Shutdown Evaluation Report

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ABSTRACT

The purpose of this report is to provide shutdown information in support of the Advanced Power Reactor 1400 (APR1400) design. In designing the APR1400, Korea Hydro & Nuclear Power Co., Ltd., recognized the significance of addressing safety during shutdown operations. The APR1400 is designed with features that enhance shutdown safety by (1) deliberate system engineering, equipment specification and plant arrangements for shutdown operation, (2) mode-dependent control logic that assists and limits operations, (3) instrumentation, displays and alarms that clearly portray plant status in each mode and (4) procedural guidance and Technical Specifications that address important shutdown evolutions. This report presents these features and evaluates them in the context of the shutdown issues identified by the U.S. Nuclear Regulatory Commission.

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ACRONYMS AND ABBREVIATIONS

AAC	alternate alternating current
ac	alternating current
ACU	air cleansing unit
ADV	atmospheric dump valve
AF	auxiliary feedwater
AFAS	auxiliary feedwater actuation signal
AFW	auxiliary feedwater
AFWS	auxiliary feedwater system
AHU	air handling unit
ALWR	advanced light water reactor
ANS	American Nuclear Society
APR1400	Advanced Power Reactor 1400
AS	auxiliary steam
ASI	axial shape index
BAST	boric acid storage tank
BOP	balance of plant
BWR	boiling water reactor
CC	component cooling
CCS	component control system
CCW	component cooling water
CCWS	component cooling water system
CD	condensate
CEA	control element assembly
CEDM	control element drive mechanism
CENP	Combustion Engineering Nuclear Power
CET	core exit thermocouple
CFD	computational fluid dynamics
CFR	Code of Federal Regulations
COL	combined license
CPC	core protection calculator
CRO	control room operator

CRT	cathode ray tube
CS	containment spray
CSAS	containment spray actuation signal
CSS	containment spray system
CVCS	chemical and volume control system
CW	circulating water
DBA	design basis accident
DBE	design basis event
dc	direct current
DC	Design Certification
DCD	Design Control Document
DEDLS	double ended discharge leg slot
DEHLS	double ended hot leg slot
DESLS	double ended suction leg slot
DF	decontamination factor
DG	diesel generator
DHR	decay heat removal
DLS	diesel loading sequence
DNBR	departure from nucleate boiling ratio
dP	pressure differential
DVI	direct vessel injection
EAB	exclusion area boundary
ECC	emergency core cooling
ECCS	emergency core cooling system
EDG	emergency diesel generator
EDS	electrical distribution system
EDT	equipment drain tank
El.	elevation
EOG	emergency operating guidelines
EPRI	Electric Power Research Institute
ESF	engineered safety features
ESFAS	engineered safety features actuation system
ESW	essential service water

ESWS	essential service water system
FC	fire control
FP	fire protection
FPD	flat panel display
FPS	fire protection system
FW	feed water
GDC	General Design Criterion
GIS	generated iodine spike
HEC	hazard elimination category
HEPA	high-efficiency particulate absorption
HJTC	heated junction thermocouple
HPI	high pressure injection
HPSI	high pressure safety injection
HSI	human system interface
HVAC	heating, ventilation, and air conditioning
HVT	holdup volume tank
HX	heat exchanger
I&C	instrumentation and control
IBD	inadvertent boron dilution
IBW	inverse boron worth
IC	inside containment
ICI	in-core instrumentation
IHA	integrated head assembly
INPO	Institute for Nuclear Power Operations
IP	instrument power
IPS	information processing system
IRWST	in-containment refueling water storage tank
KEPCO	Korea Electric Power Corporation
KHNP	Korea Hydro and Nuclear Power Co., Ltd.
LCO	limiting condition for operation
LDP	large display panel
LER	licensee event report
LOCA	loss of coolant accidents

LOCV	loss of condenser vacuum
LOOP	loss of offsite power
LPSI	low pressure safety injection
LRWLIS	local refueling water level indication system
LTOP	low temperature overpressure protection
MCC	motor control center
MCR	main control room
MMI	man-machine interface
MMIS	man-machine interface system
MS	main steam
MSIS	main steam isolation signal
MSIV	main steam isolation valve
MSLB	main steam line break
NPSH	net positive suction head
NPSHA	net positive suction head available
NR	neutron monitoring system
NRC	Nuclear Regulatory Commission
NSSS	nuclear steam supply system
NUMARC	Nuclear Utility Management and Resources Council (now Nuclear Energy Institute)
NUREG	U.S. Nuclear Regulatory Commission Regulation
P&ID	piping and instrumentation diagram
P/T	pressure/temperature
PCT	peak clad temperature
PNS	normal permanent non-safety
PORV	power-operated relief valve
POSRV	pilot operated safety relief valve
PRA	probabilistic risk assessment
PRWLIS	permanent refueling water level indication system
PWR	pressurized water reactor
PZR	pressurizer
QIAS	qualified indication and alarm system
QIAS-P	qualified indication and alarm system – post-accident monitoring system
RC	reactor coolant

RCFC	reactor containment fan cooler
RCP	reactor coolant pump
RCS	reactor coolant system
RDS	rapid depressurization system
RDT	reactor drain tank
RG	(1) Regulatory Guide (2) gas vent system
RHR	residual heat removal
RPS	reactor protection system
RPV	reactor pressure vessel
RTD	resistance temperature detector
RV	reactor vessel
RWLIS	refueling water level indication system
RWST	refueling water storage tank
SAFDL	specified acceptable fuel design limit
SC	shutdown cooling
SCS	shutdown cooling system
SDC	shutdown cooling
SDM	shutdown margin
SECY	Secretary of the Commission, Office of the NRC
SFP	spent fuel pool
SG	steam generator
SGTR	steam generator tube rupture
SI	safety injection
SIAS	safety injection actuation signal
SIP	safety injection pump
SIS	safety injection system
SIT	safety injection tank
SR	safety related
SRM	source range monitor
SRP	Standard Review Plan
SS	sampling system
STE	special test exception

SUFW	startup feedwater
SW	service water
SWGR	switchgear
TEDE	total effective dose equivalent
TS	(1) Technical Specification (2) Trade Secret
UAT	unit auxiliary transformer
UGS	upper guide structure
ULMS	ultrasonic level measurement system
UMT	unit main transformer
VCT	volume control tank
VDA	variable display area
WL	raw water system

1.0 INTRODUCTION

1.1 Purpose

This report describe the features of the Advanced Power Reactor 1400 (APR1400) design that address the issues related to shutdown evaluation and presents an evaluation of these features with respect to their ability to reduce and/or mitigate the consequences of shutdown risk. The results of a probabilistic risk assessment (PRA) of shutdown are presented in Subsection 19.1.6 of the APR1400 Design Control Document (DCD) Tier 2.

1.2 Scope

Subsections 2.1 through 2.13 present detailed description of shutdown issues. Sections 3, 4 and 5 of this report present an evaluation of the applicability of the analyses in Chapters 6 and 15 of the APR1400 DCD Tier 2 to loss-of-coolant accidents (LOCAs) and other types of accident events that are initiated from shutdown modes. Section 6 evaluates the features of APR1400 that simplify shutdown operations and thereby reduce the potential for initiating shutdown events. Conclusions are provided in Section 7.

1.3 Background

In 1988, the U.S. Nuclear Regulatory Commission (NRC) issued recommendations in Generic Letter No. 88-17 (Reference 1) to all license holders for pressurized water reactors (PWRs) 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 of the recommendation was to prevent events with a potential for core damage and/or release of radiation. NRC staff evaluations of shutdown operations in 1993 in NUREG-1449 (Reference 2) indicate that recommendations had been implemented and/or were underway at operating plants.

On February 4, 1999, the NRC withdrew a proposed rule for shutdown and low-power operations in nuclear power reactors. Despite the withdrawal, the methodology in the CESSAR Design Certification (DC) Appendix 19.8A (Reference 3) for shutdown risk analysis can be applied to the APR1400 design to address shutdown risk issues and evaluate design features with respect to their ability to reduce and/or mitigate the consequences of this risk.

In 1997, in “Briefing on Shutdown Risk Proposed Rule for Nuclear Power Plants” (Reference 4), the NRC found NUMARC 91-06 (Reference 5) to be "...an acceptable methodology... A (Shutdown Risk) program developed and consistent with NUMARC 91-06... Would most likely meet the requirements of (the proposed NRC) rule..." However, NUMARC 91-06 was developed for operating plants, and not all relevant APR1400 shutdown risk analysis issues are covered in NUMARC 91-06 (i.e., the APR1400 DC encompasses more than NUMARC 91-06).

The NRC positions on shutdown risk issues are presented in NUREG-1449 (Reference 2). Since SECY-97-168 (Reference 6) was proposed by the NRC, it should also be considered in shutdown risk responses, as identified for the following issues.

a. Primary and secondary containment capability and source term

   This issue 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.

   Primary containment and source term are addressed in SECY-97-168 (Reference 6).

b. Fire

The risk of fire during shutdown operations is higher than when the plant is in power operation. The 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 shutdown mode, resulting in a higher susceptibility of failure due to fire.

Fire protection is one of the three major parts of the proposed rule in SECY-97-168. The fire protection part of the rule is intended to extend the fire protection provisions already provided during power operation to shutdown operation. The proposed rule would require licensees to implement measures to minimize the frequency of fires during shutdown operations. It would also require that the decay heat removal (DHR) function be maintained free of fire damage, or that fire damage be limited by promptly detecting, controlling, and extinguishing fires that do occur. It would further require that contingency plans be developed to provide reasonable assurance of adequate core cooling and restoration of DHR following a fire. The provisions necessary for implementation would be documented in the licensee's fire protection plan.

c. Flooding and spills

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 (CCWS), service water system, condensers, and refueling pool seals. The issue that is addressed here is the potential for loss of DHR as a consequence of spills and internal flooding that may disable components of the shutdown cooling system (SCS).

Flooding and spills are addressed within SECY-97-168 in relation to DHR and reactor coolant system (RCS) inventory control.

d. Emergency core cooling system (ECCS) recirculation capability

This 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, including:

(1) Hydraulic effects, such as air ingestion and vortex formation

(2) Debris in the in-containment refueling water storage tank (IRWST) resulting from maintenance activities or the deterioration of insulation from actuation of containment sprays or from LOCA consequences

(3) The combined effects of items (1) and (2). This is addressed in SECY-97-168 in the same section as DHR and RCS inventory control for flooding and spills

e. Applicability of the containment analysis in APR1400 DCD Tier2, Chapter 6

This issue is addressed within SECY-97-168 in the context of maintaining a mitigation capability to provide sufficient protection against the uncontrolled release of fission products.

f. Fuel handling and heavy loads during shutdown operation

This issue is addressed in NUREG-1449 (Reference 2). See Section 5.1.2.6 and Section 6.11 in which NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants" (Reference 7), is referenced. SECY-97-168 does not address this subject.

1.4 APR1400 Features

In this subsection, the characteristics of past accident events are compared to the APR1400 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. The categories are nearly identical to the issues identified in SECY-91-283 (Reference 8). Each category encompasses similar events with the same type of event initiator. If unmitigated by automatic or manual actions, all events could eventually lead to overheating and/or physical damage to fuel with consequent radiation release, but the sequence steps in the events may differ. The same events may be grouped differently depending on the importance placed on each step in a sequence. For example, the NUREG-1449 category, "Loss of Shutdown Cooling", includes the issues as mid-loop operation, loss of DHR capability and effect of PWR upper internals.

The categories of past events used in this subsection encompass (and for some categories are identical to) the issues and 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 there is no 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 are therefore not included in the categories of past events.

Past events are grouped into the following 10 categories:

a. Loss of shutdown cooling

b. Loss of electrical power

c. Loss of reactor coolant

d. Containment integrity

e. Overpressurization

f. Flooding and spills

g. Boron and reactivity events

h. Fire protection

i. Heavy loads and fuel handling

j. Mode change events

The event initiator is identified for each past event placed into a category. 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 used in the APR1400 design.

Table 1.4-1 provides an overview of the APR1400 design features that avoid core damage during shutdown operating modes. It lists the 10 shutdown event categories and for each category, lists event initiators for past events. The initiators are presented in a generic fashion with each initiator representing many events that have occurred. For each initiator, the features of the APR1400 design that are available to prevent an occurrence of an initiator and/or to mitigate the consequences of an initiator are listed.

Table 1.4-1 (1 of 5)

Shutdown Event Categories and APR1400 Features for Prevention, Detection, and Mitigation

EVENT CATEGORY	EVENT INITIATOR	APR1400 FEATURES FOR PREVENTION, DETECTION, AND MITIGATION
1 Loss of Shutdown Cooling	SCS flow loss by pump suction vortex	A Mid-loop level maximized by locating SCS suction piping at the bottom of the hot leg. B Mid-loop level is increased by decreasing SG inlet nozzle angle from 45° to 40°. C There are no loop seals in the suction line. D One SCS suction line from each hot leg provides SCS redundancy with separation of pump suction sources. E Containment spray pumps interchangeable with SCS pumps provide redundant capacity and may take suction from IRWST to refill RCS and mitigate gas binding.
	Inaccurate mid-loop level leading to suction vortex	F With head on, refueling water level indication system (RWLIS) indications from vessel head to a level below that required for SCS operation. Level indication is accurate for intended use. G Core exit thermocouples monitor coolant temperature down to 37.8 °C (100 °F) prior to withdrawal of CETs prior to fuel shuffling. The RTDs and SCS temperatures are accurate during SCS operation. H With head off, level indication near hot leg elevation is provided by high resolution (RWLIS). I SCS performance monitored on each of two SCS pumps by pump motor current, flow rate, discharge pressure and suction pressure. Possible SCS flow variance with decay heat to minimize potential for vortexing during mid-loop operation.
	Loss of flow while head off, upper internals in vessel and cavity flooded leads to core heatup	J Internals design limits coolant flow from cavity to core, but high availability of SCS system and/or backups provides reasonable assurance of forced convection.

Table 1.4-1 (2 of 5)

EVENT CATEGORY	EVENT INITIATOR	APR1400 FEATURES FOR PREVENTION, DETECTION, AND MITIGATION
1 Loss of Shutdown Cooling (Continued)	Various low level and loss of RHR events	K Non-shared SCS allows SCS maintenance and testing during Modes 1-4 prior to cold shutdown, increasing availability in Modes 5 and 6. M Shutdown specific control room displays, Technical Specifications, and procedural guidance reduce likelihood of personnel errors. N Inadvertent errors are reduced and early operator evaluation of failures is improved by (1) LDP 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. O Prevention of inappropriate automatic actions from personnel errors by shutdown specific control logic (e.g., remove auto-closure interlocks from SCS suction valves.) P CCW availability is increased by two redundant divisions, each with two pumps and heat exchangers. Q Service water availability is increased by two redundant divisions, each with two pumps. R Each SCS division has four potential sources of ac power for increased availability.
2 Loss of Electrical Power	Equipment failure and/or inadvertent personnel error leading to loss of power and shutdown cooling	A Alternate ac gas turbine generator provides third on-site power source. B Two switchyard interfaces provide flexibility. C Shutdown specific Technical Specifications and procedural guidance reduce likelihood of personnel errors. D A reserve transformer provides an alternate supply to the safety bus if the normal source (unit auxiliary transformer) is de-energized. E Each safety division has a dedicated diesel generator. F No equipment is shared between diesels. G No equipment is shared with another unit.

Table 1.4-1 (3 of 5)

EVENT CATEGORY	EVENT INITIATOR	APR1400 FEATURES FOR PREVENTION, DETECTION, AND MITIGATION
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-1 Inadvertent errors are reduced and early operator evaluation of failures is improved by (1) LDP 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. A-2 Increase in design pressure to 63.3 kg/cm^3 (900 psig)
	Inadvertent RPV pressurization while connected systems are open causing coolant level drop in vessel	B Removal of the PZR manway does not allow significant RV head pressurization, and instruments are therefore 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 Technical Specifications require hatch and penetrations closed during mid-loop operation. Containment configuration and size allow more outage activities within containment, resulting in less time without containment integrity. B Redundancy in SCS, 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 control room displays, Technical Specifications, and procedural guidance reduce likelihood of personnel errors.

Table 1.4-1 (4 of 5)

EVENT CATEGORY		EVENT INITIATOR	APR1400 FEATURES FOR PREVENTION, DETECTION, AND MITIGATION	
5	Overpressurization	Inadvertent safety injection actuation at low temperature pressurizes RCS and SCS	A	SCS relief valves sized for maximum safety injection liquid flow.
			B	RCS is vented through the PZR manway.
			C	Ring forged RV 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) LDP 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 control room displays, Technical Specifications, 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, and floor drains.
7	Boron and Reactivity Events	Various CVCS misoperations and uncalibrated source range neutron monitors cause approach to criticality	A	Shutdown specific control room displays, Technical Specifications, 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 the need for most heat tracing.
			C	Boron dilution alarm provides warning.
			D	Provide protection for prevention of a boron dilution.

Table 1.4-1 (5 of 5)

EVENT CATEGORY		EVENT INITIATOR	APR1400 FEATURES FOR PREVENTION, DETECTION, AND MITIGATION	
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 B	Plant layout and fire barriers separate redundant divisions and systems to limit potential fire damage. 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 B C	Shutdown specific guidance limits pathways for heavy lifts. Plant arrangement minimizes potential for damaging drops. 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 B	Shutdown specific control room displays, Technical Specifications, and procedural guidance reduce likelihood of personnel errors. Inadvertent errors are reduced and early operator evaluation of failures is improved by (1) LDP 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.

