident and all criteria associated with the accident have focused on scenarios starting from 102% of rated thermal power as prescribed in 10CFR50, Appendix K. In this section, scenarios initiated from other than full power (or 102% in the case of a LOCA) will be addressed to demonstrate that the analyses performed in CESSAR-DC are the bounding cases for all modes of operation. This section will identify the scenarios anticipated for non-Mode 1 operation, and where required the acceptance criteria for the different scenarios.

5.2 ACCEPTANCE CRITERIA

The acceptance criteria for lower operating mode LOCAs is established to be the same as those for higher operating mode LOCAs (Reference 19):

1. Peak clad temperature < 2200°F,
2. < 17% peak local clad oxidation,
3. < 1% core wide oxidation,
4. Maintain coolable geometry, and
5. Maintain long-term cooling.

5.3 DISCUSSION

The types of LOCAs and corresponding break sizes in lower modes are as follows: because of the lower pressures in lower modes, only traditionally "small" break LOCAs are considered. The following lower mode scenarios are considered: most adverse misoperation of valves, the likelihood of cross train maintenance errors, and the possibility of rupture of a tributary pipe. The limiting size and location for a "small" break LOCA for Modes 3 and 4 is a direct vessel injection (DVI) line discharge leg (0.4 square feet). A DVI line break at this location would minimize injection flow into the RCS. Modes 5 and 6 consider a loss of decay heat removal (DHR) resulting from a break in the bottom of the shutdown cooling system suction or a lower head instrument line (.003 square feet).

A discussion of the equipment which would be available to mitigate the consequences of a LOCA in each mode of operation, the effectiveness of that equipment under the conditions for that mode, and the applicability of analyses to that mode are as follows:

Mode 1: In Mode 1, RCS temperature is >350°F and power is >5%. The CESSAR-DC analysis for LOCAs in this mode concludes that all criteria are met.

Mode 2: In Mode 2, RCS temperature is >350°F and power is <5%. The equipment available in Mode 1 is also available in Mode 2. Mode 2 LOCA results are bounded by Mode 1 because the decay heat and stored energy, which are of concern to meeting LOCA criteria, are proportional to the lower power level in Mode 2.

Mode 3: In Mode 3, RCS temperature is >350°F and k_{eff} is <.99. For the case where RCS pressure is >900 psia, the equipment available in Mode 1 (SIT, 4 SI pumps) is available. Because the stored energy and decay heat is lower, this parameter space in Mode 3 is bounded by Mode 1 for LOCA. For the case where pressure is <900 psia, 4 SI pumps are available. Although at a higher temperature than Mode 4, the parameter space in Mode 3 is bounded by the Mode 4 analysis because Mode 4 does not credit automatic SI actuation. Furthermore, despite the fact that SITs would not be available when the pressure is <900 psia, this Mode 3 space is bounded by the Mode 1 analysis because there is less RCS pressure to drive the LOCA leak flow.

Mode 4: RCS temperature is <350°F. Four subspaces are considered:

1. Conditions:
   pressure >900 psia and temperature >317°F. SITs and 2 HP pumps on automatic.

   Conclusions:
   Conclusions will be presented in the June 15, 1992 updated submittal of this report.
2. Conditions:
   Pressure < 900 psia and temperature >317°F: 2 HP pumps on automatic.

   Conclusions:
   Results for these conditions are bounded by the results for Subspace 4 below because 2 HP pumps are available to accommodate the higher temperature.
3. Conditions:
   Pressure > 900 psia and temperature < 317°F.

   Conclusions:
   Results for these conditions are bounded by the results for Subspace 4 below because SITs are available to accommodate the higher pressure.
4. Conditions:
   Pressure < 900 psia and temperature < 317°F.

   Conclusions:
   Conclusions will be presented in the June 15, 1992 updated submittal of this report.

Mode 5: RCS temperature ≤210°F. An analysis was performed which assumed the maximum Technical Specifications cooldown rate of 100°F/hr. Additional information will be presented in the June 15, 1992 updated submittal of this report.

Mode 6: This information will be presented in the June 15, 1992 updated submittal of this report.

Subsection 5.3.1 describes the primary system boundary conditions and event scenario for the postulated LOCA used in this study. Subsection 5.3.1 also compares and contrasts this lower operating mode event scenario to the conservative design basis licensing LOCA event normally associated with ECCS evaluation analyses.

Subsections 5.3.2 and 5.3.3 describe the System 80+ plant parameters and conditions for the analysis and the computer codes and analysis methods used in the LOCA calculations.

Subsection 5.3.4 describes the results of an analysis of a limiting LOCA during Mode 4. These calculations show that hot fuel rod conditions remain in compliance with ECCS Acceptance Criteria during an assumed 10 minute time delay for operator action without SI pump availability. Furthermore, the analysis for the postulated LOCA shows that availability of 1 SI pump at the 10 minute mark maintains the hot rod cladding temperatures in compliance with the ECCS Acceptance Criteria.

5.3.1 DESCRIPTION OF LOCA SCENARIO

Following powered operation of the NSSS, cooldown proceeds at the Technical Specifications maximum rate of 100°F/hour (the maximum cooldown rate shown in Reference 18). Therefore, a primary coolant temperature of 317°F could be reached as early as 2.4 hours after shutdown.

The equipment available to mitigate LOCAs from lower modes is basically safety injection tanks (SIT) and safety injection (SI) pumps. SITs are available for pressures >900 psia. SI pumps are automatically actuated for temperatures >317°F.

If a postulated LOCA transient were to occur at pressures and temperatures slightly below these conditions (i.e., pressure <900 psia, temperature <317°F, and time >2.4 hours), the LOCA would be significantly less dynamic than a design-basis LOCA transient from full power operating conditions (i.e., 2250 psia, ~600°F and full fission power and associated decay heat). Factors which would significantly mitigate the potential and consequences of a lower operating mode LOCA compared to a full power LOCA are (1) lower initial primary system pressure which would limit the internal forces on the piping and the duration of blowdown, (2) lower coolant flow rate out of a postulated break and slower depressurization rate which would reduce inventory loss and flashing rate, and (3) lower decay heat levels which would lessen the core boiloff rate.

Based on these factors, a postulated lower operating mode LOCA followed by, if necessary, timely operator action to initiate safety injection flow would be expected to be much less severe than a LOCA from full power conditions. The most severe lower operating mode LOCA scenario would occur for (1) the largest potential pipe break, (2) after the most rapid possible cooldown from full power, (3) after reducing temperature slightly below which no HPSI pumps are required to be on automatic, (4) after reducing pressure slightly below which SITs are not available, and (5) the longest expected time for mitigating operator action. The largest and most harmful potential pipe break, based on consideration of mechanistic breaks accounting for the system energy at the postulated cooling conditions, would be a significant leak in one of the direct vessel injection (DVI) lines of the reactor coolant system corresponding to the flow area of the DVI line. This DVI break size of 0.4 ft^2 was chosen which envelopes the size and limiting location of all traditional small break LOCAs. Decay heat levels based on a time period of 2.4 hours after shutdown is assumed. No operator action for 10 minutes is also assumed. Forced circulation through the core during lower mode operation would tend to prolong the time of adequate core cooling during a postulated LOCA; therefore, for conservatism, the RCPs are tripped before the start of the event. An aggressive cooldown of the secondary side of the steam generators would considerably benefit the RCS heat removal process for a postulated LOCA; therefore, for conservatism, the steam generator secondary sides are isolated for the event.

5.3.2 SELECTION OF REFERENCE PLANT PARAMETERS AND CONDITIONS FOR MODE 4 ANALYSIS

A limiting set of values is selected from among the plant parameters and conditions. Additional information will be presented in the June 15, 1992 updated submittal of this report.

5.3.3 ANALYSIS COMPUTER CODES

This task required the selection of realistic inputs such as decay heat to provide as much realism in representation of the system transient response during a LOCA as possible. An adaptation of the Realistic Evaluation Model (REM) for small break LOCA was selected for this reason. This model is a second-generation small break LOCA evaluation model intended to replace the 1974 EM currently used for ECCS licensing calculations. Topical reports describing the REM were submitted to the NRC starting in 1988 and are currently under review (see References 12 through 16).

For design basis LOCA calculations from plant initial primary pressures of 2250 psia, the largest break size analyzed using the CEFLASH-4AS code has historically been a 0.5 ft^2 break. For the

0.4 square foot DVI line lower mode LOCA analysis, the REM version of the CEFLASH-4AS code was used with realistic decay heat.

The REM version of the PARCH code was used for calculating hot rod heatup (References 12 through 16). The PARCH base deck included a realistic decay heat.

5.3.4 LOCA ANALYSIS FOR Mode 4

The LOCA analysis examines the hot rod heatup response during a LOCA with the ECCS Acceptance Criteria of peak cladding temperature and peak local cladding oxidation. The objective of this analysis is to determine if the calculated response of the hot rod remains in compliance with the ECCS Acceptance Criteria during the 10 minute time frame before operator actions may be credited.

A bounding analysis, starting at 2.4 hours after shutdown, allowed the LOCA to proceed without safety injection. Primary coolant inventory is assumed to be lost through an opening in the direct vessel injection line at the vessel penetration to the upper annulus. When the two-phase level falls to the top of the active core, there is a reduction in core cooling and the fuel rods begin a heatup driven by the core decay heat power and at higher temperatures by the heat added from cladding oxidation. As cladding temperatures increase, fuel rod swelling and rupture may occur.

For this analysis, the PARCH code is used for calculating hot rod heatup. The hot rod in the core is initialized by the PARCH code with 900 psia and 317°F primary coolant conditions. A limiting axial power shape is assumed for the LOCA. At 8500 seconds (~2.4 hrs), the coolant inventory and, consequently, core cooling is reduced.

Figure 5.2-1 shows the resulting hot rod heatup calculation. A realistic model for decay heat is used, which is the 1979 ANS Standard 5.1 plus two sigma uncertainty plus actinide decay.

5.3.5.1* <u>Results of LOCA Case with No ECCS Delivery for More than 10 Minutes</u>

5.3.5.2* <u>Influence of Restoring 1 HPSI PUMP Not at the Broken DVI Line</u>

5.4* RESOLUTION

* These sections will be provided in the June 15, 1992 updated submittal of this report.