2.0 Shutdown Evaluation Issues

Subsections 2.1 through 2.13 present an evaluation of the shutdown issues that are identified in Reference 3, which are as follows:

a. Procedures

b. Technical Specification improvements

c. Reduced inventory operations and General Letter (GL) No. 88-17 (Reference 1) fixes

d. Loss of decay heat removal (DHR) capability

e. Containment capability and source term

f. Rapid boron dilution

g. Fire protection

h. Instrumentation

i. Emergency core cooling system recirculation capability

j. Effects of pressurized water reactor upper internals

k. Fuel handling and heavy loads

l. Potential for draining the reactor coolant system (RCS)

m. Flooding and spills

The evaluation of each issue is divided into four topics as follows:

a. Issue – Statement of the issue in conformance with the interpretation and evaluation of the issue in NUREG-1449

b. Acceptance criteria – The acceptance criteria that are used to evaluate the APR1400 design to prevent and/or mitigate unacceptable consequences of the issue.

c. Description – The postulated plant scenarios, analyses, and evaluations that are considered to provide reasonable assurance that the issue is adequately addressed.

d. Resolution – Statement of how the issue is resolved by the APR1400 design.

2.1 Procedures

2.1.1 Issue

The operational guidance provided by the plant designer to the operator may not be sufficient to provide reasonable assurance that the plant operator can develop procedures for the detection, mitigation, and/or recovering from abnormal events initiated from shutdown operations.

2.1.2 Acceptance Criteria

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

2.1.3 Description

The APR1400 design incorporates advanced features that promote safe and simple plant operation. The features include redundancy and diversity of components and systems, dedicated and/or permanently aligned systems, and an advanced information system that better informs the operations staff of plant status and available recovery paths if an abnormal event occurs. These features also contribute to improved operability and maintainability that significantly reduce the initiating situations that have contributed to increased shutdown evaluation.

The 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 is generally in the form of suggested operational sequences that preserve the safety bases of the design. Since shutdown operations are 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 that instruct the operator in the use of design features that can detect, mitigate, and/or assist recovery from abnormal events that can occur during shutdown operations.

A summary of general operational guidance related to shutdown operations is provided in Table 2.1-1. Details are contained in the appropriate subsections.

An outline of the operational guidance developed to support the RCS reduced inventory operations is provided in the Appendix A of this report. This guidance, together with supporting information, is sufficient for an operator to develop an operational procedure for reduced RCS inventory operations. The development of a detailed procedure by the operator requires equipment characteristics of procured components and the results of the pre-operational testing in APR1400 Design Control Document (DCD) Tier 2, Chapter 14 (Reference 9), to determine system performance values. For example, to support mid-loop operation, the following would be measured during plant startup: shutdown cooling system (SCS) flows, suction line vortexing characteristics, level instrumentation calibration, and others. The data would be used to verify performance and provide operational data. Testing requirements for shutdown-oriented instrumentation are given in APR1400 DCD Tier 2, Chapter 14.

2.1.4 Resolution

The issue of procedures for shutdown operation is resolved for the APR1400 by providing operational guidance to address the use of advanced design features to detect, mitigate, and/or assist recovery from abnormal events initiated from shutdown operations.

Table 2.1-1 (1 of 3)

Summary of Procedural Guidance Related to Shutdown Operations

Topic	Procedural Guidance	Report Section
Unplanned Drainage of Reactor Coolant	• Prevention • Administrative control of major potential drain down paths identified for shutdown modes	2.12.1 2.12.2.1 2.12.3 2.12.3.2.1 2.12.4
	• Monitoring of instrumentation for RCS level, inventory and temperature controls a. Refueling pool level b. Containment sump level c. Level indicators and alarms: EDT, RDT, IRWST, HVT, VCT d. RCS operational leakage (Technical Specifications surveillance) e. RCS level indicators and alarms (1) Pressurizer level instrumentation (2) Permanent refueling water level indication system, wide range, dP based refueling water level instrumentation (3) Permanent refueling water level indication system, narrow range, dP based refueling water level instrumentation (4) Local refueling water level indication system (sight glass) (5) Ultrasonic level measurement system (6) Heated junction thermocouple f. Pressurizer pressure g. RCS temperature (1) Core exit thermocouples (2) Resistance temperature detectors (when SCS flow is lost, the RTDs are used for trending only) (3) Heated junction thermocouple h. Shutdown cooling system (1) SCS flowrate (2) SCS/CSS pump discharge pressure (3) SCS/CSS pump motor current (4) SCS/CSS pump suction pressure (5) SCS HX inlet/outlet temperature i. SG parameters	2.3.3.1 2.8 Table 2.8-1 2.12.4

Table 2.1-1 (2 of 3)

Topic	Procedural Guidance	Report Section
Unplanned Drainage of Reactor Coolant (Cont'd)	• Mitigation (immediate operator action) a. Identify leakage path b. Isolate leakage path c. Make up losses (1) Safety injection (2) SCS via IRWST (3) Containment spray from IRWST via SCS lines (4) Charging pumps (5) BAST (6) Safety injection tanks	2.3.3.4 2.12.2.3 2.12.3 2.12.4 EOG
Heavy Loads (1) Drop of transported equipment (2) Drop of fuel bundle (3) Refueling pool seal integrity (4) Loads over ICI table	• Restrictions specified for: a Lift height b Travel directions c Systems lineup (specified in, APR1400 DCD Tier 2 Chapter 9)	2.11.3
Outage Maintenance	• Strategy for shutdown operations a. Define operating and operational divisions. b. Limit maintenance activities to components and systems not included in a	2.4.3.2.2 EOG
Fire Protection	• Administratively require fire protection systems to remain operable in shutdown modes • Procedurally control: a. Combustible materials b. Housekeeping c. Hot work • Pre-fire plan a. Outline fire fighting strategies b. Monitor status of fire barriers	2.7.3.2 2.7.3.3 2.7.3.2

Table 2.1-1 (3 of 3)

Topic	Procedural Guidance	Report Section
RCS Cooling Using Feed and Bleed (other systems not available)	• RCS pressurized a. Start SI pumps b. Reduce pressure through RDS venting to IRWST (Maintain subcooled temperatures in RCS) c. Secure operating RCPs (if applicable) d. Cycle SI feed and RDS bleed to reduce RCS pressure and temperature e. When depressurized, open RDS and run SI continuously f. Align SCS HX for IRWST cooling g. Restore Normal SCS • RCS depressurized a. Start SI pumps b. Open RDS c. Secure RCP's (if RCS not vented) d. Align SCS HX for IRWST cooling e. Restore normal SCS	2.4.3.1.3.1.1 2.4.3.1.3.2.1
SG Tube Rupture	• Include in EOG a requirement to maintain a positive primary to secondary pressure differential	Table 2.6-1 Section C(A)–
Containment integrity	• Administratively require containment integrity to remain intact during RCS fillup and drainout in mode 5	2.5.4.2

2.2 Technical Specification Improvements

2.2.1 Issue

When a plant is operated within the limiting conditions for operation (LCO) provided by the Technical Specifications, the consequences of design basis events (DBEs) should be bounded by the results of the safety analyses. However, limiting conditions for operation developed for power operation may not be sufficient to provide reasonable assurance that the consequences of events initiated from shutdown modes are bounded by the analyses. Technical Specifications should include the necessary limiting conditions for operation that are applicable to shutdown modes.

2.2.2 Acceptance Criteria

Technical Specifications ensure that when the plant is operated within the LCOs applicable to the mode of operation, consequences of DBEs are bounded by the results of safety analyses for that mode.

2.2.3 Description

The APR1400 design incorporates advanced features that 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 that informs the operations staff of plant status, and available recovery paths if an abnormal event occurs. These features also contribute to improved operability and maintainability that significantly reduce the initiating situations that have contributed to increased shutdown risk.

One objective of the plant design 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 provide reasonable assurance that 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 increase risks by prolonging the shutdown period and adding to staff stress. The objective of shutdown evaluation for the APR1400 relative to Technical Specifications is to modify the existing Technical Specifications to the extent necessary to address event initiators not fully covered by the analysis of the traditional DBEs.

A summary of the proposed Technical Specification modifications is provided in Table 2.2-1. The Technical Specification modifications and additions reflect:

(1) Added redundancy and diversity of the APR1400 design that allows these modifications without affecting operational flexibility;

(2) Analysis of events initiated during shutdown operations;

(3) Assessment of the risk of operating in these plant configurations for extended periods (e.g., refueling, unplanned maintenance)

2.2.4 Resolution

The issue of shutdown specific Technical Specifications is resolved for APR1400 by modifications to the Technical Specifications based on the safety analyses performed for Modes 2 through 6. These modifications provide additional reasonable assurance that the consequences of transients and accidents that may occur during shutdown modes of operation are less limiting than those described in Chapters 6 and 15 of the APR1400 DCD Tier 2.

Table 2.2-1 (1 of 2)

APR1400 Technical Specifications Modifications Related to Improved Shutdown Operations

Item	Technical Specifications No.	Title	LCO	Bases
1	3.1.1	Shutdown Margin (SDM)	Change mode applicability. Add Kn-1 and EC requirements.	Extended applicability. Provide protection for ejected CEA and CEA group withdrawal in shutdown modes
2	3.3.2	RPS Instrumentation : Shutdown	Specify the modes of applicability in APR1400 DCD Tier 2, Chapter 16, Table 3.3.2-1. Extends SG pressure-low to Mode 3 and RC flow-low to Modes 3, 4, and 5 when the CEAs can be moved.	Provide reactor trip function for steam line break (SLB) in shutdown modes and for unplanned CEA group withdrawal.
3	3.3.5	ESFAS Instrumentation	Add Mode 4 to CSAS mode applicability	Provide reasonable assurance of availability of automatic CSAS for mitigation of LOCA event in shutdown mode.
4	3.5.3	SIS: Shutdown	Change restriction on number of SI pumps operable to 2.	Two SI trains required operable in applicable modes.
			Extend requirements for two SI trains to Modes 4, 5, and 6. (at mode 6 when RCS water level is lower than 39.7 m (130 feet))	Required for RCS inventory makeup for LOCA events in lower operating modes.
5	3.5.4	IRWST	(1) Extend operability requirements to Modes 5 and 6. (2) Specify maximum water temperature at 49°C(120°F).	(1) For compatibility with Technical Specification 3.5.3. (2) Presently stated in the Bases.
6	3.8.2	AC Sources: Shutdown	Require one circuit between the offsite transmission network to each onsite Class 1E distribution system in Modes 5 and 6.	Provide additional backup ac power source.
7	3.3.14	Boron Dilution Alarm	Both boron dilution alarms are operable in Modes 3, 4, 5, and 6.	Provide additional protection for prevention of an inadvertent boron dilution of the RCS.
8	3.3.10	Containment Bypass Instrumentation (SG Tube Rupture)	Require radiation monitoring instrumentation for (1) SG liquid blowdown (2) Steam line	Required for SG tube rupture detection in shutdown modes.

Table 2.2-1 (2 of 2)

Item	Technical Specifications No.	Title	LCO	Bases
9	3.1.6	Shutdown CEA Insertion Limits	Special test exceptions and applicability only to critical conditions included in Technical Specification 3.1.10.	Clarify applicability and STEs.
10	3.1.7	Regulating CEA Insertion Limits	STEs and applicability included in Technical Specification 3.1.9 and 3.1.10.	Same as Item 9.
11	3.3.6	ESFAS Logic and Manual Trip	Add Mode 4 to CSAS mode applicability	Provides reasonable assurance of availability of manual CSAS for mitigation of LOCA event in shutdown mode.
12	3.4.11	LTOP	Delete requirement for SI pumps.	Two required for shutdown. LTOP sizing increased and temperature lowered to avoid pressurized thermal shock.
13	3.8.5	DC Sources: Shutdown	Clarify LCO to provide most reliable line up.	Prevents loss of operable D/G due to maintenance.
14	3.8.10	Distribution Systems: Shutdown	Clarify LCO to provide most reliable line up.	Prevents loss of operable D/G due to maintenance.
15	3.9.5	SCS and Coolant Circulation (Low Water Level)	Require additional SDC division to be operable.	Allows increased reliability for shutdown cooling.
16	3.6.7	Containment Protection (Reduced RCS Inventory Operations)	Require that equipment hatch close during reduced RCS inventory operations	Minimizes the release of radioactivity from containment.

2.3 Reduced Inventory Operations and Generic Letter No. 88-17 Fixes

2.3.1 Issue

Plant events that have occurred in the industry have highlighted the need for a close examination of operations during reduced inventory conditions in the RCS. Following the Diablo Canyon incident, the NRC published Generic Letter No. 88-17 (Reference 1), which requires 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. Areas of concern include:

a. Instrumentation that would greatly improve the operator’s monitoring capability during reduced inventory operations

b. The availability of existing equipment for use in mitigating a loss of SCS or RCS inventory

c. Nozzle dam installation procedures, which would provide reasonable assurance that a vent pathway is available so RCS pressurization can be minimized if shutdown cooling is lost

d. Alternate ways to add inventory to keep the core covered if the SCS is lost

e. Administrative procedures that would avoid RCS perturbations during reduced inventory operations

f. Containment closure issues

In Generic Letter No. 88-17, the NRC specified that programmed enhancements should accomplish a comprehensive improvement in a 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. The NRC recommended that licensees implement a means of preventing accident initiation, monitoring progressions that may lead to core damage, and evaluating consequences and, where needed, provide mitigation.

2.3.2 Acceptance Criteria

The APR1400 design reflects a comprehensive consideration of shutdown and lower power risk by adequately addressing all GL 88-17 recommendations and other issues relevant to reduced inventory operations, especially in instrumentation, Technical Specifications, procedures, equipment availability, and analyses.

2.3.3 Description

During plant shutdowns, certain maintenance and testing activities require a drain down of the RCS to a partially filled condition. Normal maintenance activities include the replacement of reactor coolant pump (RCP) seals and journal bearings. A testing activity requiring RCS drain down is the inservice inspection of the steam generator (SG) 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 the mid-loop level, nozzle dams are installed on the SGs, 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 losing shutdown cooling is increased due to the possibility of vortexing at the SCS suction line interface with the hot leg. In the worst-case scenario, subsequent to vortexing in the SCS suction line, a large percentage of air is entrained into the SCS suction piping and the SCS pump performance is degraded or interrupted. If

SCS operation is not re-established, core boiling and pressurization can produce rapid core uncovery.

APR1400 design features result in practical and significant benefits during reduced inventory operations. These design features are described in Subsections 2.3.3.1 through 2.3.3.5.

2.3.3.1 Instrumentation for Shutdown Operations

Diverse, accurate, and redundant instrumentation including main control room (MCR) large display panel (LDP) and information flat panel displays (FPDs) give continuous system status and provide the operations staff with precise information to monitor reduced inventory operations and to respond to loss of shutdown cooling events if they occur. Detailed information on reduced inventory instrumentation is included in Subsection 2.8.

Analyses form the basis for instrument design and calibration to provide reasonable assurance of correct instrument operability during reduced inventory states. Phenomena that can affect instrumentation operation are considered in the recommended use of instrument types for various scenarios. Reasonable assurance of instrumentation availability during reduced inventory operations via modifications to the Technical Specifications are identified in Subsection 2.2.

A general description of the types of instrumentation that are used for reduced inventory operations is as follows:

a. The refueling water level indication system (RWLIS) consists of the permanent refueling water level indication system (PRWLIS), local refueling water level indication system (LRWLIS), and ultrasonic level measurement system (ULMS). Each consists of two trains to provide a means of indicating RCS water level to the MCR during drain-down operations. The RWLIS includes an alarm function to alert operator to low-low, low, and high RCS water level. The PRWLIS, which consists of redundant and independent wide- and narrow-range level sensors, is provided for continuous monitoring of RCS level during drain-down operations. The PRWLIS provides monitoring capability from the pre-drain-down normal level in the pressurizer (PZR) to a point lower than that required for SCS operation. The PRWLIS is calibrated for low-temperature operation.

   The wide-range PRWLIS covers drainage from the PZR to the bottom of the hot leg and is available with the head on and off the reactor vessel (RV). The narrow-range PRWLIS covers reduced inventory operations and is also available with head on and off the RV. The PRWLIS is accurate for measuring the level in the hot leg. During a drain down, level monitoring is transferred from the wide-range-level instruments to the narrow-range instruments when the greatest degree of accuracy is required during operations with level in the hot-leg region.

   Each train of the LRWLIS is a sight glass that has a minimum visible span of 3.81 m (150 in) above the bottom of the hot leg covering the reduced inventory operations. The ULMS is temporarily installed on both hot legs to monitor the water level of the hot leg during mid-loop operation.

b. 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 RV.

c. SCS operation monitoring instrumentation provides reasonable assurance of precise knowledge of the status of the operating SCS loop including pressure, temperature, flow, and pump motor current indicators.

2.3.3.2 Shutdown Cooling System Design

The functional design of the SCS is substantially complete for APR1400. Design features that improve SCS performance during shutdown operation are as follows:

a. The APR1400 SCS suction lines do not contain any loop seals. The suction piping also has high point vents in order to release the gas accumulation. SCS pumps can be restarted with suitable venting procedures, providing reasonable assurance of an expedited reflood of the shutdown cooling pump suction.

b. The two SCS suction lines are independent of and redundant to each other. Problems associated with a suction line do not limit the other shutdown cooling train from being operated after level recovery (if necessary) for continued DHR.

c. The two containment spray system (CSS) pumps are interchangeable with 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 provide reasonable assurance of pump availability during shutdown operations.

d. There are no auto-closure interlocks on the shutdown cooling suction piping valves with potential for disturbing shutdown cooling. Although previous designs (e.g., Optimized Power Reactor 1000 [OPR1000]) included interlocks to isolate the SCS in the event of an unanticipated RCS pressurization during shutdown cooling, this interlock has been deleted from the APR1400 design to reduce the likelihood of losses of SCS.

2.3.3.3 Steam Generator Nozzle Dam Integrity

The APR1400 design addresses the NRC concern of preventing significant pressurization in the upper plenum of the RV during core boiling scenarios. APR1400 procedural guidance recommends a nozzle dam installation and removal sequence by which the nozzle dams are installed in the cold legs first. After the cold leg nozzle dams are installed, the hot leg nozzle dams can be installed. Likewise, when removing the nozzle dams, the hot leg nozzle dams are removed first, cold leg nozzle dams can subsequently be removed. This installation and removal procedure maximizes the time that the SGs are available for reflux boiling in the case of a loss of shutdown cooling, and minimize the time that both hot legs are simultaneously blocked by nozzle dams. This installation procedure requires the PZR manway to be open so a hot side vent pathway exists prior to blocking both RCS hot legs with nozzle dams.

In the APR1400 design, the ability of the RCS to withstand abnormal pressurization during reduced inventory operations with the nozzle dams installed is limited by the abnormal design pressure of the SG hot and cold leg nozzle dams. Based on transient overpressure analysis performed on nozzle dams, a conservative value of [ ]TS is assumed for this pressure limit. In order to provide reasonable assurance that the nozzle dam abnormal design pressure limit is not exceeded during reduced inventory operations with boiling conditions in the RV, the APR1400 design includes a requirement that is imposed to establish a mid-loop vent pathway via the PZR manway before operating in reduced inventory. When the manway is opened to the containment atmosphere, the surge line provides sufficient venting capacity to prevent RCS pressurization and subsequent nozzle dam failure. The PZR surge line vent pathway is also of sufficient capacity to prevent core uncovery due solely to pressurization of the hot side resulting from boiling in the core coolant.

The PZR manway is closed except during normal RCS drain-down activities. During a normal drain down to reduced inventory operations, when the PZR level decreases to a pre-established setpoint, the RCS vent pathway is aligned by opening the PZR manway. Following refueling operations, RCS integrity is not re-established until the RCS coolant level reaches the PZR. Only at that point is the manway re-installed. The mid-loop vent alignment allows sufficient venting of the RCS to the PZR cubicle if SCS is lost, resulting in onset of core boiling.

Analysis is performed to investigate the possibility of nozzle dam failure during reduced inventory operations (i.e., the RCS water level is maintained at 0.91 m (3 ft) below RV flange). Figure 2.3.3-1 shows the result of hot leg pressure, and the peak pressure is TS , which is lower than abnormal design pressure of nozzle dam. Analyses have indicated that opening the PZR manway and relieving it to the PZR cubicle are sufficient for venting the RCS during RCS boiling and preventing SG nozzle dam failure.

An acceptable, conservative RCS equilibrium pressure that is below the assumed SG nozzle dam abnormal design pressure limit has been calculated when mid-loop operation is assumed to start at 4 days after shutdown. Therefore, the recommended earliest time after shutdown (from full power) for operating at mid-loop level is 4 days. Based on industry operational data, a reasonable minimum RCS cooldown from Mode 1, followed by a drain down from normal RCS level to mid-loop, can be performed in approximately 4.5 days. Therefore, the 4-day requirement does not affect the achievable start time for nozzle dam installation. The data are to be incorporated into guidance for the operator to use when planning outage evolutions. Additionally, procedural guidance regarding the earliest time after shutdown for entry to reduced inventory operations is provided in the APR1400 emergency operating guidelines (EOG). Such restrictions are implemented to minimize the consequences of a loss of shutdown cooling event during reduced inventory operations.

The specified time after shutdown is based on the following analytical results:

a. Decay heat versus time after shutdown

b. Assumption of a total loss of shutdown cooling

c. Consequential maximum RCS pressure for Mode 6 and atmospheric pressure

The use of the PZR manway (surge line) as the vent pathway yields an acceptable equilibrium pressure (below SG nozzle dam design pressure limit) at 4 days after shutdown. This capability is further supported by higher nozzle dam abnormal pressure limits than assumed in the analysis. Although this meets utility requirements for refueling scheduling, it is not recommended that the plant enter mid-loop level operations before 4 days. Conforming with procedural guidance, which requires mid-loop operation no earlier than 4 days, adds margin to the relationship between RCS pressure and nozzle dam design pressure. Furthermore, the entry time for mid-loop level reasonably limits the makeup flow required to match boil-off to within several makeup schemes available in the APR1400 design (see Subsection 2.3.3.4). It fixes the time to boil [assuming an initial RCS temperature of to greater TS than 7.5 minutes, which lengthens the time available for loss of SCS mitigation actions.

2.3.3.4 Alternate Inventory Additions and Shutdown Cooling 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 to keep the core covered becomes a first priority. Inventory makeup extends the margin of safety prior to uncovering the core.

Successful coping with a loss of shutdown cooling includes performing the steps outlined in Subsection 2.4. One of the last measures specified in that section includes adding makeup to the RCS to replenish boil-off. This is considered a last resort measure in the APR1400 design due to the multiple success paths available to restore shutdown cooling and the time available to take corrective actions.

However, as required by Generic Letter No. 88-17, sufficient existing equipment is maintained in an operable or available status to mitigate a loss of RCS inventory if core boiling or an uncontrolled and significant inventory loss occurs. Generic Letter No. 88-17 also recommends that the water addition rate

capable of being provided by each of the means should be at least sufficient to keep the core covered. Finally, Generic Letter No. 88-17 states that the path of water addition must provide reasonable assurance that makeup flow does not bypass the RV before exiting any opening in RCS.

For the APR1400 design, at least two available means of adding inventory to the RCS are available whenever the RCS is in a reduced inventory condition. Operating guidance is provided to specify the source of makeup water, the means of providing makeup to the RCS, and the recommended implementation strategy. The guidance designates makeup pathways that provide reasonable assurance that makeup water does not bypass the RV.

The water addition rate capable of being provided should be at least sufficient to makeup for the boil-off rate. This keeps the core covered and provides an adequate degree of protection for loss of shutdown cooling scenarios. With the earliest nozzle dam installation occurring at 4 days after shutdown, the decay heat present would require approximately TS of makeup flow to compensate for boil-off. This value is based on the PZR manway being opened, venting steam to the PZR cubicle. The steaming rate, and therefore the required makeup rate, reduces beyond 4 days after shutdown.

For Mode 5 reduced inventory operations, a shutdown cooling pump, containment spray pumps, or safety injection pumps are used, as described in Subsection 2.4, to provide pumped makeup if SCS is lost. Procedural guidance cautions operators on using a containment spray pump in the same loop as the affected shutdown cooling pump, especially if the shutdown cooling pump has been lost due to air entrainment/pump cavitation. The makeup pump (shutdown cooling, containment spray, or safety injection) are aligned to the IRWST as the preferred source for makeup.

For Modes 5 and 6, the charging system via a charging pump is used to provide pumped makeup if all methods of shutdown cooling and inventory replenishment delineated above is lost. The pump chosen is aligned to the boric acid storage tank (BAST).

If no method of pumped inventory addition is available, a source for inventory addition is via the safety injection tanks (SITs). This is applicable in Modes 5 and 6 and is considered a last resort. This method is implemented only if the SCS is lost along with all other means of supplying water to the RCS, and RCS boiling is occurring. At least available two tanks provide reasonable assurance that more than 98.4 m^3 (26,000 gal) of water are available for discharge.

Use of the shutdown cooling pumps, containment spray pumps, or safety injection pumps provides reasonable assurance that makeup flow does not bypass the core regardless of postulated openings in the RCS. Flow delivery is through the direct vessel injection (DVI) nozzles. Use of the charging pump also provides reasonable assurance that the makeup flow does not bypass the core since it is injected via the cold leg and enters the core via the normal charging path.

2.3.3.5 Operations

Procedural guidance for the conduct of mid-loop drain downs is provided to provide reasonable assurance that no testing or maintenance activity adversely affects the nuclear steam supply system (NSSS) during mid-loop operations. Guidance is provided to provide reasonable assurance that testing and maintenance activities performed during reduced inventory avoid operations that deliberately lead to perturbations in the RCS and all supporting systems necessary to maintain the RCS in a stable condition. These operations include but are not limited to:

a. RCS drain operations

b. Shutdown cooling testing and maintenance activities

c. Reactor coolant gas vent system testing and maintenance

d. CCW testing and maintenance

e. Withdrawal of the in-core instrumentation for refueling

f. Safety injection system testing and maintenance

g. Personnel communications system perturbations

h. In-core instrument seal table evolutions while the RV head is on and mid-loop operations are in progress.

Avoiding RCS and support system perturbations provides reasonable assurance that adequate operating, operable, and/or available equipment of high reliability is provided for cooling the RCS and for avoiding a loss of RCS cooling. These actions also maintain sufficient existing equipment in an operable or available status to mitigate a loss of SCS or a loss of RCS inventory if either occur. Adequate communications are essential to activities related to the RCS or systems necessary to maintain the RCS in a stable, controlled condition.

Due to the Diablo Canyon incident and other industry events, the requirements for evacuating personnel from the containment building, closing of the containment building equipment hatch and containment air lock doors, and isolating penetrations leading outside containment were evaluated 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 Subsection 3.5.

2.3.4 Resolution

The resolution of the reduced inventory operations issue on APR1400 is complete. Resolution consists of the results of the analyses outlined above, related evaluations in Subsection 2.4 on the availability of shutdown cooling, the Technical Specifications modifications described in Subsection 2.2, and operational guidance described in Subsection 2.1.

The APR1400 design reflects a comprehensive consideration of shutdown and low power risk by adequately addressing all Generic Letter No. 88-17 recommendations and other issues relevant to reduced inventory operations.

TS

Figure 2.3-1 Hot Leg Pressure Variation

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 DHR. 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 DHR. The following three dominant design objectives have evolved from the emphasis placed on prevention of shutdown events:

a. Provide redundant SCS capacity and identify alternate DHR capability

b. Provide instrumentation to effectively monitor shutdown operations, including critical plant configurations such as mid-loop operation

c. Provide flexible redundancy in alternating current (ac) power

The APR1400 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 APR1400 to events initiated from Modes 2 through 6, two criteria related to the potential for radiological release are adopted. Significant release can only occur from fuel cladding rupture resulting from heat-up after the coolant level drops below the top of the active core. Therefore, the first acceptance criterion is that there is no fuel cladding failure resulting from postulated events, excluding a loss-of-coolant accident (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 DHR is limited to a small fraction of 10 CFR 50.34 limits, which are specified in applicable subsections.

2.4.3 Description

In this subsection, an evaluation is presented of the APR1400 features that are designed to prevent violation of the above acceptance criteria. Subsection 2.4.3.1 examines events and event initiators that could result in the loss of shutdown cooling leading to boiling. Causes of the past events that are considered include mid-loop operation, power failure, and operator error. Appropriate procedural guidance and Technical Specification limitations are identified by the analyses and are provided in Subsections 2.1 and 2.2, respectively.

Subsection 2.4.3.2 presents the features of APR1400 that help prevent a loss of DHR due to the loss of ac power. This is one of the concerns identified in NUREG-1410 (Reference 10). This subsection describes the means of coping with a loss of DHR. This concern is also identified in NUREG-1410 and is evaluated in Subsection 2.4.3.2.

Subsection 2.4.3.3 presents the features of APR1400 that help provide reasonable assurance of the availability of the diesel generator (DG), an issue that is also identified in NUREG-1410. Availability of the DG has been a significant factor in many past events.

Taken together, these subsections demonstrate the integrated capability of the APR1400 to prevent and mitigate a loss of DHR to provide reasonable assurance that the acceptance criteria are not violated.

2.4.3.1 Shutdown Event Initiation and Analyses

2.4.3.1.1 Introduction

This subsection examines events that could result in a loss of the SCS due to various initiators (i.e., events that challenge the SCS such as loss of power, inadvertent closure of a valve in the pump suction line, and air ingestion in the pump suction) under various plant configurations and modes of operation and the ways these events can be prevented, detected, and mitigated.

The examination is divided into three parts. The first part (Subsection 2.4.3.1.1) focuses on design features that improve the SCS's resistance to initiators with an emphasis on hardware design. The second part (Subsection 2.4.3.1.2) assumes a loss of the SCS, regardless of the initiator, and describes the ability of APR1400 to recover from the event with an emphasis again primarily on hardware design. The third part (Subsection 2.4.3.1.3) recognizes the limitations of hardware design as a response to initiators and the need to demonstrate that adequate redundancy is provided to cover all possible plant configurations. This includes the plant's ability to cope with a loss of shutdown cooling. The emphasis is on operator actions, operating procedures, and Technical Specifications in the context of the various plant configurations that can exist in Modes 4, 5, and 6.

2.4.3.1.2 Resistance to Initiators

Design improvements have been made to the APR1400 SCS that reduce the likelihood of a loss of shutdown cooling. The improvements are, in part, the result of applying a "beyond single failure criteria" design philosophy to improve the SCS's ability to withstand a wide range of initiators, including a loss of power, equipment failure, control system failure, and operator error. The major design features attributed to the SCS's increased resistance are summarized in Table 2.4-1. The SCS piping and instrumentation diagrams (P&IDs) are shown in Chapter 6 of the APR1400 DCD Tier 2.

The most important feature that was added to the APR1400 SCS is the dedication of the SCS to the shutdown cooling function. No portions of the SCS are included in the emergency core cooling system (ECCS) as was done in past designs. This means that SCS components are not required to be aligned for automatic initiation of safety injection in Modes 1, 2, and 3. This single design change allows for various SCS improvements and simplifications, including the ability to take an SCS division out of service for maintenance for an extended period during normal plant operation in Modes 1, 2, and 3.

The system is composed of two identical, redundant, and totally separate trains, each capable of performing the required shutdown cooling function. Dedicated heat exchangers have been provided in each SCS train.

Each SCS train has independent suction lines from the RCS hot legs. There are no cross-connections between SCS trains. Direct vessel injection (DVI) introduced in the APR1400 and the dedicated shutdown cooling function of the SCS have enabled the trains to be separated. This allows simplification of the arrangements, resulting in greater protection for each pump from suction line failures due to air ingestion and discharge line failures resulting from pump-to-pump interaction.

Open permissive interlocks are provided on the SCS suction isolation valves to prevent these valves from being opened when RCS pressure is above the SCS entry pressure. These interlocks are enabled when RCS pressure is slightly above the shutdown cooling initiation pressure. All auto-closure interlocks to close these valves, such as closure on high pressure, have been eliminated. During an overpressurization transient, the SCS can no longer be automatically isolated from the RCS. The SCS is designed to mitigate these events with low-temperature overpressure protection (LTOP) using spring-loaded relief valves and by an increased SCS design pressure of 63.3 kg/cm^2 (900 psig).

System flow control and protection from pump overspeed have been improved. System flow control is accomplished using valving and fixed resistance orifices in each train. The orifices are sized to limit the maximum flow rate from the SCS pumps, and adjustments to shutdown cooling flow rate to match decay

heat levels are accomplished by modulating valves. This design philosophy not only minimizes seat wear due to high fluid velocities resulting from throttling but also prevents pump excessive flow conditions.

Operating procedures for the SCS during reduced inventory operation provide minimum flow rates necessary to perform shutdown cooling as a function of time after shutdown and to provide adequate boron mixing. SCS flow rate is decreased as lower decay heat levels decrease during mid-loop operation. The lower shutdown cooling flow rate increases the net positive suction head available (NPSHA) to the SCS pumps and provides greater operational margin for the RCS during mid-loop operation.

The inability to accurately measure and provide the RCS fluid levels has been the cause of many incidents resulting in the loss of shutdown cooling. APR1400 has made many improvements in the instrumentation for measuring the liquid level in the RCS and data display available to the operator in the control room. Further information on this topic is provided in Subsections 2.3 and 2.8.

Improvements have been made in the SCS instrumentation to provide the operator with more information about critical points in the system. The intent is to provide the operator with detailed system parameters so appropriate actions can be taken before the loss of shutdown cooling occurs. If a loss of the SCS occurs, these parameters aid in the correct and timely evaluation of the initiator, thus decreasing SCS recovery time. Major instruments that have been included in the APR1400 design are suction and discharge pressure indicators and SCS pump motor current indication. These instruments are all indicated in the MCR.

Suction piping arrangements have been simplified and improved. Several incidents have been attributed to the presence of loop seals in the suction lines that allow air to collect and lead to the reduction of NPSHA and air binding. APR1400 arrangements for the suction lines do not have loop seals and thereby enhance the ability of the pumps to survive low at NPSHA conditions.

The APR1400 SCS design features presented above and summarized in Table 2.4-1 provide a way to minimize a loss of the SCS. These design features also address initiators that are known to have defeated SCSs in currently operating plants. A summary of these initiators and corresponding SCS design features are provided in Tables 1.4-1.