6.0 APPLICABILITY OF CHAPTER 6 CONTAINMENT ANALYSES

6.1 INTRODUCTION

CESSAR-DC Section 6.2.1.1 discusses containment functional design. A series of loss of coolant accidents (LOCAs) and main steam line breaks (MSBLs) were analyzed to determine the resulting containment pressure and temperature for comparison with the containment design pressure and the equipment environmental qualification envelope. The highest containment pressures and temperatures occur when the NSSS stored energy is maximized and the containment heat removal capability is minimized. Maximum RCS stored energy occurs when the plant is at 102 percent power. Maximum steam generator stored energy occurs at 0 percent power (hot standby). Consistent with the Standard Review Plan (SRP), a series of LOCAs were analyzed at 102 percent power and a series of MSLBs were analyzed at powers from 102 percent to 0 percent. The 0 percent MSLB with the failure of a containment spray train is the design basis event (DBE) for System 80+. These cases are presented in CESSAR-DC.

In going from Mode 2 to 6, stored energy is removed from the NSSS. If the safeguards features available in Modes 1 and 2 were available through Mode 6, there would be no question that the DBE identified in CESSAR-DC was limiting. However, some safeguards equipment is removed from service at lower modes. Table 6-1 lists the availability of safeguards equipment credited in the containment analyses as a function of operating mode. Since the Main Steam Isolation Signal (MSIS) and Containment Spray Actuation Signal (CSAS) may be removed from service in Modes 5 and 6, an evaluation of LOCAs and MSLBs in these modes must be made. In addition, a LOCA initiated from zero power (Mode 2) was analyzed. This report discusses the results of these evaluations. The results show that the events presented in CESSAR-DC remain limiting for both containment pressure and equipment environmental qualification. Table 6-[illegible] presents a list of cases that were considered.

6.2 LOSS OF COOLANT ACCIDENTS (LOCAs)

Section 6.2.1.1 of CESSAR-DC presents the results of the containment pressure and temperature analysis for a series of hot leg, suction leg, and discharge leg LOCAs. Consistent with SRP 6.2.1.3, the cases were based on an initial power level of 102 percent. The limiting LOCA in terms of containment peak pressure is the Double Ended Hot Leg Slot (DEHLS) break.

As power level decreases from 102 percent, the hot leg coolant and core temperatures decrease. At the same time, the mass of coolant and pressure on the steam generator secondary sides increases. As a result, primary side stored energy decreases and secondary side energy increases. To show that the cases presented in CESSAR-DC are limiting, a 0 percent power LOCA has been analyzed. Although the case in CESSAR-DC which produced the highest LOCA containment peak pressure was the DEHLS, the Double Ended Suction Leg Slot (DESLS) break with minimum safety injection (most limiting cold leg break) was selected for this analysis because the effect of the increased secondary inventory at no load has more impact on cold leg breaks. Table 6-3 lists the assumptions and initial conditions for this case. This case is actually a Mode 2 case. With the RCS and SG coolant at 556 F, this represents the case with the most NSSS energy of any Mode 2 case. Table 6-4 provides the chronology of events table. The containment peak pressure for this case is 42.0 psig compared to 41.96 psig for the equivalent case at 102 percent power. Results are shown in Figures 6-1 and 6-2.

For Modes 3 and 4, the NSSS stored energy is less than for the case analyzed above. Containment spray is still available so that LOCAs for these modes would produce lower containment pressures and temperatures. For Modes 5 and 6, the Containment Spray Actuation Signal (CSAS) may not be activated. As a result, should a LOCA occur, the containment sprays would not be available; however, since the RCS coolant temperature for Mode 5 is less than 210 F, a LOCA in Modes 5 and 6 would not result in containment pressurization.

Since the LOCAs described in Chapter 6.2.1 of CESSAR-DC are more severe than a LOCA during shutdown modes as far as containment pressurization is concerned, the annulus transient described in Chapter 6.2.1.8 of CESSAR-DC bounds all modes.

6.3 MAIN STEAM LINE BREAKS (MSLBs)

In CESSAR-DC, MSLBs were analyzed at 102, 50, 20, and 0 percent power, representing Modes 1 and 2. The cases were analyzed with either the failure of an MSIV or the loss of a containment spray train. The 0 percent power case with the loss of a containment spray train produced a containment peak pressure of 48.34 psig. This pressure was the highest of any LOCA or MSLB and this case is the containment DBE. The 0 percent cases analyzed in CESSAR-DC are Mode 2 cases. Since they were based on SG pressures of 1100 psia and RCS and SG temperatures of 556 F, they represent the cases with the most stored energy for Mode 2. Mode 2 cases with less stored energy would be less limiting.

In Mode 3 and 4, the NSSS stored energy is less than Modes 1 and 2. As shown in Table 6-1, main steam isolation and containment spray are still available in these modes. Therefore, a MSLB during these modes would not be more limiting than a MSLB during Mode 1 or Mode 2.

In Mode 5 or 6, main steam isolation and containment spray may not be available. On the other hand, the RCS coolant temperature in

Mode 5 will be less than 210 F. If the SG coolant temperature was also at 210 F, no containment pressurization would occur following a MSLB. An analysis has been performed conservatively assuming that the SG coolant was still at 350 F following shutdown cooling of the RCS to 210 F. Table 6-5 lists the assumptions and initial conditions. Table 6-6 lists the chronology of events. The containment peak pressure for this case is 13.4 psig, well below the CESSAR-DC DBE result. Results for this Mode 5 MSLB are shown on Figures 6-3 and 6-4.

6.4 INADVERTENT OPERATION OF CONTAINMENT HEAT REMOVAL SYSTEMS

During shutdown the containment is purged using either the low or high volume containment purge. An inadvertent actuation of the spray system with the containment purge valves open will result in an insignificant decrease in the containment internal pressure. The parameters affecting the negative containment pressure are the containment atmosphere initial conditions and the spray water temperature. The source of the spray water for the System 80+ design is the IRWST. Figure 3.5.4.1 of Chapter 16 of CESSAR-DC which specifies acceptable IRWST temperatures for a range of containment atmosphere temperatures is based on the properties of steam and is not mode dependent. Current Technical Specifications state that Figure 3.5.4.1 applies to Modes 1 through 4. Section 2.2 of the June 15, 1992 updated submittal of this report will expand the applicability of the figure to include Modes 5 and 6.

6.5 CONCLUSION

CESSAR-DC Section 6.2.1 provides containment analyses to support the establishment of the containment design pressure and temperature and an envelope for equipment environmental qualification. A spectrum of primary and secondary line breaks were analyzed. With the exception of the 0 percent power MSLB cases (Mode 2), all of the cases analyzed were for Mode 1. As discussed in the sections above, NSSS stored energy decreases in going from Mode 2 to Mode 6. Safeguards equipment important to containment analyses (containment spray and main steam isolation) are available in all modes with the possible exception of Modes 5 and 6. The analyses above show that by the time the plant is in Modes 5 or 6, the NSSS energy has been reduced to the point where if a postulated primary or secondary line break were to occur, the resulting containment pressure and temperature would be less severe than those presented in CESSAR-DC even with containment spray and main steam isolation unavailable. This result ensures also that the annulus transient presented in CESSAR-DC bounds all Modes.

The inadvertent containment spray actuation event presented in CESSAR-DC Section 6.2.1 is used to determine the maximum external containment design pressure. The analysis presented is not mode dependent.

TABLE 6-1

ESFAS INSTRUMENTATION

SIGNAL	APPLICABLE ModeS
CSAS	1,2,3,4 (Note 1)
MSIS	1,2,3,4

Note 1: Table 3.3.10-1 of CESSAR-DC Chapter 16 does not show CSAS applicable for Mode 4. Section 2.2 of the June 15, 1992 updated submittal of this report will expand the applicability of CSAS to include Mode 4.

TABLE 6-2

CASES ANALYZED

MODE	LOCA	MSLB
1	102% Power DEHLS, DESLS, DEDLS CESSAR-DC	102%, 50%, 20% CESSAR-DC
2	0% DESLS Current Report	0% CESSAR-DC
3	Analysis Not Required Note 1	Analysis Not Required Note 1
4	Analysis Not Required Note 1	Analysis Not Required Note 1
5	Analysis Not Required Since RCS T < 210 F	Case Analyzed With RCS T At 210 F and SG T At 350 F Current Report
6	Analysis Not Required Since RCS T < 135 F	Analysis Not Required Since Less Limiting Than Mode 5

Note 1: Less NSSS stored energy than Modes 1 and 2. Same ESF available as in Modes 1 and 2.

TABLE 6-3

INITIAL CONDITIONS FOR LOCA INITIATED FROM ZERO POWER

Parameter	Value
Reactor Coolant System	
Average Coolant Temperature, F	556
Containment	
Initial conditions are consistent with CESSAR-DC Chapter 6, Table 6.2.1-18.	