2.4.3.1.3 Recovery from Initiators

In recognition of the fact that some initiators may defeat the SCS, both SCS trains and two divisions of ac power are operable during Modes 5 and 6, allowing safety injection and containment spray equipment in the redundant division to undergo maintenance activities as necessary.

Table 1.4-1 summarizes the design features incorporated into the APR1400 design to prevent, detect, and mitigate the effects of the past shutdown related events. The detailed listing of all potential initiators is not provided in this subsection. Instead, initiators that result in the loss of shutdown cooling are divided into four groups.

The four groups may aid in constructing diagnostic loss of SCS procedures. The groups relate the initiator to a location in the system with respect to the shutdown cooling (SC) pump. The instrumentation provided for monitoring the pump's performance identifies whether the failure is in the suction line, discharge line, the pump itself, or a power failure. With proper diagnostic information from these groups, the operator can perform appropriate recovery actions to restore shutdown cooling. Table 2.4-2 identifies the groups, some representative initiators in each group, a brief description of the event, and the instrumentation available to detect the event. Subsections 2.4.3.1.2.1 through 2.4.3.1.2.4 examine how shutdown cooling can be recovered using this information, assuming loss of an SCS train.

2.4.3.1.3.1 Group I Initiators

Group I initiators include a failure in the suction side of the SC pump. Suction line initiators are the most common during the mid-loop operation and include air ingestion, inadvertent closure of a valve in the suction line, failure of a relief valve to close, leakage from the system, and procedural errors. The result of any of these initiators is to reduce the NPSHA for the SC pump.

Information provided to the operator in the control room for detecting and diagnosing these events includes various alarms and indicators. The SCS includes an alarm for a low-flow condition during shutdown cooling. This alarm is the initial indication of a possible suction line initiator since its set point is above the onset of cavitation. The alarm used for SCS flow is set to indicate a drop in flow from the design value of 18,926 L/min (5,000 gpm) to approximately 15,709 L/min (4,150 gpm) and especially a low flow rate of 14,383 L/min (3,800 gpm) in mid-loop operation. This, in conjunction with a low suction pressure, fluctuating motor current, and near normal discharge pressure (during the onset of cavitation), confirms a Group I initiator (SC pump suction).

2.4.3.1.3.1.1 Recovery during Mode 5

The equipment available to recover from a Group I initiator depends on the mode of operation and includes the redundant SCS train, one of two containment spray (CS), and two of four safety injection (SI) pumps. Technical Specifications, Subsection 3.8.2, requires two ac sources to be available to each division of class 1E ac power during reduced inventory operations in Modes 5 and 6. See Subsection 2.2 of this report. The containment spray (CS) pumps are identical to the SC pumps and provide a redundant source for shutdown cooling flow during Modes 5 and 6. The SI pumps provide a viable source of shutdown cooling flow in Mode 5 because their capacity matches the reduced decay heat generation rate.

If the redundant SCS train cannot be used to recover from a Group I initiator during Mode 5, the CS pumps can be used to re-establish inventory control and shutdown cooling. The CS pumps can be aligned to take suction from either the RCS hot legs or the in-containment refueling water storage tank (IRWST). During a Group I initiator, however, the CS pumps, which are normally aligned to the IRWST, can be used to re-establish inventory control by injecting IRWST water into the RV through the DVI nozzles. This alignment can also provide shutdown cooling using the SCS heat exchanger. This response requires operator action to open one (normally locked closed) cross-connect valve and the actuation of the CS pump from the control room. Once the event is terminated, either the original SCS train or the redundant train can be activated to resume shutdown cooling.

If the CS pump is not functional, the SI pumps can be used to re-establish inventory control by injecting IRWST water into the RV. Shutdown cooling is then performed by either redundant SCS train after the RCS level is recovered by break flow if the initiator provided an opening in the system. In the extreme, a single SI pump can provide sufficient flow to match boil-off, thereby extending operator response time to identify the initiator and terminate the event.

2.4.3.1.3.1.2 Recovery during Mode 6

The equipment available to recover from a Group I initiator during Mode 6 includes the redundant SCS train and possibly the opposite division’s CS pump.

In this mode, the primary recovery system is the redundant SCS train, but if it is not available, a CS pump may be aligned to take suction from the RCS hot leg and discharge into the DVI nozzles. The success of this action is dependent on the particular Group I initiator since the operable CS pump must use the same RCS suction as the defeated SCS train, and the opposite division’s CS pump may be inoperable due to maintenance.

2.4.3.1.3.2 Group II Initiators

Group II initiators include a failure in the discharge side of the SC pump. Discharge line initiators include inadvertent closure or opening of a valve and the inadvertent actuation or leakage from a relief valve. The result of Group II initiators is to change the SCS system resistance curve. The pump responds in accordance with its characteristic curve. For the closure of a valve in the discharge line, the system resistance increases, resulting in a decrease in shutdown cooling flow and an increase in power consumption with a concurrent increase in discharge head. Pump minimum flow recirculation lines prevent pump operation at shutoff. For the inadvertent opening of a valve, there is a reduction in system resistance that produces an increase in flow and a decrease in power consumption at a lower pump head.

Information provided to the operator in the control room to detect and diagnose these events includes the same alarms and indicators described in connection with Group I initiators (Subsection 2.4.3.1.3.1). The instrumentation critical to identifying a Group II initiator includes the SC pump discharge pressure, flow rate, and motor current indication (see Table 2.4-2).

2.4.3.1.3.2.1 Recovery during Mode 5

The equipment available to recover from a Group II initiator during Mode 5 includes the redundant SCS train, one CS pump, and two SI pumps.

In this mode, the primary recovery system is the redundant SCS train because the potential to lose the other SCS pump is unlikely. For the low probability case in which the redundant SCS train cannot be used, the CS pump may be used to re-establish inventory control and shutdown cooling. If the CS pump is not functional, the SI pumps can be used to re-establish inventory control. Shutdown cooling is then performed by either redundant SCS train after the RCS level has been recovered by break flow.

2.4.3.1.3.2.2 Recovery during Mode 6

The equipment available to recover from a Group II initiator during Mode 6 includes the redundant SCS train and possibly the CS pump. The success of using the CS pump is dependent on the specific Group II initiator since the CS pump shares injection lines with the defeated SCS train.

2.4.3.1.3.3 Group III Initiators

Group III initiators include a mechanical failure of the SC pump. Table 2.4-2 shows examples. Recovery from Group III initiators includes activating the redundant SCS train or aligning the CS pump.

2.4.3.1.3.4 Group IV Initiators

Group IV initiators are due to a loss of ac power. Recovery from Group IV initiators includes automatic actuation of battery power, actuation of the alternate alternating current (AAC) facility or, if the loss of power is local to the train, activating the redundant SCS train or CS pump. The APR1400 design ac power availability is described in Subsection 2.4.3.2.

2.4.3.1.4 Recovery Based on Plant Configuration

Subsection 2.4.3.1.3 provides a general description of the recovery from initiators. This subsection examines the recovery from initiators for several plant configurations and modes. This analysis illustrates the capability of the APR1400 to recover from losses of shutdown cooling and identifies new procedural requirements and Technical Specifications to minimize such losses and facilitate mitigative actions.

Figure 2.4-1 facilitates the identification of several major plant configurations of interest for shutdown risk. The termination points shown in the figure relate to Table 2.4-3, which provides an analysis of the configuration identifying possible initiators, Technical Specifications, alternative support equipment, and systems available to mitigate losses of shutdown cooling and recovery actions from initiators.

2.4.3.1.5 Conclusions

The APR1400 SCS design features provide the necessary redundancy, flexibility, and diversity to reduce the likelihood of losing DHR due to a loss of the SCS. 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 reasonable assurance that a loss of DHR is unlikely.

2.4.3.2 APR1400 AC Power Reliability

2.4.3.2.1 Introduction

This subsection presents the APR1400 features that increase 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 RCS and provides redundancy and flexibility to provide reasonable assurance that re-energizing the Class 1E buses is possible if power is interrupted.

2.4.3.2.2 Description

Electrical power sources need to be carefully managed during shutdown operations to maintain the 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 providing reasonable assurance that an alternate source is available. The potential for a complete loss of DHR 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 are guided by Technical Specifications for shutdown operations and reduced inventory. Technical Specifications are 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 provides 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 APR1400 electrical system design provides the redundancy and flexibility to provide reasonable assurance that 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 kV safety buses with three sources of electrical power. The three sources are:

a. Normal permanent non-safety bus (PNS-bus)

b. Alternate-standby transformer

c. Emergency diesel generator

The normal source (PNS-bus) of power to the safety bus has three sources of electrical power. The three sources are (1) normal – division-related unit auxiliary transformer (UAT) powered from preferred offsite power I through the unit main transformer (UMT), (2) alternate – division-related standby transformer served from preferred offsite power interface II, and (3) backup – 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:

a. Preferred offsite power I

b. Preferred offsite power II

c. Diesel generator

d. 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 maintain other diverse sources of reliable electrical power to the 4.16 kV safety bus.

Along with the electrical system design features, the APR1400 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 provide reasonable assurance that 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 are provided in Subsections 2.1 and 2.2.

2.4.3.2.3 Conclusion

The APR1400 electrical distribution system design features provide the necessary redundancy, flexibility, and diversity to reduce the likelihood of losing DHR 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 reasonable assurance that a loss of the DHR is unlikely.

2.4.3.3 APR1400 Diesel Generator Availability

2.4.3.3.1 Introduction

The availability of the DG and the diesel loading sequencer (DLS) to automatically start and load during shutdown modes of operation is one of the issues identified in NUREG-1410. The availability of the DG instrumentation and control system to provide reliable indications and automatic trip signals for DG 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 DG operational problems are significant to the maintenance of DHR as presented in Subsection 2.4.3.1.

The DG and 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 DG provides signals to start the diesel for emergency operation, applicable protective trips to prevent or limit damage to the DG at all times, and DG status to the control room and to the local control panel. The status includes trip signals (alarms, indications, and recordings), parameter indications, and alarms for abnormal parameters. In addition, controls for starting, stopping, synchronizing, and loading the DG are provided in the control room and at the local control panel.

2.4.3.3.2 Description

The DG and DLS need to maintain a consistent means of operation independent of the plant operation condition. This provides reasonable assurance that the operating staff is not required to learn different operating schemes and therefore reduces potential human error.

The APR1400 DG and DLS provide this simplicity of operation. The DG is the emergency source of power to the Class 1E bus. The DG and the DLS are available for operation during shutdown conditions unless undergoing maintenance. The Class 1E buses are monitored for undervoltage and degraded voltage conditions. If either condition is sensed, the DG is started and the DLS is initiated (see APR1400 DCD Tier 2, Figure 6.3-5). For a loss of power to the Class 1E bus, the response of the DG and DLS is not dependent on plant operational modes. Therefore, the response of the APR1400 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 and eliminates the necessity of two sets of procedures that are dependent on plant operating conditions. It also eliminates extra required training for the operation staff. (Details on the emergency diesel generators are provided in Subsection 8.3.1.1.3.)

The DG I&C system needs to provide reasonable assurance that the DG is protected during all modes of operation. However, certain protective trips need to be bypassed during emergency operation.

The APR1400 DG protection system provides automatic trips to prevent or limit damage to the DG. The protection trips provided during emergency operation are:

a. Engine overspeed

b. Generator differential protection

c. Low-low lube oil pressure

d. Generator voltage – controlled overcurrent

These trips are provided in accordance with Regulatory Guide (RG) 1.9, Position 7. All other trips are bypassed during emergency operation. (See Subsection 8.3.1.1.3 for a description of the trips that are bypassed during emergency operation.) The protection circuitry is dependent on the initiating signal and not on plant operational modes. The sensing of an undervoltage or degraded voltage condition during shutdown causes an automatic DG 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 DG independent of plant operating mode.

The I&C system needs to provide reasonable assurance that the operator is informed of the DG'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 APR1400 control room Human System Interface (HSI) presents the operator with the information and controls necessary to complete any tasks identified in a task analysis and system design. The task analysis and system design for DG operation identifies the parameters, alarms, and controls required to operate the DG. This identified status and control scheme is presented to the Main Control Room (MCR) Operator on the Operation Console. The presentation of this information is accomplished in accordance with a structural and hierarchical format. This formatting provides the operator with parameter displays, alarm status, alarm categorization, and alarm priority. This method of information presentation provides the MCR Operator with the tools necessary to monitor and/or diagnose DG status.

The APR1400 local control panels for the DG provides the plant equipment operator with the same information and controls that are available to the MCR Operator.

2.4.3.3.3 Conclusion

The design features of the APR1400 DG I&C systems provide starting signals for the DG and DLS initiation and protective trip signals for DG emergency operation and provide DG status information to the control room and local control panel, which allows the operator to operate, monitor, and diagnose DG and DLS operation. These features of the APR1400 design enhance the operator’s interface with the emergency equipment and reduce the potential of human error.

2.4.4 Resolution

The issue regarding vulnerability during shutdown modes to a loss of DHR is resolved for APR1400 by the design features for the SCS, instrumentation and controls, electrical power distribution system, new Technical Specifications and procedure guidance described in this report. These features demonstrate the reduced potential for significant radiological releases from fuel cladding failure due to postulated events and radiological releases from a loss of DHR due to loss of SCS events. In particular, features of the SCS and electrical distribution system provide the necessary redundancy, flexibility, and diversity to significantly reduce the likelihood of losing DHR.

Procedural guidance is provided to the combined license (COL) applicant via the EOG to provide reasonable assurance that the charging and boric acid makeup pumps are not be made unavailable at the same time during Modes 5 and 6 reduced inventory to further enhance the capability to provide alternate RCS makeup. See APR1400 DCD Tier 2, Chapter 6, Figures 6.3.2-1A, 1B, and 1C for safety injection piping and instrumentation diagrams.

Table 2.4-1

Summary of APR1400 SCS Design Features that Increase Resistance to Initiators

- SCS nozzle at bottom of hot leg
- Dedicated shutdown cooling function
- Independent suction lines for each train
- NO auto-closure interlocks in suction valves
- NO cross train communication
- Increased system design pressure
- Improved flow control
- Improved protection against pump excessive flow conditions
- Flexibility to reduce flow rates to maximize NPSHA
- Improved RCS level instrumentation at mid-loop operation
- Installation of separated core exit thermocouples for measuring core exit temperature at mid-loop operation
- Instrumentation to indicate incipient pump cavitation
- NO loop seals in suction lines
- Improved ac power reliability

Table 2.4-2 (1 of 2)

SCS Instrumentation

Initiator	Item	Result	Indicators/alarms	
I - Failure in the Suction Line				
Inadvertent Signal Closes Motor Operated Valve	SI-651, 653, 655 or SI-652, 654, 666	Loss of cooling flow	Low flow alarm Fluctuating discharge pressure Current fluctuations Low suction pressure Position indication on valve operators	FI-302A&FI-305B, P-302 & P-305, I-302 & I-305, P-300, P-301
Operator Error in Closing SCS Suction Isolation Valve	SI-106 & SI-107	Loss of cooling flow	Low flow alarm Fluctuating discharge pressure Current fluctuations Low suction pressure Position indication on valve operators	FI-302A&FI-305B, P-302&P-305, I-302 & I-305, P-300 P-301
Low RCS Level Resulting in Vortex Formation And Air Entrainment		Pumps cavitation resulting in loss of cooling flow	Low flow alarm Fluctuating discharge pressure Current fluctuations Low suction pressure RCS level	FI-302A&FI-305B, P-302&P-305, I-302 & I-305, P-300, P-301
II - Failure in the Discharge Line				
Inadvertent Signal Closes Motor Operated Valve	SI-310 & 312, 601 or SI-311 & 313, 600	System resistance increases causing the pump to operate near shutoff	Low flow alarm Fluctuating discharge pressure Current fluctuations Low suction pressure RCS level	FI-302A&FI-305B, P-302 & P-305, P-300 & P-301, I-302 & I-305
Operator Error in Closing SCS Discharge Isolation Valve	SI-579, 578	System resistance increases causing the pump to operate near shutoff	Low flow alarm Fluctuating discharge pressure Current fluctuations Low suction pressure RCS level	FI-302A&FI-305B, P-302 & P-305, P-300 & P-301, I-302 & I-305
Inadvertent Signal Opens Motor Operated Valve	SI-690, 691	System resistance decreases causing increase in pump flow. Also, flow split may cause RCS to heat up as less flow is delivered.	High flow indication Low discharge pressure Suction pressure decreases Increased power consumption Position indication on valve operators	FI-302A&FI-305B, P-302 & P-305, P-300 & P-301, I-302 & I-305

Table 2.4-2 (2 of 2)

Initiator	Item	Result	Indicators/alarms	
II - Failure in the Discharge Line (Cont'd)				
Valves in the IRWST Test Path Not Closed Following Completion of SCS Full Flow Test	SI-315,693 & 301 or SI-314, 688 & 300	Pumps drain the RCS Inventory into the IRWST through the test path. Then lose suction as the fluid in the hot leg drops	Liquid level instrumentation for midloop operation Rapid decrease in RCS pressure Rapid decrease in RCS level Liquid level alarms in the IRWST Temp indication in the IRWST	See section 2.3 L-391A & L-390B T-390A01/02/ 03/04 T-391A01/ 02/03/04 T-392A 01/02/03/04
			After level decreases below midloop Low flow alarm Normal discharge pressure Current fluctuations Low suction pressure	FI-302A&FI-305B, P-302 & P-305, I-302 & I-305, P-300 & P-301
Inadvertent Cross Connect to the Containment Spray System	SI-341, 343	Loss of coolant flow	Low RCS level	
III - Failed Pump				
Shaft Failure		Loss of coolant flow	Low flow alarm No discharge pressure No suction pressure	FI-302A&FI-305B, P-302 & P-305, P-300 & P-301

Table 2.4-3 (1 of 8)

Termination Points

Termination Point 1	
Plant Configuration	Modes 4, 5 or 6
Initiators	Loss of power
Technical Specification Requirements	LCO 3.8.1 ~ 3.8.10
Recovery From Initiators	See Sections 2.4.3.2 and 2.4.3.3.
Termination Point 2	
[DELETED]	
Termination Point 3	
Plant Configuration	Mode 4. IRWST full.
Initiators	Group I - III (for RCS pressure less than 31.6 kg/cm^{2}A (450 psia)), Group IV RCS line break.
Technical Specification Requirements	LCO 3.4.6 Two RCS loops or two SCS trains or any combination of these to be operable. One RCS loop or SCS train to be in operation. LCO 3.5.1 Four SITs operable when PZR pressure is greater than 50.3 kg/cm^{2}A (715 psia). LCO 3.5.3 Two SIS trains operable. LCO 3.5.4 IRWST operable. LCO 3.6.6 Two CSS trains operable
Alternative Support Equipment/Systems	None required.
Recovery From Initiators	Shutdown cooling is provided by sources other than the SCS when the RCS pressure is above 31.6 kg/cm^{2}A (450 psia). During these conditions, the ECCS is operable. The SIS is available by automatic actuation down to RCS pressures of 28.1 kg/cm^{2}A (400 psia) (SIAS cutout pressure) and manual actuation at any time. The CSS is operable throughout Mode 4. Below 31.6 kg/cm^{2}A (450 psia), Group I - IV initiators can be mitigated per Section 2.4.3.1.