TABLE 6-4

ACCIDENT CHRONOLOGY FOR LOCA INITIATED FROM ZERO POWER

Time,sec	Event
0.00	Break occurs
18.00	Start Safety Injection Tank Injection
21.98	Peak Containment Pressure before End of Blowdown
22.00	End of Blowdown
25.90	Downcomer Full
71.00	Containment Spray Injection
71.80	Peak Containment Pressure Subsequent to End of Blowdown
100.00	Safety Injection Tank Empty
108.30	End of Reflood
202.57	End of Post Reflood

TABLE 6-5

INITIAL CONDITIONS FOR MSLB INITIATED FROM MODE 5

Parameter	Value
Reactor Coolant System	
Average Coolant Temperature, F	210
Steam Generator Secondary Pressure, psia	132.8
Steam Generator Secondary Temperature, F	350
Containment	

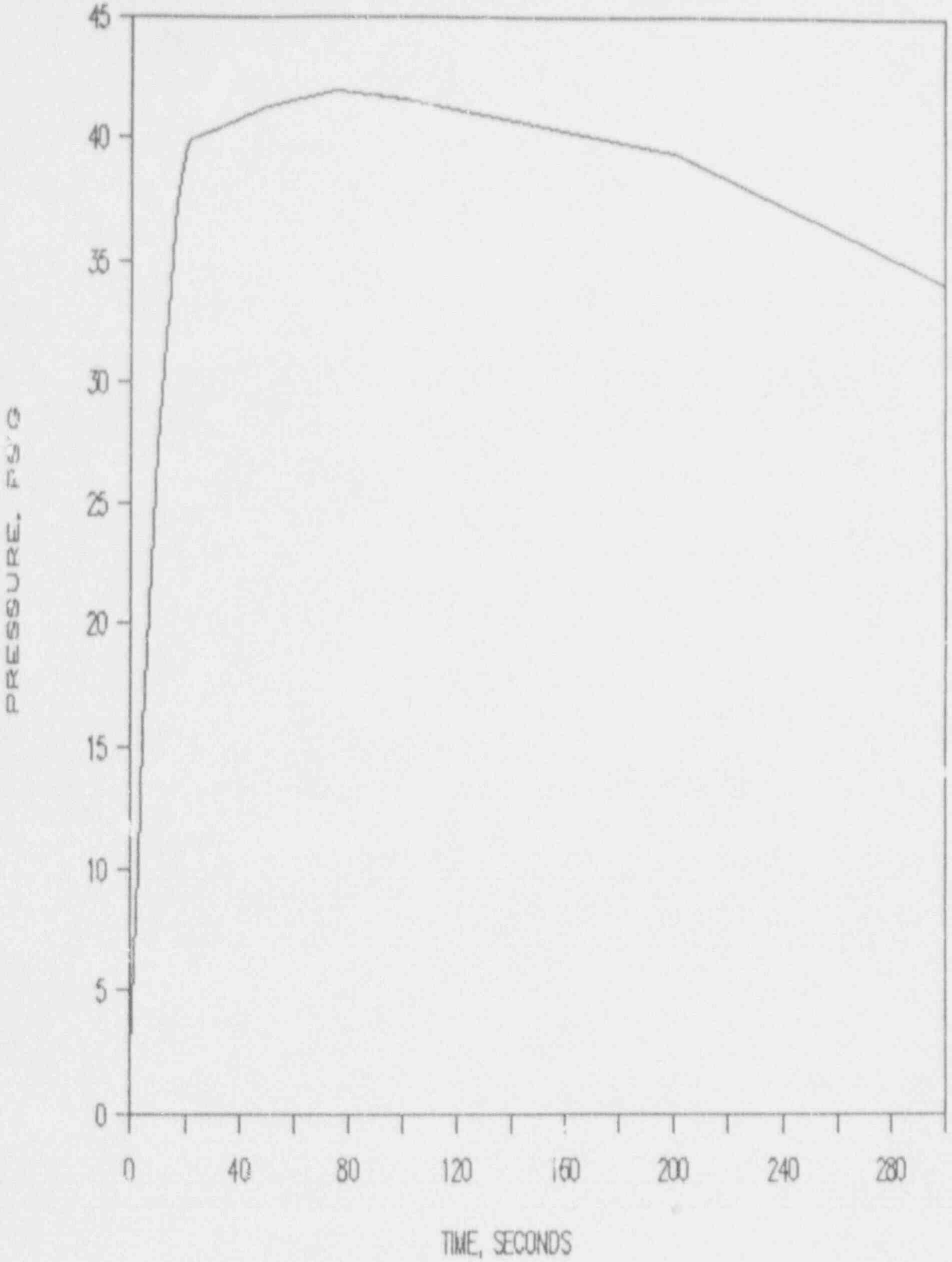
Initial conditions are consistent with CESSAR-DC Chapter 6, Table 6.2.1-18.

TABLE 6-6

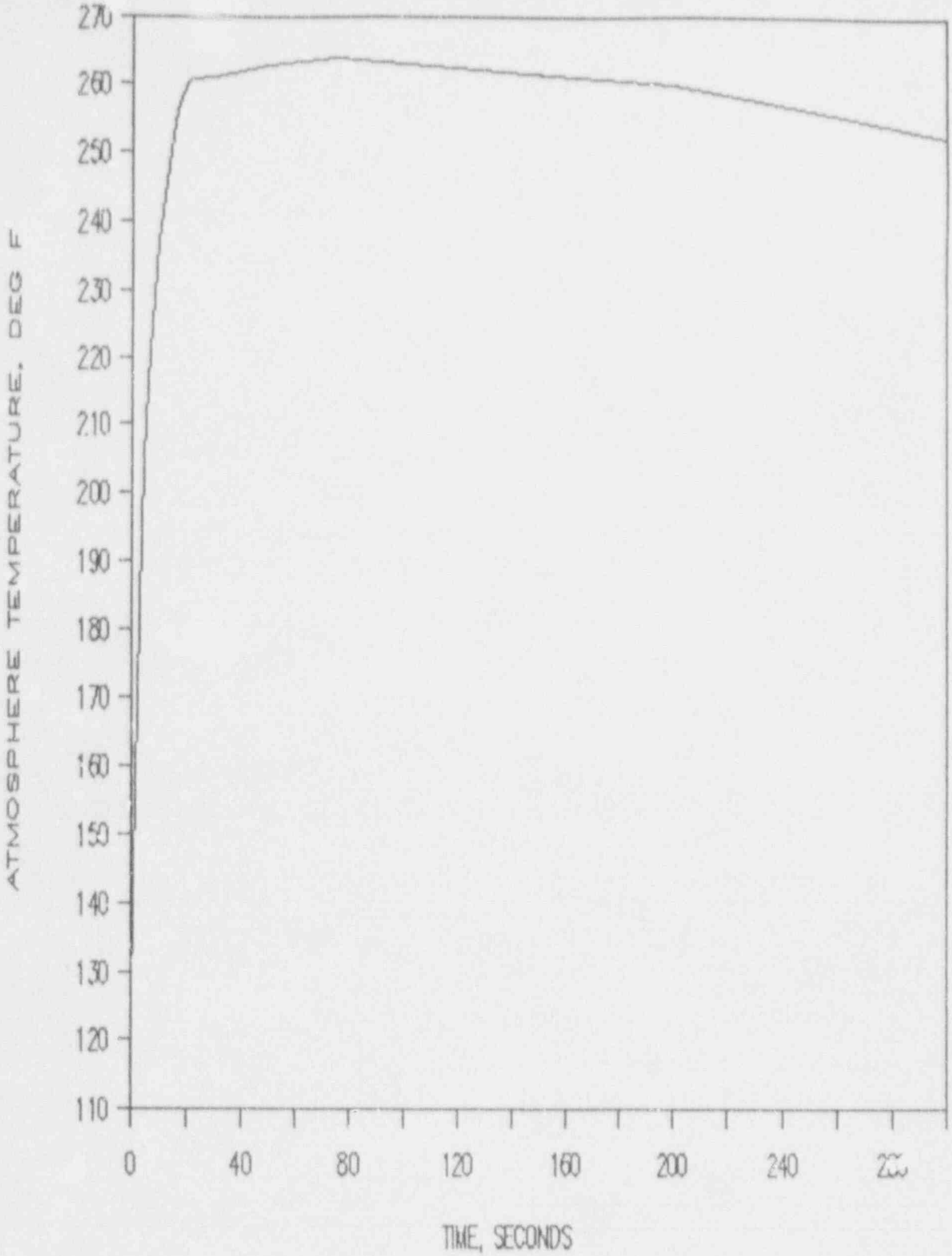
ACCIDENT CHRONOLOGY FOR MSLB INITIATED FROM MODE 5

Time, sec	Event
0.00	Break occurs
148.71	Peak Containment Temperature Peak Containment Pressure

Note: MSIS and Containment Sprays are assumed not to be available in this mode



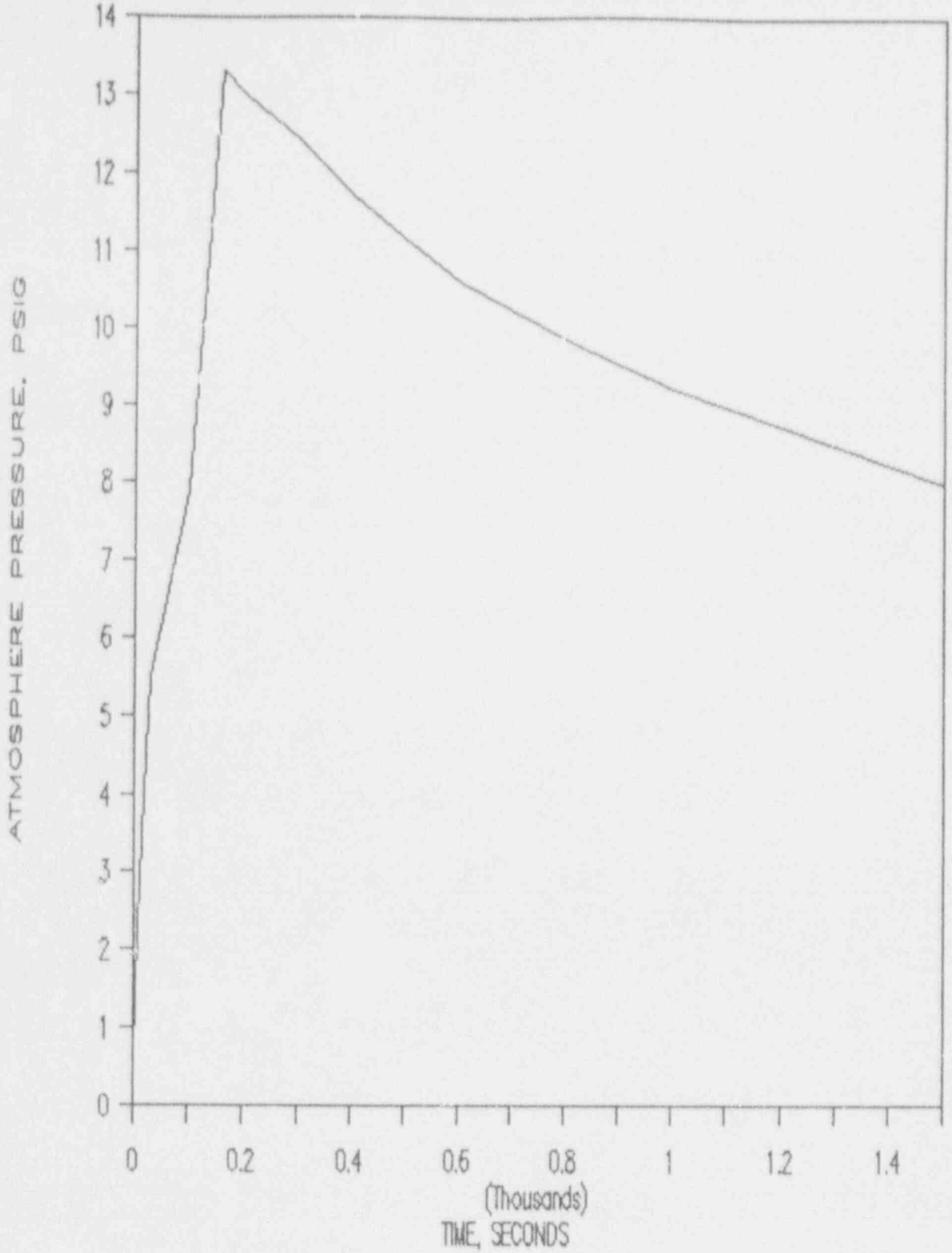
SYSTEM 80+™	CONTAINMENT PRESSURE vs. TIME FOR LOCA FROM ZERO POWER	Figure 6-1



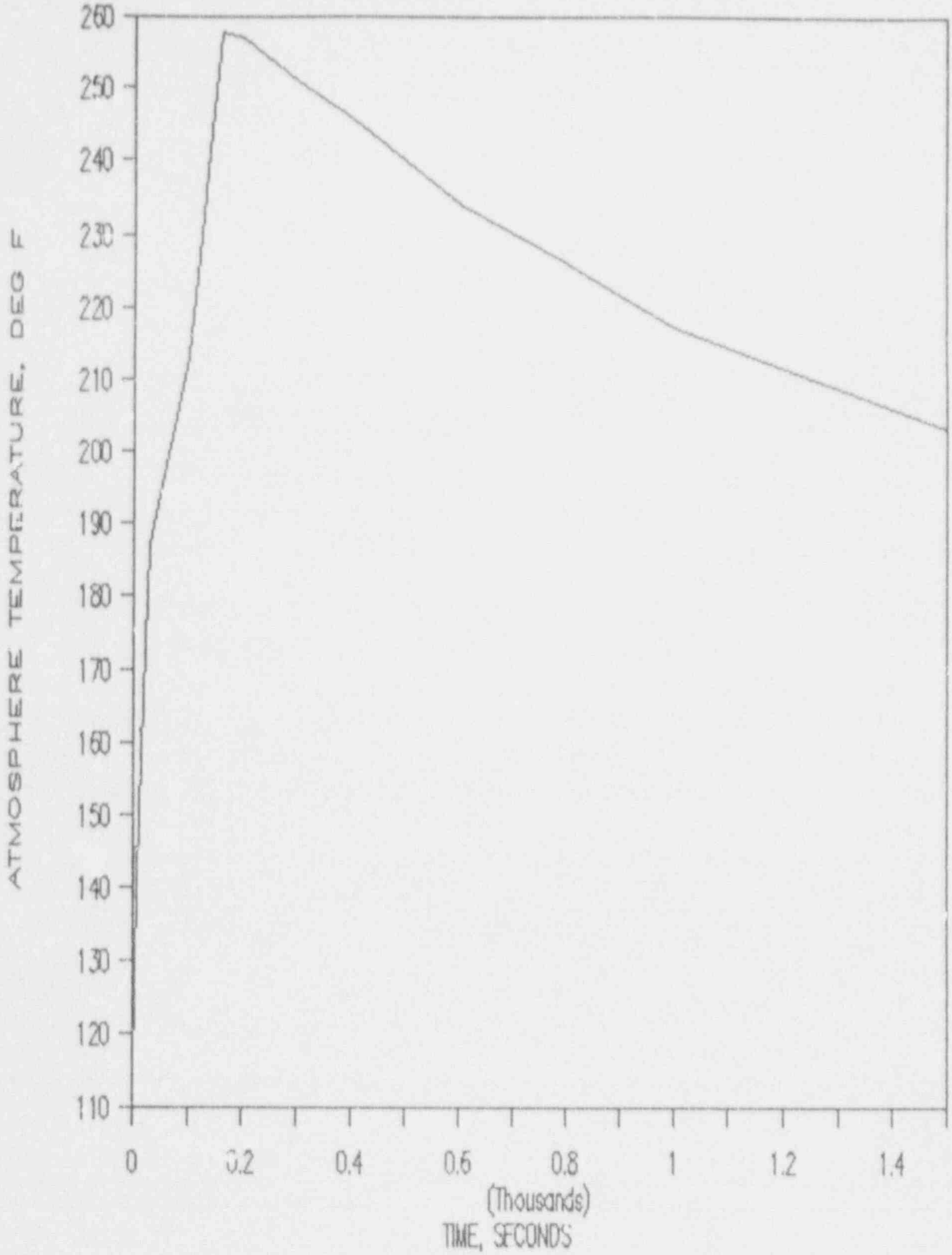
SYSTEM 80+™

CONTAINMENT ATMOSPHERE TEMPERATURE vs. TIME FOR LOCA FROM ZERO POWER

Figure 6-2



SYSTEM 80+™	CONTAINMENT PRESSURE vs. TIME FOR MSLB FROM MODE 5	Figure 6-3



SYSTEM 80+™	CONTAINMENT ATMOSPHERE vs. TIME FOR MSLB FROM MODE 5	Figure 6-4

7.0 SYSTEM 80+ DESIGN FEATURES FOR SIMPLICITY OF SHUTDOWN OPERATIONS

7.1 INTRODUCTION

The System 80+ evolutionary ALWR design takes maximum benefit from prior design and operating experience. It is an objective that this benefit be evident in design features that aid outage planning, reduce operator stress and simplify operator training. Previous sections of this report describe and evaluate features of System 80+ that will improve the overall shutdown operations. In the following discussion, the outage benefits accruing from these design features are presented.

7.2 DISCUSSION

There are several features of the System 80+ design which will aid in the management of an outage. Many of these same features relieve some of the stresses and pressures placed on the licensed operators. These features are presented in Sections 7.2.1 through 7.2.9.

7.2.1 TECHNICAL SPECIFICATIONS FOR REDUCED INVENTORY

Technical Specifications for the System 80+ design specify restrictions on operation in Reduced Inventory. Reduced Inventory is determined by water level in the Reactor Coolant System (RCS). This plant operational condition is defined as "RCS level greater than 3 feet below the reactor vessel flange". These specifications provide guidance to the operations staff and the outage management team. This guidance ensures that equipment assumed to be available for accident mitigation is operable. This aids the outage planners by identifying equipment on which maintenance cannot be accomplished during Reduced Inventory. This planning process reduces the chance for removal of components relied upon for detection and response to accidents. These specifications provide the Senior Licensed Operator a clear standard for determining minimal equipment availability. This will alleviate some of the stress placed on the operator to make the final judgement as to whether equipment is required to be operable for Reduced Inventory operations.

7.2.2 SHUTDOWN COOLING SYSTEM

The Shutdown Cooling System (SCS) design provides two divisions for decay heat removal capability. These two divisions are completely redundant, that is, they share no components or equipment. This redundancy provides the operator with a standby decay heat removal system if any component fails to perform its function. This assurance of a standby system availability reduces some of the pressures and stresses placed on the operator when no standby system is available.

Another feature of the SCS is that it is not a subsystem of the Emergency Core Cooling System. Therefore, the SCS is not required to be operable in Modes 1 through 4. This feature of System 80+ design increases the availability of SCS during Mode 5, Mode, 6 and Reduced Inventory. In addition, this feature aids the outage planning team by allowing maintenance and repair of SCS components and equipment to be accomplished during power operations. This feature eliminates the necessity of finding a window during the outage to allow work on decay heat removal equipment.

Besides the features discussed above, the SCS is designed to provide faster venting of the pumps. If SCS pumps become vapor bound due to misoperation, venting is required. The vent piping for the pumps is hard piped and directed to the floor drain sumps. This allows the plant equipment operator to quickly vent the pump without attaching vent rigs. These vent rigs waste valuable time if recovery from a loss of decay heat removal is required.

7.2.3 CONTAINMENT SPRAY SYSTEM

The Containment Spray System (CSS) design provides two divisions of equipment which can be utilized as an alternate decay heat removal flow path. The CSS pumps are interchangeable with the SCS pumps. This feature provides the operations staff with increased flexibility in the area of forced circulation. Therefore, this alternate alignment for forced coolant flow during shutdown conditions reduces stresses placed on the operators since it increases the redundancy and therefore reliability of the System 80+ decay heat removal capability. With these alternate pumps, the operators have assurances that redundant equipment is available.

7.2.4 COMPONENT COOLING WATER SYSTEM

The Component Cooling Water System (CCW) design has two redundant divisions. Each of the two divisions contains two pumps and two heat exchangers.

This interdivisional redundancy of system components provides flexibility for the management of the maintenance outage. Therefore, major components (e.g., pumps and heat exchangers) requiring maintenance can be removed from service without affecting the availability or reliability of the interdivisional equipment. This enhancement of system design provides the outage planner with options to facilitate easier outage scheduling.

7.2.5 STATION SERVICE WATER SYSTEM

The Station Service Water System (SSW) design has two redundant divisions. Each of the two divisions contains two pumps. This interdivisional redundancy of the pumps provide flexibility for outage planning. The outage planner can schedule a pump for maintenance without affecting the availability or reliability of the interdivisional equipment nor the redundant division equipment.

7.2.6 ELECTRICAL DISTRIBUTION SYSTEM

The Electrical Distribution System (EDS) design has two redundant safety divisions. (Refer to Figure 2.4-1.) Each division is capable of being powered from four separate and diverse sources. These sources include:

1. Switchyard Interface I,
2. Switchyard Interface II,
3. Diesel Generators, and
4. Combustion Turbine.

The EDS provides the outage planner with the flexibility to remove a source of power for maintenance and still maintain other reliable sources to the safety buses. Therefore, required maintenance activities are scheduled without reducing the reliable sources of power to unacceptable levels. These same features provide the licensed operator with alternate sources to which safety buses can be aligned. The operator is aware of these approved alternate alignments through procedures and training. Therefore, the stress placed upon the operator to align to any available source regardless of guidance is reduced. In addition, operator training is facilitated by the procedural guidance.