Table 2.4-3 (2 of 8)

Termination Point 4	
Plant Configuration	Mode 5 RCS in reduced inventory. Nozzle dams installed. IRWST full.
Initiators	Group I-IV. RCS line break.
Technical Specification Requirements	LCO 3.4.7 One CS pump operable. LCO 3.4.8 Two SCS trains operable. One SCS train operating. LCO 3.4.11 LTOP operable or midloop vent operable. LCO 3.5.3 Two SIS trains operable. LCO 3.8.2 ac power (shutdown)
Alternative Support Equipment/Systems	Pumps Charging pump. Boric acid makeup pump. Tanks Safety injection tanks (SIT) Boric acid storage tank (BAST)
Recovery From Initiators	Regain inventory control SC, CS or SI pumps can be used to inject IRWST water into the RCS to regain water level. If these pumps are not functional inventory control can be established using a charging pump (or alternatively with a boric acid makeup pump) by injecting BAST water into the RCS. Regain shutdown cooling capability Shutdown cooling can be regained by using the redundant SCS train once level is recovered. If the redundant SC pump is not functional shutdown cooling can be established using the CS pump. If the CS pump is not functional, shutdown cooling can be established by feed and bleed using SI pumps and flow through break because IRWST is full of water.
Time To Boil	Approximately [ ] TS

Table 2.4-3 (3 of 8)

Termination Point 5	
Plant Configuration	Mode 5. RCS in reduced inventory. Nozzle dams not installed. RCS closed (mid loop vent or RCP seals). IRWST full.
Initiators	Group I-IV. RCS line break.
Technical Specification Requirements	LCO 3.4.7 One CS pump operable. LCO 3.4.8 Two SCS trains operable. One SCS train operating. LCO 3.4.11 LTOP operable or mid-loop vent operable. LCO 3.5.3 Two SIS trains operable.
Equipment/Systems	Pumps Charging pump. Boric acid makeup pump. Tanks Safety injection tanks (SIT) Boric acid storage tank (BAST) Steam generators
Recovery From Initiators	Regain inventory control SC, CS or SI pumps can be used to inject IRWST water into the RCS to regain water level. If these pumps are not functional inventory control can be established using charging pump (or alternatively with a boric acid makeup pump) by injecting BAST water into the RCS. Regain shutdown cooling capability Shutdown cooling can be regained by using the redundant SCS train once level is recovered. If the redundant SC pump is not functional shutdown cooling can be established using the CS pump. If the CS pump is not functional, shutdown cooling can be established by feed and bleed using SI pumps and flow through break because IRWST is full of water.

Table 2.4-3 (4 of 8)

Termination Point 6	
Plant Configuration	Mode 5. RCS in reduced inventory. Nozzle dams not installed. RCS open (PZR manway) IRWST full.
Initiators	Group I-IV. RCS line break.
Technical Specification Requirements	LCO 3.4.7 One CS pump operable. LCO 3.4.8 Two SCS trains operable. One SCS train operating. LCO 3.4.11 LTOP operable or mid-loop vent operable. LCO 3.5.3 Two SIS trains operable. LCO 3.8.2 ac power (shutdown)
Alternative Support Equipment/Systems	Pumps Charging pump. Boric acid makeup pump. Tanks Safety injection tanks (SIT) Boric acid storage tank (BAST) Steam generators
Recovery From Initiators	Regain inventory control SC, CS, SC or SI pumps can be used to inject IRWST water into the RCS to regain water level. If these pumps are not functional inventory control can be established using charging pump (or alternatively with a boric acid makeup pump) by injecting BAST water into the RCS. SITs can also be used. Regain shutdown cooling capability Shutdown cooling can be regained by using the redundant SCS train. If the redundant SC pump is not operable shutdown cooling can be regained using the CS pump. IF the CS pump is not functional, shutdown cooling can be established feed and bleed using SI pumps and PZR manway.

Table 2.4-3 (5 of 8)

Termination Point 7	
Plant Configuration	Mode 5. RCS not in reduced inventory. Nozzle dams installed. IRWST full.
Initiators	Group I-IV. RCS line break.
Technical Specification Requirements	LCO 3.4.7 One CS pump operable. LCO 3.4.8 Two SCS trains operable. One SCS train operating. LCO 3.4.11 LTOP operable or mid-loop vent operable. LCO 3.5.3 Two SIS trains operable.
Alternative Support Equipment/Systems	Pumps CS pump. SI pump. Charging pump. Boric acid makeup pump. Tanks Safety injection tanks (SIT) Boric acid storage tank (BAST)
Recovery From Initiators	Regain shutdown cooling capability Shutdown cooling can be regained by using the redundant SCS train once level is recovered. If the redundant SC pump is not functional shutdown cooling can be established using the CS pump. If the CS pump is not functional, shutdown cooling can be established by feed and bleed using SI pumps and flow through break because IRWST is full of water.

Table 2.4-3 (6 of 8)

Termination Point 8	
Plant Configuration	Mode 5. RCS water level above reduced inventory. Nozzle dams not installed. IRWST full
Initiators	Group I-IV. RCS line break.
Technical Specification Requirements	LCO 3.4.7 One CS pump operable. LCO 3.4.8 Two SCS trains operable. One SCS train operating. LCO 3.4.11 LTOP operable or mid-loop vent operable. LCO 3.5.3 Two SIS trains operable.
Alternative Support Equipment/Systems	Pumps CS pump. SI pump. Charging pump. Boric acid makeup pump. Tanks Safety injection tanks (SIT) Boric acid storage tank (BAST) Steam generators
Rcovery From Initiators	Regain shutdown cooling capability Shutdown cooling can be regained by using the redundant SCS train once level is recovered. If the redundant SC pump is not functional shutdown cooling can be established using the CS pump. If the CS pump is not functional, shutdown cooling can be established by feed and bleed using SI pumps and flow through break because IRWST is full of water.

Table 2.4-3 (6 of 8)

Termination Point 9	
Plant Configuration	Mode 6 Refueling pool empty IRWST full
Initiators	Group I-IV LOCA
Technical Specification Requirements	LCO 3.5.3 Two SIS trains operable. LCO 3.9.5 Two SCS trains operable. One SCS train operating
Alternative Support Equipment/Systems	One CS pump available.
Recovery From Initiators	Regain shutdown cooling capability Shutdown cooling can be regained by using the redundant SCS train once level is recovered. If the redundant SC pump is not functional shutdown cooling can be established using the CS pump. If the CS pump is not functional, shutdown cooling can be established by feed and bleed using SI pumps and flow through break because IRWST is full of water.
Termination Point 10	
Plant Configuration	Mode 6 Refueling pool filled. Reactor vessel head off. Upper internals in place. IRWST empty.
Initiators	Group I-IV RCS line break.
Technical Specification Requirements	LCO 3.9.4 For high water level, one SCS train operable and in operation.
Alternative Support Equipment/Systems	Instrumentation Refueling Pool water level indication in addition to high and low level alarm. Pumps Charging pumps. Boric acid makeup pump. Tanks Boric acid storage tank (BAST)

Table 2.4-3 (7 of 8)

Termination Point 10 (Cont'd)	
Recovery From Initiators	Regain shutdown cooling capability Shutdown cooling can be regained by using the redundant SCS train. If the redundant SCS pump is not functional, shutdown cooling can be established by either passive or active means as described in section 2.10.3. If shutdown cooling has been defeated due to an inter-system LOCA, shutdown cooling can be regained by matching boil-off using the charging pumps (or alternatively with a boric acid makeup pump) injecting BAST water.
Termination Point 11	
Plant Configuration	Mode 6. Refueling pool filled. Reactor vessel head off. Upper internals removed. IRWST empty.
Initiators	Group I-IV RCS line break.
Technical Specification Requirements	LCO 3.9.4 For high water level, one SCS train operable and in operation.
Alternative Support Equipment/Systems	Instrumentation Refueling Pool water level indication in addition to high and low level alarm. Pumps Charging pumps. Boric acid makeup pumps. Tanks Boric acid storage tank (BAST)
Recovery From Initiators	Regain shutdown cooling capability Shutdown cooling can be regained by using the redundant SCS train. If the redundant SCS pump is not functional, shutdown cooling can be established by feed and bleed. If shutdown cooling has been defeated due to an inter-system LOCA, shutdown cooling can be regained by matching boil-off using the charging pumps (or alternatively with a boric acid makeup pump) injecting BAST water.

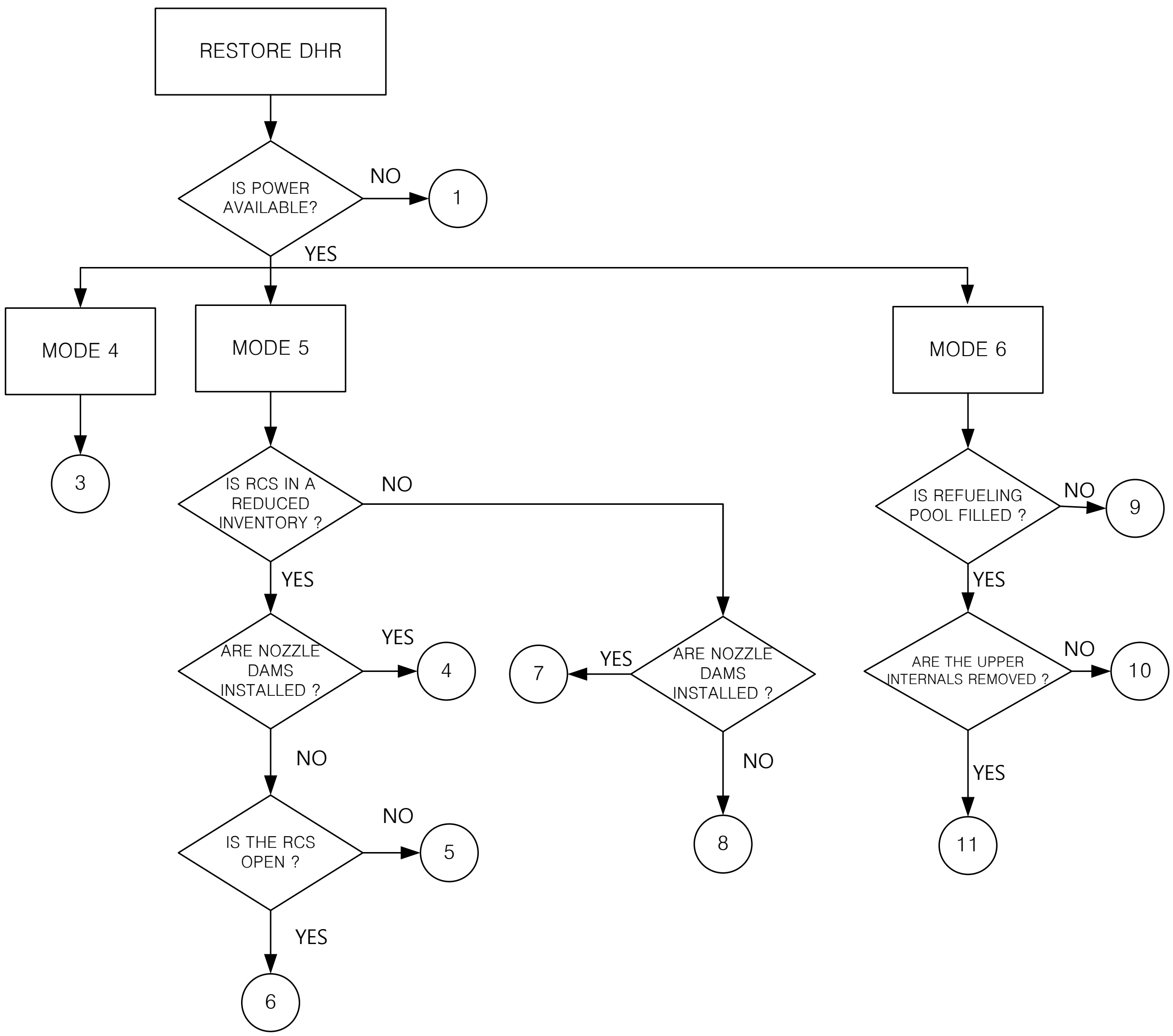

Figure 2.4-1 Plant States and Termination Points for Restoration of DHR

2.5 Containment Capability and Source Term

2.5.1 Issue

This subsection addresses the capability of the containment system to protect the public from the radiation released due to the postulated events that may occur during the shutdown and refueling operation when the time the containment is open.

This issue is focused on the events initiated during Mode 5 or 6 since the containment remains closed during other operation modes. The initiating events that are considered during Modes 5 and 6 are the loss of DHR capability initiated by a loss of shutdown cooling or a loss of coolant caused by operator error or a pipe break.

Following a loss of DHR that is the result of operator error, a radiological release from the RCS to the environment through the open containment can occur when the time for the core coolant to reach saturation temperature is less than the time needed to restore RCS cooling or, if RCS cooling is not restored, the additional time needed to evacuate, close and isolate the containment. The amount of time it takes for the coolant to reach saturation temperature after the event occurs is a function of plant conditions.

The amount of time for restoration of RCS cooling includes the amount of time required to detect that the DHR capability has been lost plus the amount of time to restore shutdown cooling or initiate an alternate means of cooling. The time required to detect loss of DHR depends on the instrumentation available to detect that RCS cooling has been lost. The time required to restore DHR depends on the available systems and emergency operational procedures.

Once the RCS cooling capability has been lost, plant personnel must be evacuated and the containment must be sealed before the RCS begins to boil. The time required to close and isolate the containment depends on:

a. Design features, operational procedures, conditions and status of the equipment to close the containment openings such as containment penetrations, equipment hatches and personnel airlocks

b. Procedures for routing material and lines through these openings

c. Training of plant personnel

d. Environmental conditions including pressure, temperature and radiation within the containment after the core is uncovered

2.5.2 Acceptance Criteria

Radiological acceptance criteria applicable to the shutdown evaluation are based on the limits applied to design basis accidents (DBAs) during full power operation. In addition, the containment temperature is also within the allowable limit such that reasonable assurance is provided that plant personnel have access to the containment to close containment within an acceptable time.

2.5.2.1 Radiological Dose Limits at Site Boundary

Since there is no regulatory guideline on the offsite dose limits for shutdown and low power events, it is appropriate to apply the same radiological acceptance criteria as those applied to the DBAs during full power operation. Therefore, the radiological dose limit to the individual at the Exclusion Area Boundary(EAB) due to an event resulting from a pipe break is set to 250 mSv Total Effective Dose

Equivalent (TEDE), which is applicable to LOCAs at full power as specified in 10 CFR 50.34(a)(1)(ii) (Reference 11).

- TEDE less than 250 mSv for a pipe break during shutdown cooling operation

The radiological acceptance criteria due to an event resulting from a loss of DHR are determined to meet a small fraction of the dose limits in 10 CFR 50.34(a)(1)(ii). A fraction of 10 percent is selected taking into consideration the higher event frequency. Therefore, a 25 mSv TEDE at the EAB for any 2 hours is applied to the loss of DHR during shutdown cooling operation.

- TEDE less than 25 mSv for a loss of DHR during shutdown cooling operation

2.5.2.2 Temperature Considerations

Other than the radiation levels, environmental condition such as temperature or relative humidity influence the allowable working time to close all containment openings. NUREG-1449 describes the allowable environmental condition to protect the personnel health at work inside the containment. It notes an upper temperature limit of 71 °C (160 °F) to avoid burning the lungs.