Another feature of the EDS design is the use of 4 safety buses, 2 per Division. The 1E loads are evenly distributed on the buses to ensure redundancy of system components. For example, each bus powers 1 of 4 Component Cooling Water pumps. This feature provides flexibility for the outage planner. One bus can be removed from service for maintenance and redundant components still have a power supply available.

7.2.7 NUPLEX 80+ ADVANCED CONTROL COMPLEX

The NUPLEX 80+ Advanced Control Complex (ACC) is an integral part of the System 80+ design. The design goals of NUPLEX 80+ include the integration of NSSS and balance of plant systems into a unified control complex, reduction of human errors that affect plant safety and improving the reliability of the man-machine interface through redundancy, segmentation and diversity.

Control room information provided by NUPLEX 80+ is consistent with operator information requirements when performing operational tasks during plant evolutions or responding to unexpected conditions. The operator can obtain information from a number of sources in the NUPLEX 80+ ACC. These sources include:

- A large plant overview status board known as the Integrated Process Status Overview (IPSO).

- Alarm tiles and associated message windows.

- Discrete indicators provide frequently used and important information.

- CRT displays containing essentially all power plant information.

More detailed information on the NUPLEX 80+ ACC can be found in Chapter 18 of System 80+ CESSAR-DC.

The NUPLEX 80+ ACC utilizes the same parameter conventions for the indicators and alarms for shutdown operations as required for power operations. The features of each which simplify operator training, aid in outage planning and reduce operator stress are described below.

- IPSO - IPSO provides the operators, especially Senior Reactor Operators with an overview of plant status during shutdown conditions. This overview allows the operators to view system status during outage activities. Having an overview of the plant will reduce uncertainty of the availability of required safety systems. This knowledge of the availability reduces the stress placed on the operators by uncertainties.

- Alarms - Mode and equipment dependent alarms are a special feature of NUPLEX 80+ ACC. This feature eliminates the alarming of alarms not applicable to the current mode, operating conditions, or equipment status. A large amount of maintenance activities involved with the outage affect control room alarms. The mode and equipment dependent alarms eliminate operator response to nuisance alarms caused by authorized work being performed. Outage work may be planned, alarms disabled and unnecessary investigation by operators into these alarms eliminated.

- Discrete Indicators - Discrete Indicators provide several simplifying attributes for operators. Automatic ranging scales on Discrete Indicators allow accurate indication over the entire range of system design with the same indicator. Using the same indicator for all conditions of operation, including shutdown, avoids confusion for the operators. This feature allows training to utilize the same indicators on a simulator. It also eliminates the utilization of indicators solely for shutdown. Discrete Indicators receive multiple channel input signals to be displayed on one indicator. These signals are validated and provided the operator with reliable indication even if some channels are removed from service. Individual channel inputs to the Discrete Indicators alarm to alert the operator when one has been removed from service. This allows the operator to check the status of information provided. Using validated displays reduces stress on the operator by eliminating doubt of instrument availability and accuracy.

- CRT Displays - CRT displays for the NUPLEX 80+ ACC are arranged in a structured information hierarchy. This structure provides the operator information consistent with operational needs. Levels of display information start with IPSO and continue through detailed plant information. A feature of the CRT display is graphic representation of systems. This reinforces system layout training and leads to better understanding by the operator. It provides consistency for the operators which reduces stress. Color representation of valves to indicate operable/inoperable status gives the operator the information to determine flowpath status. This feature also provides status of maintenance in progress on important valves in the plant.

7.2.8 REDUCED INVENTORY INSTRUMENTATION

The System 80+ design provides the instrumentation to insure the Control Room Operator (CRO) is informed of the decay heat removal system performance and the reactor coolant system level and temperature. The instrumentation includes:

1. Reactor Coolant System Level,
   - Heated Junction Thermocouple
   - dP
2. Reactor Coolant Temperature,
   - Core Exit Thermocouple
   - Heated Junction Thermocouple
3. Shutdown Cooling System Flow,
4. Shutdown Cooling System Pressure,
5. Shutdown Cooling System Temperature, and
6. Shutdown Cooling Pump Motor Current.

See Section 2.8, Instrumentation, of this report for a discussion of this instrumentation. The instrumentation is coupled with the Nuplex 80+ Advanced Control Complex (ACC) to provide the CRO with indications and alarms to monitor reduced inventory operations. (See Section 7.2.7 above for a description of Nuplex 80+ ACC features).

8.0 CONCLUSIONS

This section will be provided with the July 31, 1992 updated submittal of this report.

9.0 REFERENCES

1. Letter LD-92-038, C. B. Brinkman (ABB) to D. M. Crutchfield (NRC) dated March 25, 1992.

2. Memo Letter, "Summary of Meeting Held on December 18, 1991 Regarding Shutdown Risk", T. V. Wambach (NRC), dated January 30, 1992.

3. NUREG-1449, "Shutdown and Low-Power Operation at Commercial Nuclear Power Plants in the United States", Draft Report dated February, 1992.

4. USNRC Generic Letter No. 88-17, "Loss of Decay Heat Removal", dated October 17, 1988.

5. USNRC AEOD Special Report, "Review of Operating Events Occurring During Hot and Cold Shutdown and Refueling", dated December 4, 1990.

6. NUREG-0800, USNRC Standard Review Plan, Revision 1, July, 1981.

7. Jaquith, R.E., et.al., "Probabilistic Risk Assessment for the System 80+ Standard Design", Combustion Engineering Inc., DCTR-RS-02, January, 1991.

8. Letter from James H. Wilson (NRC) to E. E. Kintner (EPRI) dated September 5, 1991.

9. Brockhold, G. (EPRI), Memo to ALWR Utility Steering Committee, ALWR-92-18, January 15, 1992.

10. (Later)

11. ANSI/ANS-58.8-1984.

12. CEN-373-P, Volume 1, "Realistic Small Break LOCA Evaluation Model, Calculational Models," April 1988.

    Volume 2, "Application of Evaluation Model," December 1988.

    Volume 2 Supplement 1-P, "Application of Evaluation Model to Calvert Cliffs Units 1&2, September 1989.

    Volume 3, "Computer Program Input and Output Description," December 1988.

13. Letter LD-88-030, "Submittal of Realistic SBLOCA Evaluation Model," A. E. Scherer (ABB CE) to J. A. Norberg (NRC), April 27, 1988.

14. Letter LD-88-155, "Submittal of Volumes 2 and 3 of Combustion Engineering's Realistic Small Break Loss-Of-Coolant-Accident Evaluation Model," A. E. Scherer (ABB CE) to J. A. Norberg (NRC), December 9, 1988.

15. Letter LD-89-001 "Addendum to Volume 3 of Combustion Engineering's Realistic Small Break Loss-of-Coolant Accident Evaluation Model," A. E. Scherer (ABB CE) to J. A. Norberg (NRC), January 11, 1989.

16. Letter LD-89-099, "Supplement 1 to Volume 2 of Combustion Engineering's Realistic Small Break LOCA Evaluation Model," A. E. Scherer (ABB CE) to J. A. Norberg (NRC), August 28, 1989.

17. ANSI-N18.2-1973.

18. "System 80+, CESSAR-DC," Technical Specification, 3.4.3.

19. 10CFR50.46.

20. CENPD-138, Supplement 2-P, "PARCH, A FORTRAN-IV Digital Program to Evaluate Pool Boiling, Axial Rod and Coolant Heatup," January, 1977.

APPENDIX A

RESPONSES TO

REQUESTS FOR ADDITIONAL INFORMATION

APPENDIX A

TABLE OF CONTENTS

APPENDIX A (Continued)

TABLE OF CONTENTS

A1.0 INTRODUCTION

Letters from the NRC that transmitted Requests for Additional Information (RAIs) on shutdown related topics are listed in Section 3.0 of this Appendix as References A-1 through A-6. ABB provided individual responses to some of these RAIs in References A-7 through A-13. These responses are reproduced in this Appendix. The remaining RAIs related to shutdown risk were listed in the ABB letter, Reference A-14 along with a commitment to provide responses via this report. These responses are also given in this Appendix, where the response may refer to the content of applicable sections of this report. The combination of the specific responses provided in the Appendix along with the overall report content fulfills all outstanding commitments to provide shutdown risk information in support of the CESSAR-DC review process.

A2.0 RAIs

The following pages provide the usual format for question and response to the RAIs. Those not included at this time will be submitted in the June 15, 1992 or the July 31, 1992 updated submittals scheduled for this report.

Question 440.16(j):

Safety analysis reports (SARs) typically concentrate on power operation when consideration is given to many of the potential operational transients. The recent experience from the events in operating reactors indicated that further evaluation for plant operation at lower modes may be required. Hence, it may be prudent to address non-power operation in more depth than has been traditional. What plans exist, if any, with respect to this topic and the System 80+ program?

Response 440.16(j):

The System 80+ program encompassed an extensive evaluation to assess the vulnerability of the System 80+ design to various transients. The results of this evaluation are provided in the System 80+ Shutdown Risk Evaluation Report. Sections 2.4, 2.5, 2.6, 4.0, 5.0, and 6.0 of this report present transient analyses covering the extent of events of concern for shutdown operations. Sections 2.4 through 2.6 concentrate on the loss of decay heat removal and rapid boron dilution events. Section 4.0 represents the evaluation of the impact of CESSAR-DC Chapter 15 events occurring during shutdown modes of operation. Section 5.0 represents the evaluation of the impact of the CESSAR-DC Section 6.3 loss of coolant accidents occurring during shutdown modes of operation. Section 6.0 evaluates the impact of the CESSAR-DC Section 6.2 events on the containment response when these events are initiated from shutdown modes of operation.

In addition, to supplement the above evaluations Section 3.0 of the System 80+ Shutdown Risk Evaluation Report presents a Probabilistic Risk Assessment which covers various event sequences which can occur during shutdown operations.

This response and the mentioned report sections fulfill the commitment in Reference A-14 relevant to this RAI.

Question 440.49

Provide a discussion of the procedures and plant systems used to take the plant from normal operating conditions to cold shutdown conditions. This discussion should include, heat removal, depressurization, flow circulation, and reactivity control.

*Response 440.49

The principal systems utilized in taking the plant from Mode 1, Power Operation, to Mode 5, Cold Shutdown are:

Reactor Coolant System
Feedwater System
Feedwater Control System
Reactivity Control System
Boron Control System
Chemical & Volume Control System
Shutdown Cooling System
Pressurizer Level Control System
Steam Bypass Control System
Pressurizer Pressure Control System
Liquid and Gaseous Waste Management Systems
Main Steam System
Condensate System

Reactivity control capability is discussed in Sections 7.7.1.1.1; 7.7.1.1.7; 9.3.4.1.3.3; 9.3.4.2.1 (last paragraph), and 9.3.4.2.3C. Power is reduced by increasing the boron concentration in the RCS to reduce k-effective to ≤ 0.99. At low power the rods are inserted. The operator borates to the cold shutdown boron concentration consistent with the Technical Specifications prior to the beginning of cooldown. This margin is maintained throughout cooldown by making up shrinkage volume by means of the CVCS with water at the cold shutdown margin boron concentration.

Cooldown is effected by the systems described, and techniques discussed in Sections 5.4.7.2.6.A; 9.3.4.2.3C; 7.7.1.1.2.1; 7.7.1.1.2.2; 7.7.1.1.4; 7.7.1.1.5 and 10.4.7.2.4. Additionally, the following precautions, limits and techniques are utilized during cooldown:

1. The reactor coolant pumps continue to run until they are manually tripped

   o Four RCPs shall not be operated below approximately 500°F

   o During cooldown RCP 1A and 1B shall be running to maintain pressurizer spray capability until they are required to be shutdown for some other reason

- The RCPs shall not be operated when the system pressure is below cavitation or seal operation limits

- The RCS pressure is maintained at 2250 psia until cooldown is initiated.

3. Pressurizer pressure and level controls are placed in manual mode at the beginning of cooldown, and power to heaters is reduced.

4. flow is maintained throughout cooldown by RCPs and/or shutdown cooling system pumps.

5. The bubble in the pressurizer is maintained as long as possible.

6. Volume Control Tank (VCT) gas space is vented to reduce fission gas and hydrogen gas prior to cooldown.

7. Letdown flow is directed, as required, to the gas stripper to remove dissolved gas.

8. Initially, heat is removed from RCS by dumping steam:

   o Steam may be dumped to the condensers through the Steam Bypass System, or to the atmosphere through the Atmospheric Dump Valves (ADV).

   o Feed control is in manual during cooldown using the startup pump and manual control valves.

   o The MSIS setpoint is adjusted to 200 psi below existing steam pressure as cooldown progresses.

9. As RCS water cools, pressure is decreased by manually adjusting pressurizer spray to cool the vapor space. Pressurizer pressure is controlled such that saturation margin limit is not exceeded, and such as to comply with the pressure-temperature curves specified for the plant.

10. As pressurizer pressure decreases the SIAS & CIAS setpoints are decreased to 400 psi below existing pressurizer pressure.

11. RCS cooldown rate shall be maintained within Technical Specifications (TS) at all times during cooldown.

12. Pressurizer water temperature should exceed RCS water temperature by no more than 350°F and no less than 50°F whenever there is a bubble in the pressurizer.

Response 440.49 (continued)

13. Auxiliary pressurizer spray is utilized to reduce pressurizer pressure whenever normal spray is inadequate or not available.

14. When the pressurizer pressure is approximately 400 psia and the RCS temperature decreases to 350°F, cooldown is transferred to the Shutdown Cooling System (SCS). Cooldown from this point is fully described in Section 5.4.7.2.6A. Steaming and feed may be terminated.

*This response was previously transmitted by Reference A-12

Question 440.70

Describe the means provided for ECCS pump protection including instrumentation and alarms available to indicate degradation of ECCS pump performance. The staff's position is that suitable means should be provided to alert the operator promptly to possible degradation of ECCS pump performance. All instrumentation associated with monitoring the ECCS pump performance should be operable without offsite power, and should be able to detect conditions of low discharge flow.

*Response 440.70

The following instrumentation is used to determine Safety Injection and Shutdown Cooling pump performance. This instrumentation is shown in Figures 6.3.2-1A, B, and C and described in Section 6.3.5.3 of CESSAR-DC.

Channel	Function	Control Room Features
P-302, 305	SCS Pump Discharge Pressure	Indication
P-306, 307, 308, 309	SI Pump Discharge Pressure	Indication
P-319, 329, 339, 349	SI Line Pressure	Indication Alarm (High)
P-390, 391	SI Hot Leg Injection Pressure	Indication Alarm (High)
F-302, 305	SCS Line Flow	Indication Alarm (Low)
F-306, 307, 308, 309	SI Pump Discharge Flow	Indication Alarm (Low)
F-311, 321, 331, 341	DVI Nozzle Injection Line Flow	Indication
F-390, 391	SI Hot Leg Injection Flow	Indication

The normal power supplies for the above instrumentation are the 120 VAC vital I&C buses, which are powered from either the Class 1E 480 VAC buses or the 125 VDC buses through inverters (See CESSAR-DC Figure 8.3.2-2). If offsite power is lost, the Class 1E 480 VAC buses may be powered via the 4.16 KV safety buses by the emergency diesel generators (See CESSAR-DC Figure 8.3.1-1) or by the alternate AC power source (gas turbine).

Testing to confirm that SI and SCS pump performance is within specification is included in the Safety Injection System Test sections of Preoperational Tests, Section 14.2.12.1 of CESSAR-DC. In addition, Technical Specification 3.5.2 (CESSAR-DC Section 16.3.2) provides requirements for testing safety injection flowrates.

*This response was previously transmitted by Reference A-12

Question 440.91:

Most of the Chapter 15 and 6.3.3 (LOCA) analyses are performed based on the event being initiated at full power operation. The staff requires that C-E provide an assessment on the consequences of the transients and accidents initiated at low power levels or lower modes of plant operation such as shutdown operations. This is required to demonstrate that the analyses performed in CESSAR-DC are the bounding cases for all modes of plant operation.

Response 440.91:

This assessment is provided in the System 80+ Shutdown Risk Evaluation Report. Section 4.0 of the report contains the assessment of CESSAR-DC Chap r 15 events and Section 5.0 contains the assessment of CESSAR-DC Section 6.3.3 events. Also see the response to RAI 440.16(j).

The response and the mentioned report sections fulfill the commitment in Reference A-14 relevant to this RAI.

Question 440.109 (15.6.3)

Provide the results of an analysis for the potential boron dilution event during the recovering phase following a SGTR when backfill from the secondary system through the ruptured steam generator occurred.

*Response 440.109

The System 80+ Emergency Procedure Guides will include steps to prevent backfill from the secondary system through the ruptured steam generator by maintaining a positive pressure difference between the primary and secondary systems. (See Section 2.1 of the System 80+ Shutdown Risk Evaluation Report.) Therefore, boron dilution should not occur and has not been analyzed. A further note is made that backfill is not necessary to prevent overfilling of the larger System 80+ steam generator as the result of a SGTR event.

*This response was previously transmitted by Reference A-11. Changes have been made as noted by the bars in the margin.

Question 440.139

As mentioned in RAI 440.109, discuss the potential for boron dilution during the recovering phase following a SGTR when backfill from the secondary system through the ruptured S/G occurs. This analysis should also be provided in support of GSI-22, CESSAR-DC Section 15.4.6, etc. ...

*Response 440.139

The issue of a "potential" boron dilution resulting from a SGTR accident was addressed in the response to RAI 440.109.

*This response was previously transmitted by Reference A-11

Question 440.148:

Will there by any maintenance activities for the System 80+ that will require isolation of IRWST pump suction inlets (or allow foreign material in the sump with potential for blockage)? If so, this would preclude operation of safety systems. What guidance can be provided to minimize this potential risk? Have TSs been provided limiting such maintenance activities?

Response 440.148:*

No maintenance activities that will require isolation of the IRWST pump suction inlets are possible because the inlets will be submerged during all modes of operation. Maintenance in the IRWST is only possible during mode 6, when IRWST inventory has been transferred to the refueling pool. During refueling operations, the Shutdown Cooling System pumps utilize the IRWST ECCS suction connections to fill the refueling cavity. Due to NPSH and vortexing considerations, the suction inlets are sufficiently submerged to protect the SCS pumps while the pumps are in opera

While maintenance is being performed in the IRWST, the possibility exists for foreign material to accumulate in the tank. The System 80+ design includes provisions to prevent this debris from entering or blocking the ECCS suction lines. The suction lines are isolated from areas of high maintenance (i.e., away from the IRWST spargers) to decrease the possibility that debris will reach the inlets. Large vertical screens capable of filtering particles greater than 0.09 inches diameter are located within the IRWST to effectively block debris. Should maintenance be required in areas where maintenance-generated trash would not be filtered by the debris screens, a mesh "cage" that completely surrounds the suction inlets would prevent debris from entering the lines. A complete discussion of this issue is provided in section 2.9.3 of the Shutdown Risk Evaluation Program Report.

Using procedural guidance provided by the plant designer in Section 2.1 of the final submittal of the System 80+ Shutdown Risk Evaluation Report, the owner-operator will develop plant specific procedures that require that maintenance-generated trash be removed from the IRWST before refilling the tank.

* This reponse supplements this report and together they fulfill the commitment to respond in Reference A-14.

Question 410.54:

The safety evaluation of both the new and spent fuel storage areas includes an evaluation of the effects of dropping a fuel assembly and its handling tool from a height of two feet above the storage rack. Provide the following additional information in accordance with SRP 9.1.2, Item III.2.e guidance: Verify that the drop of any allowed lighter loads at a greater height does not result in a higher potential energy than a fuel assembly and its handling tool dropped from its normal operating elevation. Perform an evaluation of this in accordance with SRP 9.1.4 guidance.