In the analyses, the temperature of 71 °C (160 °F) is used as the allowable upper limit and the containment temperature should be maintained to be less than this value during the required time to close all containment openings including hatch or personal air locks at an accident.

2.5.2.3 Airborne Radioactivity Concentrations

The emergency operators, who take action to close the containment, should also be protected from radiological exposure from the airborne radioactivity that is released from the RCS to the containment atmosphere. According to the scoping analyses performed by NRC staff in NUREG-1449, the internal dose rates may not be a serious issue if there is no fuel cladding leak and the RCS is cleaned up prior to start of shutdown cooling operation. In addition, the operators would be expected to wear breathing apparatus, and the inhalation doses would be much lower than the estimated values in NUREG-1449. Therefore, the airborne concentrations are not addressed in this analysis.

2.5.3 Description

2.5.3.1 Problem Formulation

The concern of interest is whether the pre-determined radiological dose limits are exceeded due to shutdown and low-power events while the containment is open. The amount of radioactive release depends on factors such as the events that are considered, containment integrity and the Technical Specifications and procedures for closing the containment.

Assumptions used in the analysis are chosen conservatively so the time required to close the containment prior to exceeding the offsite dose limit is minimized. Factors affecting the time to close the containment include radiological and environmental conditions, number and location of closure bolts, provision for loss of ac power, keeping tools needed for closing the equipment hatch near at hand and training and rehearsing personnel in the closure procedure. The analysis focuses on the temperature condition within the containment following the initiation of the event since the other factors are handled by the operational program.

The results of the analyses are used to support recommended changes to Technical Specifications and/or procedures.

2.5.3.2 Containment Integrity

The integrity of the containment is to provide reasonable assurance that the release of any radioactivity does not exceed the limits specified in 10 CFR 50.34(a)(1)(ii). Containment integrity is maintained in accordance with Technical Specifications for Modes 1, 2, 3, 4, and 5 with reduced inventory and for Mode 6 with reduced inventory or core alterations.

In Modes 1, 2, 3, and 4, the containment is required to be operable per Technical Specifications. Integrity exists when the items defined in the Subsection 1.1, Definitions of the Technical Specification are satisfied. Additional Technical Specifications for containment personnel locks and containment isolation valves provide actions and surveillance requirements to provide reasonable assurance that containment integrity is not compromised.

In Mode 5 with the RCS in reduced inventory and Mode 6 during core alteration or reduced inventory, containment integrity is maintained in accordance with Technical Specifications.

2.5.3.2.1 Integrity Requirements

2.5.3.2.1.1 Modes 1 through 4

Maintaining containment integrity in Modes 1 through 4 is accomplished by providing reasonable assurance of conformance with Technical Specifications. Prior to entry into Mode 4 from Mode 5, all surveillance requirements are verified in accordance with the applicable procedures.

2.5.3.2.1.2 Mode 5

Mode 5 is divided into two operational conditions:

a. RCS level above reduced inventory

b. RCS level below reduced inventory

Entry into or out of these operational conditions is controlled by procedures and Technical Specifications and requires verification by the senior reactor operator.

2.5.3.2.1.2.1 Reactor Coolant System Level above Reduced Inventory

There are no Technical Specification requirements on containment integrity in Mode 5 when not in reduced inventory. Therefore, proceeding from Mode 4 to Mode 5 does not require conformance with Technical Specifications dealing with containment integrity. In Mode 5, equipment for maintenance and refueling outages and support personnel are moved into and out of containment through the one equipment hatch and two personnel airlocks. Surveillance testing of containment penetrations is completed in Mode 5 and verified in accordance with site-specific procedures.

2.5.3.2.1.2.2 Reactor Coolant System Level below Reduced Inventory

In Mode 5, the RCS may be drained to facilitate installation of the SG nozzle dams and to accomplish other maintenance items. Draining the RCS to a reduced inventory level (greater than 0.9 m (3 ft) below the reactor flange) requires monitoring for any leakage of radiation through the penetrations (according to the containment penetration Technical Specifications). To maintain containment integrity, the equipment hatch and one of the two doors on each personnel airlock must be closed.

2.5.3.2.1.3 Mode 6

The potential for fuel handling accidents in Mode 6 is the basis of the requirement that containment

integrity be maintained. Thus, the equipment hatch and one of the two doors on each personnel airlock must be closed during core alterations.

Entry from Mode 5 to Mode 6 may require verification of containment penetration status. Since Mode 5 has two operational states, containment configuration must be satisfied and verified by the Senior Reactor Operator. Entry into Mode 6 from Mode 5 for a reduced inventory operation requires monitoring of containment penetrations for radiation leakage. Entry into Mode 6 from Mode 5, not at reduced inventory, requires verification of penetration status.

2.5.3.2.2 APR1400 Containment Features

2.5.3.2.2.1 Building Arrangement and Ventilation

The containment openings are surrounded by the auxiliary building except that the equipment hatch is connected to the roof of the auxiliary building.

The auxiliary building controlled area heating, ventilation, and air conditioning (HVAC) system serves for the suitable temperature, pressure, and radiation conditions for radiologically controlled areas except for the fuel handling area in the auxiliary building. During normal operation conditions, four 100 percent capacity normal exhaust air cleaning units (ACUs) equipped with high-efficiency particulate air (HEPA) and charcoal filters, two per division, operate to maintain the suitable environmental conditions. When high radiation is detected in the exhaust duct upstream of the normal exhaust ACU, the normal exhaust ACU operates the same as system normal in order to maintain the areas under a slightly negative pressure with respect to the surrounding areas to limit release of airborne radioactivity. Upon receipt of an engineered safety features actuation system (ESFAS)-safety injection actuation system (SIAS), the exhaust air is collected and processed through four 100 percent capacity emergency exhaust ACUs, two per division, prior to release to the atmosphere.

The fuel handling area HVAC system is designed to maintain the appropriate indoor environmental conditions and the fuel handling area at a negative pressure relative to the atmosphere to prevent potential radioactive releases. During the normal operation conditions, one 100 percent capacity normal exhaust ACU with HEPA filters is provided. Upon receipt of an ESFAS-fuel handling area emergency ventilation actuation signal (FHAEVAS) or high radiation signal, the air flow is directed to two 100 percent capacity emergency exhaust ACUs equipped with HEPA and charcoal filters to mitigate potential release from the spent fuel pool.

2.5.3.2.2.2 Personnel Airlocks

The personnel airlocks allow passage of the work force into and out of the containment during all modes of operation. The APR1400 has two personnel airlocks, one at El. 103 ft 9 in (Figure 2.5-1) and another at El. 159 ft 9 in (Figure 2.5-2).

Each personnel airlock is a right circular cylinder approximately 3.05 m (10 ft) in diameter with a door at both ends. The airlocks form part of the containment pressure boundary. Therefore, closing and sealing the airlocks prevent leakage of radioactive material.

The design and testing of the personnel airlocks provide reasonable assurance that the airlocks can withstand pressures in excess of the maximum pressure following the DBA inside containment. Closing a single door provides reasonable assurance of containment integrity.

Each door contains double seals and local leakage rate testing capability to provide pressure integrity. To give the effect of a leak tight seal, the personnel airlock design uses pressure-seated doors. Any leakage passes into the Auxiliary Building.

Each personnel airlock is provided with limit switches on both doors that provide control room indication of door position. The doors are interlocked to prevent simultaneous opening, which would compromise containment integrity during Modes 1 through 4.

The normal alignment of the personnel airlocks during the various modes of operation is listed in Table 2.5-1.

In Mode 5 with inventory greater than the reduced level (less than 0.9 m (3 ft) below the flange), both personnel airlocks can be opened only during an outage when it is necessary to transfer equipment into and out of containment. Closure can be initiated by dispatching personnel from the control room if containment integrity needs to be restored. Closure of both doors can be accomplished within 10 minutes.

2.5.3.2.2.3 Equipment Hatch

The containment equipment hatch provides a means for moving large equipment and components into and out of containment. In the APR1400 design, the hatch is 7.92 m (26 ft) in diameter and located at El. 167 ft 6 in (Figure 2.5-2). Normal alignment of the equipment hatch during the modes of operation is listed in Table 2.5-1.

When closed, the hatch is part of the containment pressure boundary. Sealing is by means of a double seal, which is Type B leak rate tested in accordance with 10 CFR 50 (Reference 12), Appendix J, prior to entry into Mode 4.

The equipment hatch is removed following cleanup of containment atmosphere and entry into Mode 5 at full inventory. A manual and power operated system for removing and handling the hatch cover is provided and includes trolleys, tracks, and hoists to raise the equipment hatch vertically, allowing equipment to be transferred in and out of containment without interference. It is designed to minimize hatch movement, thus reducing closure time.

The removing and handling system uses ac power. In the event of the failure of the power source, the system is designed to be operated manually.

Before proceeding to Mode 5 at reduced inventory or Mode 6 with reduced inventory or core alterations, the equipment hatch is closed. With ac power, closure time is less than 1 hour.

After being set in place, the hatch is bolted. Technical Specifications require all bolts to be in place and tightened in Modes 1 through 4. In Mode 5 with reduced inventory and in Mode 6, Technical Specifications require that four bolts be in place and tightened. This minimum number of bolts is sufficient to secure the hatch so that no visible gap can be seen between the seals and sealing surface.

The hatch is designed to be pressure seated. Thus, any increase in pressure inside the containment acts to seal the hatch. In addition, any radiation leakage is into the auxiliary building.

2.5.3.2.2.4 Penetrations

The APR1400 DCD Tier 2, Table 6.2.4-1, identifies the fluid system penetrations in the containment vessel. Each penetration is provided with a means of isolation by the Containment Isolation System (see APR1400 DCD Tier 2, Subsection 6.2.4).

Procedures to meet Technical Specification surveillance requirements are provided for maintaining proper valve alignment to provide reasonable assurance of containment integrity prior to entry into Mode 4, Mode 5 (at reduced inventory), or Mode 6 (reduced inventory or core alterations). In Mode 5 with the RCS level greater than reduced inventory, these penetrations are leak tested in accordance with 10 CFR 50 (Reference 11), Appendix J. Misalignment of the valves can result in leakage paths limited by size of

these normally small diameter valves (less than 2 cm (0.75 in)).

2.5.3.3 Events Analyzed

The events that are analyzed are considered to evaluate the APR1400 containment's capability to protect the public from the consequences of radiation release when the containment is open.

Since the equipment hatch is closed in all modes except Mode 5 with full inventory as presented in Table 2.5-1, the radiological consequences for all events in these modes are expected to be bounded by the analysis results performed in Mode 1. In Mode 5 with full inventory, however, the equipment hatch is assumed to be open, and the radiological consequence analysis is performed to demonstrate conformance with the acceptance criteria in 10 CFR 50.34(a)(1)(ii) (Reference 11).

A qualitative evaluation for the postulated events which can occur at Modes 5 and 6, has been performed and the results are presented Table 2.5-2. The events considered include the DBAs addressed in Chapter 15 of the APR1400 DCD Tier 2 and several other events that can be postulated during shutdown cooling and refueling operation. As addressed in Table 2.5-2, the radiological consequences for the most of the events such as main steam line break, RCP locked rotor, control element assembly (CEA) ejection, letdown line break, and SG tube rupture are evaluated to be bound by those for the same events that occur during Mode 1. For the other events (e.g., fuel handling accident, loss of shutdown cooling, inventory boil-off during refueling) would not cause any significant impact to the environment since these events can occur only when the containment is closed. As a result, the only event that needs a quantitative radiological consequence analysis is Mode 5 LOCA at full inventory.

In the analysis for LOCA at full inventory, the time-dependent profiles of radiation dose at the EAB and temperature inside containment are calculated from the time the pipe break occurs in the reactor coolant pressure boundary. Then, the minimum time required to close the equipment hatch is the time from the detection of the event and the time to exceed the dose limit. It is conservatively assumed that the equipment hatch and two personnel airlocks are fully open. However, the in-containment thermodynamic analyses conservatively assume that the size of the opening areas are a half of the size used in the radiological analyses. The thermodynamic and radiological analyses are performed assuming that the equipment hatch and personnel airlocks remain open throughout the calculation. Table 2.5-3 presents the modeling of the opening area applied to each analysis case.

2.5.3.4 Analysis

Calculations are performed to predict pressure, temperature within the containment and offsite dose at EAB for the Mode 5 LOCA at full inventory case as described above. The three steps that are followed to perform the calculations are described in Subsection 2.5.3.4.1 through 2.5.3.4.3.

2.5.3.4.1 Event Sequence

For a LOCA initiated from Mode 5 with full inventory, a break in a DVI line of 0.0372 m^2 (0.4 ft^2) is considered. The entry time into mode 5 operation was used 3.45 hour after shutdown based on the maximum cooldown rate of 55.6 °C/hr (100 °F/hr). The initial condition of RCS pressure and temperature was selected at 28.12 kgf/cm^2A (400 psia) and 99 °C (210 °F) respectively. Mass and energy release data for Mode 5 LOCA at full inventory are divided into two cases depending on the number of SI pumps that are assumed to operate after the LOCA; one for one SI pump operation and the other for two SI pumps operation. The SI pump injection was assumed to begin 32 minutes after the LOCA manually by the operator with a typical symptom of LOCA. The SIT operation is not available in this analysis. The time history mass-energy releases for these cases are listed in Table 2.5-4.

2.5.3.4.2 Thermodynamic Conditions

The GOTHIC computer code is used to predict pressure, temperature versus time for a given rate of mass, and energy flow into the containment. The model includes provisions for varying, as a function of time, flow areas open to the ambient.

The containment model for the analysis assumes:

a. At the break point, the discharged fluid is instantaneously mixed into thermal equilibrium with the components of vapor region.

b. The liquid region pressure is the same as containment atmosphere pressure. Temperature difference may exist between both regions.

c. The mass and energy transfer such as evaporation, condensation, and boiling between the vapor and droplet region are only assumed. The heat transfer between vapor and liquid region is neglected.

GOTHIC Diffuse Layer Model (DLM) heat transfer coefficient is used for condensing heat transfer to the containment passive heat sinks.

The code models an IRWST for the accumulation of condensed water. The heat transfer between water and vapor in the containment and IRWST is assumed to be neglected to minimize condensation to the subcooled water.

Based on values for the open areas, the analysis predicts thermodynamic conditions inside the containment versus time for the mass flow rate and enthalpy release for the Modes 5 LOCA event. The following nominal initial conditions are assumed in the calculations;

- Containment volume : 8.858 x 104 m3 (3.128 x 106 ft3) (excluding the IRWST)
- Initial pressure : 1.03 kg/cm2 (14.7 psia)
- Initial temperature : 37.78 °C (100 °F)
- Initial relative humidity : 50 percent

The containment thermodynamic conditions are functions of size of the open area of the containment penetrations, such as equipment hatch and personnel airlocks, and how long they remain open. In the containment temperature analysis, it is conservatively assumed that containment penetrations have smaller size of the open area than the value used for the radiological analysis. Modeling of the containment openings is presented in Table 2.5-3.

Heat losses due to active heat sinks (fan coolers) are ignored. Modeling of the passive heat sinks is selectively taken into account for the two analyses as described in Table 2.5-5.

2.5.3.4.3 Radiological Releases

The radiological consequence analyses for Mode 5 LOCA with full inventory are performed to determine the offsite doses to the public located at EAB using the guidance in RG 1.183 (Reference 13) and the APR1400 plant-specific design inputs for the following release paths:

- Containment leakage
- Engineered safety feature (ESF) leakage

2.5.3.4.3.1 Evaluation Model

The following transport models of radioactive materials are applied to evaluate radiological consequences due to a Mode 5 LOCA as shown in Figure 2.5-3.

2.5.3.4.3.1.1 Containment Leakage

Since the safety injection pumps are assumed to be manually actuated at 32 minutes after the reactor trip by a pipe break, no fuel damage is postulated for the Mode 5 LOCA with full inventory.

The RCS fluid is released into the containment through the broken pipe and assumed to be directly released to the environment through the opening areas of equipment and personnel airlocks without mixing with the containment air for conservatism. All of the radionuclides released into the containment are assumed to flash immediately to vapor and be released to environment without any reduction or mitigation.

Since the safety injection pumps take suction from the IRWST, all radionuclides in the IRWST are also taken into account in the analysis.

2.5.3.4.3.1.2 Engineered Safety Feature System Leakage

The ESF systems, located outside of containment, that recirculate IRWST water are assumed to leak during their intended operation. This release source includes leakage through valve packing glands, pump shaft seals, flanged connections, and other similar components.

All of the radionuclides except for noble gases released into the containment through the pipe break are assumed to instantaneously and homogeneously mix in the IRWST water.

The ESF leakage is assumed to be retained on the floor of the equipment compartment in the auxiliary building, and a portion of the radioactive materials in the ESF leakage flashes and becomes airborne in the auxiliary building and then released to the environment through the auxiliary building ventilation exhaust system. No credit is taken for filtration by the auxiliary building controlled area emergency exhaust ACU.

2.5.3.4.3.2 Input Parameters and Initial Conditions

The radiological consequences of the Mode 5 LOCA are analyzed using a conservative set of assumptions and the APR1400 design inputs. Input parameters values used in the analysis are presented in Table 2.5-6.

The RCS isotopic iodine concentrations are based on the Technical Specification for RCS equilibrium activity, and the noble gas concentrations are based on 1 percent fuel defect. Consistent with RG 1.183, iodine spiking is not considered.