*Response 410.54:

The spent fuel racks have been evaluated and the results show that the rack k_{eff} will be less than .95 under the following postulated accident conditions:

(1) Drop of a fuel assembly handling tool from its maximum lift height over the fuel racks.

(2) Drop of a fuel assembly and the handling tool from their maximum lift height over the fuel racks.

(3) Drop of other items, such as a failed fuel canister with a fuel assembly, from their maximum lift height over the fuel racks.

*This response was previously transmitted by Reference A-13

Question 410.54

The response to RAI 410.54 is incomplete. Please provide the values for the maximum lifting height assumed for each case analyzed.

*Response 410.54

Lift heights and grapple weights are not known prior to final procurement of the refueling machine used in the fuel building. System 80+ spent fuel racks can absorb an impact energy of 93,100 inch-lbs without exceeding the rack K_{eff} criteria. The refueling equipment will be designed so that the maximum impact energy resulting from a dropped fuel assembly and handling tool, a dropped handling tool or any other dropped fuel handling related load from their maximum respective lift heights will not exceed the energy absorbing capacity of the fuel rack while maintaining K_{eff} criteria. In addition, the owner-operator, using procedural guidance provided by the plant designer in Section 2.1 of the final submittal of the System 80+ Shutdown Risk Evaluation Report, will develop administrative controls to limit the size and lift height of any other non-fuel handling loads that are carried over the fuel racks such that this maximum impact energy is not exceeded.

*This response was previously transmitted by Reference A-13. Changes have been made as noted by the bars in the margin.

Question 410.64

You have identified the different storage densities for regions I and II of the spent fuel pool (50% and 7% , respectively) in your submittal. Provide pertinent information concerning the design criteria and anticipated controls to be implemented for the storage of spent fuel assemblies in the above regions.

* Response 410.64

Both Region I and II storage areas are designed to accommodate fuel assemblies with initial enrichment up to 5 weight percent U-235. Region I has no restriction on burnup history of stored fuel assemblies. Region II is restricted for storage of fuel having a minimum cumulative burnup which is dependent on the initial enrichment for each fuel assembly. The burnup versus enrichment curve is internally documented. This restriction on fuel storage in Region II will be imposed by administrative controls developed and implemented by the Owner-Operator.

The following will be added in Section 9.1.2.2.2: "A fuel assembly may be stored in Region II only if it has the minimum burnup required for an assembly of its initial enrichment. The Owner-Operator will develop and implement administrative controls to permit storing a fuel assembly in Region II only if it meets established burnup versus initial enrichment requirements."

*This response was previously transmitted by Reference A-8

Question 410.65

You have stated in Section 9.1.2.3.1.3 that one of the accidents considered in the design of the spent fuel pool storage racks is a fuel assembly and its handling tool "falling into a blocked-off fuel storage cavity." Supply additional information concerning the mechanical blocking assemblies to allow determination of the extent of penetration of a fuel assembly into a blocked cavity.

*Response 410.65

The spent fuel racks provide storage for 363 fuel assemblies in Region I (50% density) and 544 fuel assemblies in Region II (75% density). The restricted rack cells contain cell blockers which prevent the placement of a fuel assembly into the restricted cells. The racks have been analyzed based on a postulated accident condition of a fuel assembly fully inserted into a restricted cell. Taking pool boron concentrations into consideration, the results show that the rack k_{eff} is less than .95.

The cell blockers cannot be inadvertently removed once installed as special tooling is required to unlock and remove them from the spent fuel storage racks.

*This response was previously transmitted by Reference A-8

Question 410.66

Your submittal does not provide information concerning the handling of heavy loads in the vicinity of the spent fuel pool. Provide an evaluation of the capability of the spent fuel loading pit to withstand a dropped heavy load. The evaluation should include a shipping cask drop without breach of the pit area or loss of spent fuel pool water.

*Response 410.66

The spent fuel cask laydown area is separated from the spent fuel pool by a gate and a structurally reinforced concrete wall. The gate is closed, sealed, and locked during all cask handling operations. The floor in the laydown area has been designed to withstand the impact of a shipping cask dropped from a height of 30 feet without breaching the integrity of the floor plate.

Any small water loss as a result of local damage to the laydown area wall liner cannot be communicated to the spent fuel pool due to the closed gate and the integrity of the independent spent fuel pool liner. Damage to the gate is prevented during cask handling by stops on the bridge crane rail that limit cask travel and by the recessed gate design.

Design features to address the spent fuel cask drop accident are summarized in CESSAR-DC, Section 15.7.5, Amendment H.

*This response was previously transmitted by Reference A-8

Question 410.72

Provide the fuel building layout drawings which show the (1) overhead heavy load paths and (2) safety-related equipment locations in the vicinity of those paths susceptible to damage by failure of electrical interlocks, swinging of the load, or other mechanisms for causing damage.

* Response 410.72

The containment polar crane, the cask handling crane, and the fuel handling crane are designed to prevent the drop of a heavy load such as the reactor vessel head and the spent fuel shipping cask. In addition, predetermined load paths for major lifts (see Figures 9.1-19 and 9.1-20), operator training, and regular crane maintenance minimize the possibility of load mishandling.

Limit switches, electrical interlocks and mechanical interlocks prevent improper crane operation which might result in a fuel handling accident. This is also discussed in Section 9.1.4.2.1.7. The spent fuel cask handling hoist is restricted from movement over the new and spent fuel storage areas when the fuel racks contain fuel assemblies. The new fuel handling hoist is restricted from movement over the spent fuel storage area when the spent fuel racks contain fuel assemblies.

In accordance with the regulatory position of Regulatory Guide 1.13 and General Design criteria 61 of Appendix A to 10 CFR 50, the hoists are also restricted from passing over the spent fuel pool cooling system or ESF systems which could be damaged by dropping the load.

Set points for the hoist interlocks are set to preclude falling or tipping of the loads into the fuel storage areas.

Typically, administrative controls prepared by the Owner-Operator preclude movement of heavy loads within the containment building pool when the refueling machine contains a fuel assembly. During heavy load movement, the fuel transfer tube valve is closed to avoid water level changes in the fuel building during postulated accident conditions such as dropping the heavy load on the reactor vessel pool seal.

THe first sentence of the last paragraph of Section 9.1.4.3.1 has been modified to state: "Administrative controls prepared by the Owner-Operator..."

*This response was previously transmitted by Reference A-8

Question 410.73

Provide containment layout drawings showing the reactor vessel head storage location, the upper guide structure storage stand, the load paths from the reactor to those locations, and safety-related equipment in the vicinity of the load paths susceptible to damage by load handling accidents.

* Response 410.73

Figure 9.1-19 depicts the load paths of the reactor vessel closure head, the core support barrel (CSB), and the upper guide structure (UGS) from the reactor vessel to their respective storage areas during the refueling outage. The designated load path for each component passes over the reactor vessel flange. An analysis has shown that in the event the reactor vessel head or the internals are dropped on the reactor vessel, the core will be maintained in a coolable condition.

Typically, operating procedures prepared by the Owner-Operator control the lift height of the UGS to minimize its clearance with the pool floor and the polar crane is positioned to insure direct travel from the reactor vessel to the UGS storage area. Additionally, the ICI holding frame is installed over the seal table at the operating floor level during fuel handling. This prevents the UGS from moving over the seal table. Therefore, operating procedures preclude the UGS from being lifted above the seal table or being closer than approximately five feet to the seal table, thereby making seal table damage a remote possibility. However, under a postulated load drop on the seal table, the seal table would fail resulting in containment pool draindown to the reactor vessel flange area. Since ICI tubes are only restrained laterally and not vertically, the tubes would be bent down to the level of the reactor vessel flange. Any tube failure, therefore, would in all likelihood be at or near the reactor vessel flange level which would result in a water level within the reactor vessel similar to that prior to reactor vessel head removal. The accident condition would not be any more severe than that analyzed for the reactor vessel head drop on the reactor vessel flange.

There are no other unprotected safety-related components within the load paths of the reactor vessel head, UGS and CSB. An unprotected component is defined as a component that is not protected by the pool walls and/or operating floor.

The transfer tube valve will be closed during these handling evolutions to preclude water level changes in the fuel building.

The following sentence will be added to the end of the first paragraph of Section 9.1.4.3.1: "The Owner-Operator's operating procedures will control the load paths and height of the reactor vessel closure head, the core support barrel and the upper guide structure above the pool floor.

*This response was previously transmitted by Reference A-8

Question 410.103

a. Section 9.1.1.1 states compliance with the "intent" of Regulatory Guide 1.13 as a design basis. Considering that Regulatory Guide 1.13 pertains to spent fuel storage, explain what parts of the Guide, and to what extent, are met by the new fuel storage design.

b. Section 9.1.1.3.3 states that "new fuel storage racks and facilities are qualified as Seismic Category I." Identify the "facilities" which are so qualified.

c. Section 9.1.1.2 does not provide sufficient descriptive information on features illustrated in the figures. For instance, what is the function of "L" insert slots and boxes? How are the "cell blockers" attached to the structure? What is the equipment in the "new fuel inspection area"? What is their seismic classification?

d. The new fuel storage capacity changed from 166 in Amendment E to 121 in Amendment I. What is the design basis for the storage capacity of the system?

e. According to SRP Section 9.1.1, the design of the new fuel storage facility is acceptable if the integrated design is in accordance with, among other criteria, General Design Criteria 61 and 62 of 10 CFR 50, Appendix A. Specific criteria necessary to meet the requirements of GDC 61 and 62 are ANS 57.1 and ANS 57.3 as they relate to the prevention of criticality and to the aspects of radiological design. Provide information on the extent of compliance of the design to ANS 57.1 and ANS 57.3.

f. According to SRP Section 9.1.1, design calculations should show that the storage racks and the anchorages can withstand the maximum uplift forces available from the lifting devices without an increase in k_{eff}. A statement in the Safety Analysis that excessive forces cannot be applied due to the design is acceptable if justification is provided.

g. It is the position of the Plant Systems Branch that the vaults and racks of the new fuel storage facility are to be designed to preclude damage from dropped heavy objects. Provide the design features included in the design which either preclude the fall of heavy objects onto the racks or preclude damage from a drop of the load with the maximum potential energy.

h. Reference to Section 9.1.1.3.1.2.D in Section 9.1.1.3.1.1, regarding potential moderators such as fire extinguishing aerosols, appears to be in error. Should it be 9.1.1.3.1.2.C?

i. According to SRP Section 9.1.1, the failure of non-seismic Category I systems or structures located in the vicinity of the new fuel storage racks should not cause an increase in k_{eff} beyond the maximum allowable. Provide analysis that this condition is met or include in your application a commitment to the above condition as a design criterion.