The RCS mass release rates shown in Table 2.5-6 are determined to envelop the two cases in Table 2.5-4 because there is no difference in the resultant radiological consequences for the two cases.

The flashing fraction is determined based on the enthalpy difference under the circumstance of coolant leakage by assuming the leakage to be a constant enthalpy process. For all of the radionuclides released into the containment through the pipe break, the flashing fraction is conservatively assumed to be 1. For ESF leakage, the IRWST water is conservatively assumed to remain at less than 100 °C (212 °F), and the flashing fraction of 10 percent is used to be consistent with RG 1.183.

The 2-hour χ/Q value of 1.0E-03 sec/m^3 is used to calculate the potential offsite dose at the EAB.

2.5.3.4.4 Results

The thermodynamic and radiological analyses were performed assuming that the equipment hatch and personnel airlock areas remain closed or open throughout the progress of the event sequences.

The time to close the containment is determined to meet the dose limit of 250 mSv TEDE at the EAB, as described in Subsection 2.5.2.1. The temperature inside the containment for personnel to conduct containment closure is limited to 71 °C (160 °F), as described in Subsection 2.5.2.2.

2.5.3.4.4.1 Offsite Dose at EAB

The radiological consequences due to a Mode 5 LOCA are presented in Table 2.5-7 and Figure 2.5-4. The results show that the offsite dose is 150 mSv which is within the dose limit of 250 mSv. As shown in Figure 2.5-4, the minimum closure time is greater than 2 hours.

2.5.3.4.4.2 Containment Temperature

The transient behaviors of containment temperatures are shown on Figure 2.5-5. For the case that 2 SIPs are in operation, the containment temperature is always maintained at less than 71 °C (160 °F). Although, for the case of 1 SIP operation, the containment temperature increases with time as the mass and energy release to the containment continues, the temperature is maintained at less than 71 °C (160 °F) during approximately more than 70 minutes after accident initiation. This satisfies the requirement that the containment temperature be maintained at less than 71 °C (160 °F) during the time required to close the equipment hatch.

2.5.4 Resolution

Analyses have been performed to provide reasonable assurance that the public radiation dose limit is not exceeded during the shutdown and low power operation modes. Through a qualitative evaluation on a variety of DBEs, a Mode 5 LOCA with full inventory was selected as the representative event to be quantitatively analyzed. Following a set of thermal-hydraulic analyses and radiological consequence analysis, it is concluded that the offsite dose at the EAB conforms with the acceptance limits of 250 mSv. In addition, the containment temperature is maintained less than the upper temperature limit of 71 °C (160 °F) for sufficient time to close the equipment hatch.

Table 2.5-1

Containment Openings

Opening	Area m^2 (ft^2)	Normal Status			
		Modes 1 - 4	Mode 5 Full Inventory	Mode 5 Reduced Inventory	Mode 6 Reduced Inventory
Equipment hatch	49.32 (530.93)	Closed	Open	Closed	Closed
Personnel airlocks (2)	7.30 (78.53)/Airlock	Closed	Open	1 door closed per airlock	1 door closed per airlock

Table 2.5-2 (1 of 2)

Qualitative Evaluation for Anticipated Events during Modes 5 and 6

Mode	Event	Containment Penetration Status	Review
5	Main Stem Line Break	Open or Closed	As addressed in APR1400 DCD Tier 2, Subsection 15.1.5 (MSLB outside the containment during Mode 1), all of the radioactive materials in the RCS are released to the environment through SG tube leaks. The allowable dose limit of 250 mSv is met. When the equipment hatch is open in Mode 5 and the event of the break of steam line inside the containment occurs, the result is expected to be same as that for Mode 1. Therefore, the quantitative evaluation is not needed. The impact of the primary coolant discharged to the IRWST through POSRV is expected to be neglected.
5	RCP Rotor Seizure	Open or Closed	According to APR1400 DCD Tier 2, Subsection 15.3.3, the radioactive materials are released to the environment via the MSSVs or ADVs. Since the Mode 5 event is anticipated to experience the same discharge path, the release amount due to the opening of the equipment hatch is not expected to increase the release.
5	CEA Ejection	Open or Closed	Since the pressure in RCS is not high enough to cause a CEA ejection event, the quantitative evaluation is not needed.
5	Letdown Line Break	Open or Closed	According to APR1400 DCD Tier 2, Subsection 15.6.2 (Letdown line outside the containment upstream of the letdown isolation valve during Mode 1), all of the vaporized radioactive materials in the RCS are released to the environment and the allowable dose criterion is satisfied. When the equipment hatch is open in Mode 5 and the event of the letdown line break is assumed to occur inside the containment, the result is expected to be bounded by the results of the Mode 1 event. Therefore, the quantitative evaluation is not needed.

Table 2.5-2 (2 of 2)

Mode	Event	Containment Penetration Status	Review
5	Steam Generator Tube Rupture	Open or Closed	According to APR1400 DCD Tier 2, Subsection 15.6.3, the radioactive materials are released to the environment via the MSSVs or ADVs or condenser. Since the Mode 5 event is anticipated to experience the same discharge path, the amount of release due to the opening of the equipment hatch is not expected to increase.
5	LOCA	Open	As a result of a DVI line break, it is estimated that all of the radioactive substances in the RCS are released to the environment through the opening of the equipment hatch. Radiological consequence analysis for this event has not been performed and is therefore not addressed in Chapter 15 of the APR1400 DCD Tier 2.
6	Fuel Handling Accident	Closed	Since the containment penetrations are closed, the quantitative evaluation is not needed.
5	Loss of Shutdown Cooling	Closed	Since the containment penetrations are closed, the quantitative evaluation is not needed.
6	Inventory Boil-off during Refueling	Closed	Since the containment penetrations are closed, the quantitative evaluation is not needed.

Table 2.5-3

Modeling of Containment Openings

Case	Containment Temperature Analysis	Radiological Analysis
Mode 5 LOCA	Equipment hatch : 1/2 open Two personal airlocks : 1/2 open	Equipment hatch : open Two personal airlocks : open

Table 2.5-4 (1 of 2)

Mass and Energy Release for Mode 5 LOCA at Full Inventory

Time (sec)	Safety Injection Pump 1		Safety Injection Pump 2	
	Mass flow (kg/sec)	Enthalpy (kJ/kg)	Mass flow (kg/sec)	Enthalpy (kJ/kg)
1.6	1189.11	414.46	1189.11	414.46
3.0	995.85	413.99	995.85	413.99
4.6	923.42	413.84	923.42	413.84
6.2	750.77	413.62	750.77	413.62
7.8	542.53	413.51	542.53	413.51
9.4	540.59	413.80	540.59	413.80
11.0	531.07	414.21	531.07	414.21
12.6	537.49	414.75	537.49	414.75
14.2	535.78	415.37	535.74	415.37
15.8	533.24	416.06	533.24	416.06
19.0	532.55	417.48	532.53	417.48
28.6	526.28	420.83	526.23	420.84
39.8	515.33	423.64	515.58	423.65
49.4	512.21	425.84	512.25	425.86
59.0	495.73	427.97	495.75	428.00
68.6	484.13	430.02	484.26	430.04
79.8	463.82	432.20	463.77	432.21
89.4	430.01	433.95	429.78	433.96
99.0	410.28	435.67	410.03	435.67
199.8	310.71	432.28	309.56	432.75
299.0	246.17	436.10	264.19	435.15

Table 2.5-4 (2 of 2)

Time (sec)	Safety Injection Pump 1		Safety Injection Pump 2	
	Mass flow (kg/sec)	Enthalpy (kJ/kg)	Mass flow (kg/sec)	Enthalpy (kJ/kg)
399.8	212.98	437.72	212.92	437.37
499.0	192.44	450.65	197.39	449.56
599.8	198.65	463.47	181.14	462.60
699.0	188.06	475.57	215.95	471.72
799.8	215.66	481.85	218.52	482.89
899.0	218.20	490.81	216.12	491.68
999.8	167.52	587.83	157.01	597.21
1099.0	13.63	2687.38	13.44	2702.55
1199.8	14.09	2699.22	13.99	2706.05
1299.0	14.39	2701.85	14.27	2702.06
1399.8	14.74	2680.19	14.76	2693.53
1499.0	14.91	2698.07	15.00	2700.65
1599.8	14.96	2701.16	15.03	2676.04
1699.0	15.17	2709.94	15.23	2687.04
1799.8	15.50	2682.10	15.41	2701.06
1899.0	15.61	2698.65	15.55	2718.46
2000.0	14.69	2621.36	13.88	2584.57
2499.0	84.93	761.91	141.50	524.56
2999.8	50.74	964.11	133.59	470.57
3499	11.81	2681.24	178.01	456.22
3999.8	45.80	980.98	127.39	456.63
4499	66.80	789.42	155.67	487.64
4999.8	66.64	779.57	123.50	474.45
5499	79.14	716.01	75.22	554.99
5999.8	56.38	833.18	66.26	585.12
6499	81.35	695.39	158.25	436.09

Table 2.5-5

Modeling of Passive Heat Sink

Case	Containment Temperature Analysis	Radiological Analysis
Mode 5 LOCA	Modeled	Not modeled

Table 2.5-6

Major Input Parameters Used in Radiological Consequences Analysis for Mode 5 LOCA

Parameter Category	Parameter	Value
Source Terms	Reactor core power level	4,062.66 MWt
	Percentage of fuel assumed to experience DNB	0 percent
	Percentage of fuel assumed to melt	0 percent
	Initial RCS mass	517,154 kg (1,140,129 lbm)
	Initial RCS iodine specific activity	3.7×10^{4} Bq/g (1.0 μCi/g) DE I-131
	Initial RCS noble gas specific activity	2.15×10^{7} Bq/g (580 μCi/g) DE Xe-133
	Initial RCS others specific activity	RCS concentrations based on 1.0 percent fuel defect
	Initial IRWST specific activity	IRWST concentrations based on 1.0 percent fuel defect
	Primary coolant density	1.0 g/cm^3 (62.4 lbm/ft^3)
Containment Leakage Transport Model	Iodine chemical form	
	Aerosol (CsI) Elemental Organic	95.0 percent 4.85 percent 0.15 percent
	Enveloped RCS mass release rates	
	0.0 ~ 3.0 sec 3.0 ~ 6.2 sec 6.2 ~ 7.8 sec 7.8 ~ 59.0 sec 59.0 ~ 199.8 sec 199.8 ~ 299.0 sec 299.0 ~ 999.8 sec 999.8 ~ 1099.0 sec 1099.0 ~ 2499.0 sec 2499.0 sec ~	1200.0 kg/sec 1000.0 kg/sec 800.0 kg/sec 600.0 kg/sec 500.0 kg/sec 400.0 kg/sec 300.0 kg/sec 200.0 kg/sec 100.0 kg/sec 200.0 kg/sec
	RCS leak flashing fraction	1
	Credit for radioactive decay during Hold up in containment In transit to dose points	 Not applicable Not applicable
	Containment aerosol natural deposition removal	Not applicable
ESF Leakage Transport Model	Minimum IRWST water volume	2.44×10^{3} m^3 (8.61×10^{4} ft^3)
	Chemical form of iodine in ESF Elemental Organic	 97 percent 3 percent
	ESF leakage rate into auxiliary building	3.79×10^{-2} m^3/hr (1.34 ft^3/hr)
	ESF leakage flashing fraction	0.1
	ESF leakage initiation time	0.0 min
	Auxiliary building controlled area exhaust ACU filter efficiencies Elemental and organic iodine Particulate	 0 percent 0 percent

Table 2.5-7

Radiological Consequences of Mode 5 LOCA at Full Inventory

Release Path	TEDE Dose (mSv) at EAB
Containment leakage	150.41
ESF leakage	0.0002
Total	150.41
Allowable TEDE limit	250.00

TS

Figure 2.5-1 Personnel Airlock at El. 100'

TS

Figure 2.5-2 Personnel Airlock and Equipment Hatch at El. 156'

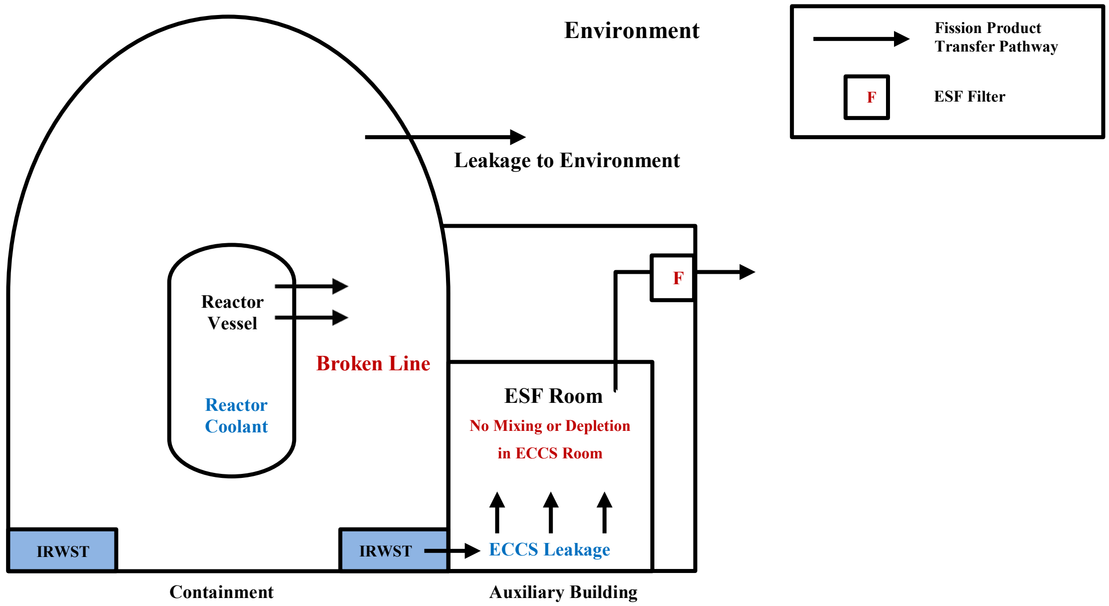

Figure 2.5-3 Radioactivity Transport Model for Mode 5 LOCA

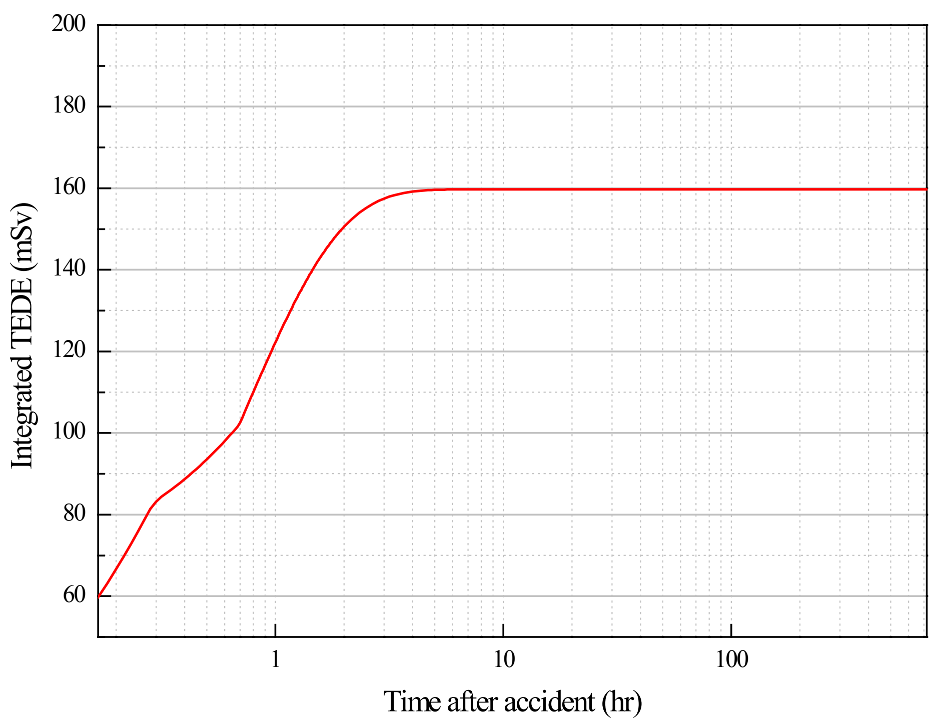

Figure 2.5-4 Integrated TEDE in the Event of a Mode 5 LOCA

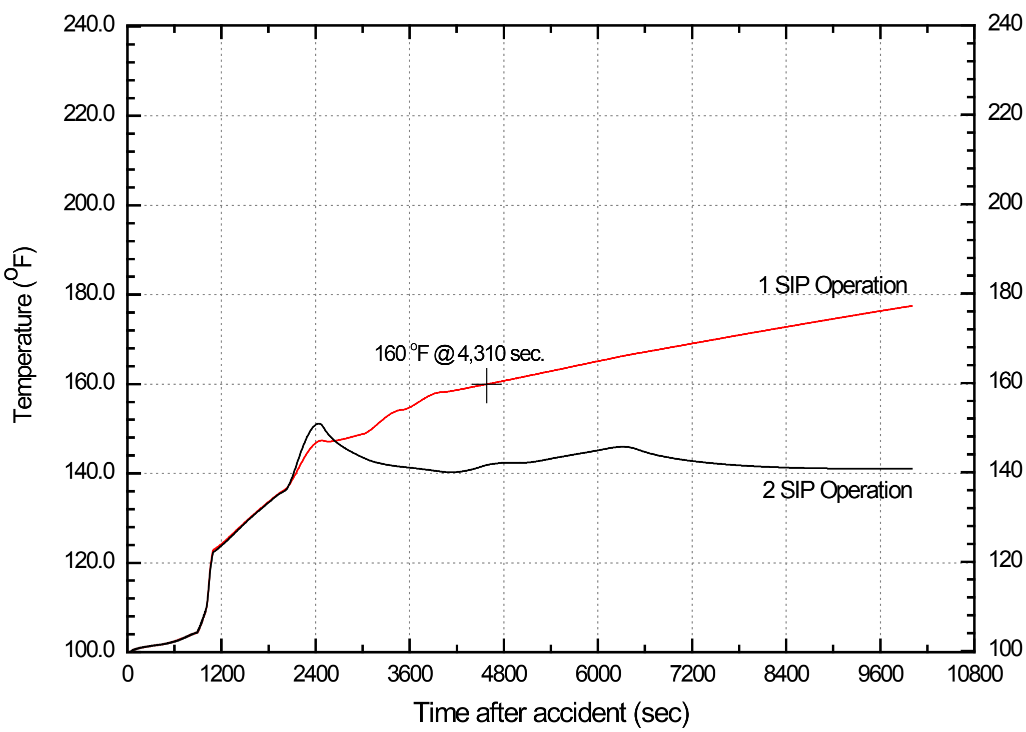

Figure 2.5-5 Containment Temperature in the Event of a Mode 5 LOCA

2.6 Rapid Boron Dilution

2.6.1 Issues

The issues related to rapid boron dilution are divided into the following three categories:

a. The introduction of deborated water into the RCS via the shutdown cooling system (SCS), which flows into the RCS through the direct vessel injection (DVI) lines, during maintenance of inline components.

b. Introduction of a water slug into the RCS during startup or refueling operations, including an example from NUREG-1449. In that example, a LOOP has occurred, and the charging pumps are returned online, powered by the emergency diesel generators (EDGs). If the plant is 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 RV. If it is then assumed that offsite power is restored and the RCPs are restarted, then a water slug of deborated water can be injected into the core.

c. A potential boron dilution resulting from inleakage from the secondary side of an SG during an SG tube rupture (SGTR) event.