*Response 410.103

a. Although Regulatory Guide 1.13 pertains to the design of spent fuel storage racks, it is also used for the design of the new fuel racks. The applicable portions of the Regulatory Guide that are met are defined by Paragraphs 9......1.A and 9.1.1.1.C.

b. The "facilities" associated with new fuel storage consist of the storage vault and the rack restraint system. The seismic category of other building components associated with handling fuel assemblies is noted in Table 3.2-1. (see response to NRC RAI 210.1)

c. The L-insert slots are provided in the wall of the fuel rack cavity (box) to permit the L-insert to be locked to the fuel cavity by its locking tab after it has been installed. The design of the locking tab and slot is such that the L-inserts can be remotely removed from the fuel racks, if required.

The cell blockers are installed in the fuel racks before the fuel assemblies are placed in the fuel rack and before the pool is flooded. The design is basically two concentric tubes with end restraints that limit the engagement of the tubes in the rack cavity wall (to avoid protrusion into an adjacent fuel rack cavity). The tubes are collapsed, installed into the fuel rack cavity, expanded into the holes in the fuel rack cavity wall, then locked together with a captured pin. In this manner the cell blockers are positively locked to the fuel racks but can be remotely removed if desired.

*This response was previously transmitted by Reference A-13. Changes have been made as noted by the bars in the margin.

Question 410.107

a. Evaluate the structural design features of the refueling cavity water seal that would preclude a leak or failure from occurring. Include the possibility of a fuel assembly or other structure dropping on the seal.

b. If a seal failure/leak occurred, determine the time to lower a fuel assembly below the reactor vessel flange level before unacceptable dose rates from a lowered water level above spent fuel in the reactor core.

c. For a postulated seal failure/leak, evaluate containment dose rates from a lowered level above spent fuel in reactor core.

d. For a postulated seal failure/leak, evaluate the following parameters: makeup capacity, emergency procedures, fully loaded spent fuel pool thermal-hydraulic and dose effects including dose rate to someone trying to manually close the transfer tube valve to hydraulically isolate the spent fuel pool from the leak, time to cladding damage without operator action. Specifically provide the maximum allowable time to isolate the spent fuel pool from the transfer tube and refueling pool before there are unacceptably high dose rates in the spent fuel pool area and inadequate spent fuel pool cooling due to the level dropping below the minimum NPSH requirement above the elevation of the pool cooling suction inlet piping.

*Response 410.107

a. The refueling pool seal is designed to be installed in one piece between the reactor vessel flange and the pool floor. All fabrication welds will be liquid penetrant inspected prior to installation to ensure adequacy. After the seal assembly has been set in place, it will be permanently attached to the reactor vessel flange and to an embedment plate in the pool floor. Penetrations in the seal plate for ventilation and access to the ex-core instrumentation will be covered by bolted access hatches equipped with double seals when the pool is flooded. The annulus between the seals will be pressure tested after the hatches have been installed to determine the sealing adequacy.

*This response was previously transmitted by Reference A-13. Changes have been made as noted by the bars in the margin.

The pool seal is designed to withstand OBE displacements without leakage. The pool seal is designed to limit potential leakage resulting from SSE displacements. Pool seal inspection will be required as part of the post seismic recovery procedure. The pool seal is also designed to accommodate, without leakage, relative displacements between the pool floor and the reactor vessel due to normal plant operation.

During refueling operations with the pool flooded, the heavy lift components that pass over the pool seal are the reactor vessel head, the upper guide structure with its lift rig, and the upper guide structure lift rig. Administrative controls as provided in CESSAR-DC Section 9.1.4 require that prior to transfer of heavy loads over the pool seal, the fuel transfer tube valve or the gate between the fuel building transfer system canal and the spent fuel pool shall be closed. This is done to preclude any change to the spent fuel pool water level during a postulated heavy load drop on the pool seal which may result in containment pool draindown. In addition, administrative controls as provided in CESSAR-DC Section 9.1.4 preclude the movement of heavy loads over the pool seal if the refueling machine contains a fuel assembly. The refueling machine is designated seismic category II so that it will not fall on the pool seal during the seismic accelerations. The maximum clearance between the bottom of the refueling machine and the top of the pool seal is less than two inches to minimize the impact energy for the postulated accident condition of a dropped fuel assembly on the pool seal. The pool seal has been designed so that it will not leak as a result of this impact load.

b. It has been determined that a 24 square inch opening in the pool seal will result in pool draindown to the reactor vessel flange level in approximately 4 hours without additional water being added to the pool. It has also been determined that present plant systems are capable of maintaining the pool water level in the event there is a 24 square inch opening in the pool seal. A fuel assembly can be lowered below the reactor vessel flange level from the fully withdrawn position within 3 minutes.

c. With the water at the reactor vessel flange level, the radiation level at the pool seal area as a result of the fuel assemblies within the core will not be significantly greater than that with the reactor vessel head in place. The exposed CEA extension shafts will result in an increase in the overall radiation level. However, if it is necessary to do maintenance on the reactor vessel pool seal, temporary shielding can be placed around the extension shafts to reduce the radiation levels in the work area.

d. The responses to parts a, b, and c of this RAI address the concerns of this question.

Question 280.1

Provide the fire protection analysis and/or interface requirements to ensure that safe shutdown can be achieved, assuming that all equipment in any one fire area will be rendered inoperable by fire and that re-entry into the fire area for repairs and operator actions is not possible with exception of the control room. For the control room, provide the fire protection analyses and/or interface requirements having an independent alternative shutdown capability that is physically and electrically independent of the control room. Also, provide the fire protection requirements for redundant shutdown systems in the reactor containment building that will ensure, as much as practicable, that one shutdown division will be free of fire damage. Additionally, also ensure that smoke, hot gases, or the fire suppressant will not migrate into other fire areas to the extent that they could adversely affect safe shutdown capabilities, including operator actions.

*Response 280.1

A fire protection analysis of each fire area is conducted as part of the Fire Hazards Analysis. The System 80+ design basis, as stated in CESSAR-DC Section 9.5.1 (as revised in Amendment I), is to assure the ability to achieve Safe Shutdown following fire in any fire area outside of containment. This includes loss of all equipment in any given area and effects of electrical interaction which may disable equipment outside of the immediate area. The plant is arranged so that Safe Cold Shutdown can be achieved following fire in any area outside of containment without need for repairs or extraordinary operator action. Emergency shutdown from outside the control room is described in CESSAR-DC Sections 7.4.1 1.10 and 7.4.2.5. Outside of containment, redundant divisions of safety related equipment are separated by three hour fire rated boundaries. In the control complex (and most locations in the Nuclear Annex) redundant safety related divisions of protective electrical channels are separated by three hour fire rated barriers so that loss of all equipment in these areas would not affect either division of safety related equipment required to achieve cold shutdown. Inside containment Engineering Analysis conducted as part of the Fire Hazard Analysis assure that fire at any location which can disable more than one channel of cold shutdown equipment will not affect the ability to achieve cold shutdown using equipment which would not be affected by fire at that location.

Smoke control is recognized as an important element of the Plant Fire Protection design features. In the subsphere area, containment, Fuel Pool building, Reactor Annex and Diesel Generator building the HVAC System has smoke control capability by allowing any area to be purged with 100% outside air. In the control complex, dedicated smoke exhaust fans are provided for the control room and TSC. In addition, a smoke exhaust system is provided for each channel of safety related equipment. A connection to the normal HVAC system intake is used for fresh air supply. Smoke detectors are installed in return air ducts to alarm and annunciate in the control room. Smoke dampers are arranged for remote operation from the control room.

The System 80+ design does not have connections (door or ventilation openings) between redundant safety-related divisions. This further mitigates the possibility that smoke and products of combustion of fire suppression agents will affect redundant safety-related equipment.

*This response was previously transmitted by Reference A-8

A3.0 REFERENCES

A-1 Letter, Reactor Systems Branch RAIs, T. V. Wambach (NRC) to E. H. Kennedy (C-E), dated December 24, 1990

A-2 Letter, Reactor Systems Branch RAIs T. V. Wambach (NRC) to E. H. Kennedy (C-E), dated January 31, 1991

A-3 Letter, Reactor Systems Branch RAIs, T. V. Wambach (NRC) to E. H. Kennedy (C-E), dated May 13, 1991

A-4 Letter, Reactor Systems Branch RAIs, T. V. Wambach (NRC) to E. H. Kennedy (C-E), dated August 21, 1991

A-5 Letter, Plant Systems Branch RAIs, T. V. Wambach (NRC) to E. H. Kennedy (C-E), dated October 10, 1991

A-6 Lette Risk Assessment Branch RAIs, T. V. Wambach (NRC) to E. H. Kennedy (C-E), dated October 30, 1991

A-7 Letter LD-91-013, E. H. Kennedy (C-E) to USNRC, dated March 15, 1991.

A-8 Letter LD-91-014, E. H. Kennedy (C-E) to USNRC, dated March 26, 1991.

A-9 Letter LD-91-019, E. H. Kennedy (C-E) to USNRC, dated May 6, 1991.

A-10 Letter LD-91-024, E. H. Kennedy (C-E) to USNRC, dated May 16, 1991.

A-11 Letter LD-91-062, E. H. Kennedy (C-E) to USNRC, dated November 27, 1991.

A-12 Letter LD-91-071, E. H. Kennedy (C-E) to USNRC, dated December 24, 1991.

A-13 Letter LD-92-017, C. B. Brinkman (C-E) to USNRC, dated February 12, 1992.

A-14 Letter LD-92-008, E. H. Kennedy (C-E) to USNRC, dated January 29, 1992.