These issues are addressed in Subsections 2.6.3 and 2.6.4.

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 General Design Criteria (GDC) 10, 15, and 26. The criteria are as follows:

a. Pressure in the reactor coolant and main steam systems are maintained below the RCS pressure-temperature (P-T) limits (see the limits in “Pressure-Temperature Limits Methodology for Reactor Coolant System Heatup and Cooldown,” Reference 14) or below 110 percent of the design value, whichever is less.

b. Fuel cladding integrity is maintained by providing reasonable assurance that the minimum departure from nucleate boiling ratio (DNBR) remains above the 95/95 DNBR limit for PWRs based on acceptable correlations (see NUREG-0800, Subsection 4.4; Reference 15).

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

d. An incident of moderate frequency in combination with any single active component failure, or single operator error, is considered and is an event for which an estimate of the number of potential fuel failures is provided for radiological dose calculations. For such accidents, the number of fuel failures are assumed for all rods for which the DNBR or critical power ratio falls below those values cited above for cladding integrity unless it can be shown, based on an acceptable fuel damage model (see NUREG-0800, Subsection 4.2), that fewer failures occur. There is 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, Subsection 15.4.6, with the exception of Item 5. The criteria state that the available operator action time should 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 Description

2.6.3.1 Identification of Dilution Sources

The possible flow paths of non-borated water that could result in a water slug being injected into the RCS that subsequently finds its way into the core are presented in Table 2.6-1. Considering restrictions on operations, the only source of non-borated water is the DVI lines. The maximum slug volume was determined to be 120 ft^3 (3.4 m^3). Considering the issues identified in Subsection 2.6.1, the conclusion shown in the resolutions of Table 2.6-1 is that for the APR1400 design, the scenarios defined by these issues do not result in a potential source of a non-borated water slug.

2.6.3.2 Event Analyzed

The heterogeneous dilution (rapid boron dilution) event may be possible under shutdown conditions in PWRs. Therefore, the conditions of Modes 3, 4, 5, and 6 are evaluated to determine analysis cases for the heterogeneous dilution event. Since Mode 6 is defined as a refueling operation mode, the heterogeneous dilution event is not credible in Mode 6. During startup operations, the RCS temperature of Mode 3 is highest, and that of Mode 5 is lowest. In addition, the required boron concentration for the relevant modes differs, and if the plant is in startup mode, it means that a deboration is in progress. Therefore, the heterogeneous dilution event in Mode 3, Mode 4, and Mode 5 is credible. For Mode 4, however, the results for heterogeneous dilution events are bounded by Modes 3 and 5 since the conditions of Mode 4 are between those of Mode 3 and Mode 5.

Based on the evaluation above, the following two cases are considered for the heterogeneous dilution.

In Case 1, it is assumed that the plant is in Mode 5 with reduced inventory and the boron concentration maintains its level to correspond to the shutdown margin with all rods in except the largest worth rod stuck out (N-1 conditions). This configuration results in the minimum RCS inventory and the minimum boron concentration in the RCS with the minimum shutdown margin just before the injection of the pure water slug into the safety injection system (SIS) lines.

In Case 2, it is assumed that the plant is in Mode 3 with reduced inventory and the boron concentration corresponds to N-1 conditions. Since the density of fluid is minimized in Mode 3, this assumption is conservative for the event.

The initial conditions and assumptions used in rapid boron dilution event analysis are given in Table 2.6-2. Non-safety classified systems are not credited for the rapid boron dilution event analysis. The maximum capacity of four SI pumps with common cause failure of SIS check valves is assumed to maximize the injection of the unborated water to the RCS.

2.6.3.3 Mathematical Model

Three-dimensional computational geometry model is useful for simulating the complex behavior of boron mixing with the nonsymmetric injection of the deborated water plug because of the local recirculation due to the geometric discontinuity.

However, in the APR1400 design, a two-dimensional grid system is used to simulate the complex behavior of boron mixing because each DVI nozzle is symmetrically arranged at 45°, 135°, 225°, and 315° from the reference hot leg (see Figure 2.6-1). Therefore, the axisymmetric analysis is enough for the APR1400, and the RV from downcomer annulus to the top of the fuel alignment plate is modeled.

The above scenarios were modeled using FLUENT, a computational fluid dynamics (CFD) software package. The FLUENT code is a general-purpose CFD code developed for the simulation of fluid flow, heat transfer, and chemical reaction. It can solve time-dependent and time-averaged elliptic fluid flow

equations for mass continuity, momentum, turbulence quantities, and the scalars for energy. Variables such as temperature, density, and eddy viscosity are also calculated and stored in the output.

2.6.3.4 Results

The results of the above analysis demonstrate that with a rapid injection of an unborated water slug of 120 ft^3 (3.4 m^3) into the RCS, in conjunction with the operational constraints as stated in the Technical Specifications identified in Subsection 2.6.3.2, that the maximum positive reactivity addition for both Cases 1 and 2 is less than the available shutdown margin of 6.5 percent (Mode 5) and 5.5 percent (Mode 3). The maximum positive reactivity insertion therefore results in a K_{eff} that is below criticality.

2.6.3.5 Conclusion

The analysis confirms that the acceptance criteria, as stated in Subsection 2.6.2, are met. The core remains substantially subcritical, and the RCS pressure and DNBR limits are therefore not violated.

2.6.4 Resolution

The design of the APR1400 minimizes the possibility of a rapid boron dilution event. Analyses have shown that the core remains subcritical when the maximum credible water slug is flushed through the RCS.

Therefore, the issue of a rapid boron dilution event for APR1400 can be considered resolved.

Table 2.6-1(1 of 3)

Qualitative Review Results for Possible Paths on Unborated Water

System	Flow Path	Resolution
A. Safety Injection System		
1. Standby	A. RCS leakage through first isolation check valves (SI-217, 227, 237, 247)	A. A diluted slug of water (assumed 0 ppm boron) with a volume of 0.85 m^3 (30 ft^3) per DVI line. It can have a significant effect on the external heterogeneous dilution.
	B. Leakages through SIS hot leg injection isolation valves (SI-522, 524, 532, 534) dilute SIS hot leg injection lines	B. Hot-leg injection is only used 2 to 4 hours post-LOCA when shutdown margin is large. Slug mixes with highly borated water before entering the core and has no significant effect on the heterogeneous dilution.
	C. Inadvertent refill of SIS sections with unborated water after maintenance	C. Operation and maintenance procedures require that unborated sources of water not be used to refill the SIS. There are no practical sources to refill SIS sections with unborated water in the APR1400 SIS design.
2. SIS Operation	None	Since SI pumps take suction from borated in-containment refueling water storage tank (IRWST), there is no risk inadvertent boron dilution by SI pumps.

Table 2.6-1 (2 of 3)

System	Flow Path	Resolution
B. Shutdown Cooling System		
1. Standby (Isolated)	A. Leakage of RCS fluid through first isolation valve (SI-651, -652)	A. • Leakage is into a borated SCS, does not result in a slug of pure water • Operator is required to warm up SCS and check boron concentration before injecting into RCS per operational procedures
	B. Leakage of CCW through a ruptured SC HX tube	B. • Leakage Is into a borated SCS • Dilution is bounded by check valves and normally closed gate and globe valves • Maximum ΔP is 10.5 kg/cm^2 (150 psi). Pressure quickly stabilizes before significant dilution results. • Approximate dilution of 0.0038 m^3 (1 gal) • Operator checks boron concentration upon SCS heatup, detects dilution, and corrects before injection
	C. Inadvertent refill of SCS sections with non-borated water post-maintenance	C. Owner/operator procedures require that No non-borated sources of water are used to refill the SCS. (No practical sources exist in the APR1400 design).
2. SCS Operation (Non-isolated)	A. Leakage of CCW through ruptured SC HX tube	A. • If CCWS pressure > SCS pressure, CCW inflow mixes with flow from hot leg, boron concentration > 0 ppm. • Pressure from operating SCS pump Is likely to create a ΔP such that CCW inflow is precluded. • Loss of CCW inventory into SCS is eventually detected by CCW surge tank low level alarms. • Possible volume of leakage and resulting boron concentration analysis, inadvertent boron dilution event described in Chapter 15 of APR1400 DCD Tier 2. • This event is not coincident with the charging pump event of Chapter 15, Subsection 15.4.6, of APR1400 DCD Tier 2

Table 2.6-1 (3 of 3)

System	Flow Path	Resolution
C. Reactor Coolant System Steam Generators	A. SG tube rupture resulting in secondary flow to RCS	A. Procedures require that a positive ΔP exists between primary and secondary sides of SG.
	B. Leakage of secondary fluid through ruptured tube during hydrostatic test with fuel in core, ΔP of 56.2 kg/cm^2 (800 psi) (after major steam generator maintenance)	B. Operator detects pressure change and corrects before starting RCPs. It is unlikely that fuel would be in the core during this test.
D. Chemical and Volume Control System (CVCS)	A. Upon startup from Mode 6, all offsite power is lost. Then RCPs and charging pumps shut off. As the charging pumps are restarted, powered by the EDG, the charging pumps begin to supply pure water to the vessel. Once offsite power is restored, RCPs restart and pump a slug of diluted water into the core.	A. Charging pumps can be powered by the EDG. However, when offsite power is restored, the charging pump is maintained as it is (stopped). Pumps are manually operated after power is restored, and procedures dictate operator action. Therefore, pure water is not flowing into the core automatically after the restoration of power.
	B. Unable to borate volume control tank (VCT) due to nitrogen gas binding of boric acid makeup pumps (BAMPs)	B. The boric acid storage tank (BAST) is not pressurized by any gas and is vented to gaseous water management system . In addition, the pipe routing is designed to prevent the accumulation of any gas at BAMPs. Therefore, the gas binding of BAMPs is precluded.
	C. Injection of pure water into the RCS from the reactor makeup water storage tank	C. CVCS configuration prevents this situation. A charging restricting orifice is installed downstream of the charging pump to limit the charging flow during shutdown modes, and this configuration therefore delays the time for the pure water to reach the core. In addition, there is a boron dilution alarm if there is a boron dilution event, and the alarm helps the operator to isolate the pure water line.

Table 2.6-2

Major Assumptions and Initial Conditions for Rapid Boron Dilution

Parameters		Conditions Case 1	Conditions Case 2
1	Operational mode	Mode 5	Mode 3
2	RCS temperature, °C (°F)	98.9 (210)	295 (563)
3	RCS pressure, kg/cm^2A (psia)	Atmospheric	81.6 (1,160)
4	CEA configuration	N-1	N-1
5	Available shutdown margin (%Δρ)	6.5	5.5
6	Water slug volume and boron concentration, m^3 (ft^3) / ppm	3.4 (120) / 0 ppm	3.4 (120) / 0 ppm
7	Water slug injection method	4 SIPs, max. flow	4 SIPs, max. flow
8	Water slug temperature, °C (°F)	10 (50)	10 (50)
9	Maximum critical boron concentration (C_{crit}, ppm)	912	821
10	Minimum inverse boron worth (IBW ppm / %Δρ)	73	78
11	Single failure	(1)	(1)

(1) No single failure affects the event consequences.

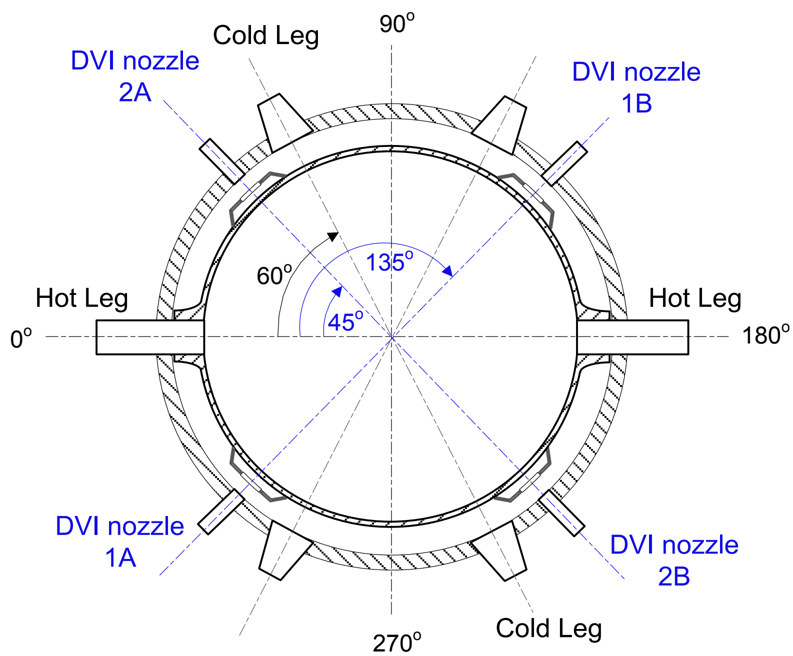

Figure 2.6-1 General Arrangement of DVI Nozzles

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. The 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 susceptibility of failure due to fire.

2.7.2 Acceptance Criteria

The major goal of the shutdown fire analysis is to provide reasonable assurance that for a fire in any fire area of the plant, redundant and/or diverse means of equipment remain operable such that the capability to remove decay heat from the core exists independent of equipment or systems located within or affected by the fire in the affected fire area.

Generic Letter No.88-17 (Reference 1) contains NRC recommendations for all holders of licenses for PWRs 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 is to prevent the recurrence of events that have the potential to damage the core and/or release radiation. Generic Letter No. 88-17 also indicates that the NRC considers the methodology in NUMARC 91-06 acceptable. NUMARC 91-06 states that proper outage planning and control, with a full understanding of the major vulnerabilities that are present during shutdown conditions, is the most effective means of enhancing safety during shutdown.

SECY-97-168 (Reference 6) provides a revised shutdown rulemaking package, which consists of three parts: (1) shutdown operations, (2) fire protection, and (3) spent fuel storage pool operations. The objective of the rule is to establish a clear, flexible, risk-informed, and enforceable regulatory framework for providing reasonable assurance that cold shutdown, refueling, and fuel (storage) pool operations continue to be conducted in a safe manner. Although the NRC withdrew the proposed rule 10 CFR 50.67 for the shutdown risk analysis and spent fuel pool operation on February 4, 1999, the above documents are considered to provide both technical and licensing bases for performing the shutdown fire analysis.

The acceptance criteria in SECY-97-168 are used as an acceptable approach to the shutdown fire analysis of the APR1400 design. The major acceptance criteria in SECY-97-168 are as follows:

a. Minimize the frequency of fires during shutdown and their potential consequences in the areas where a fire could impair the DHR system in operation.

b. To accomplish the above objective, the following criteria are met:

   (1) Provide strict administrative controls, including the well-planned procedures to control combustible materials used during an outage, control interruption of fire barriers, and control potential sources of ignition in all areas in which fire could impair the DHR function.

   (2) Limit the level of fire damage, which necessitates the use of fire protection features such as early fire detection capability, fire suppression and fire containment capabilities. The reason for the fire protection provision is that the potentially short time before core damage occurs makes it difficult to restore a fire-damaged system to service.

   (3) Develop and implement contingency plans for maintaining the fuel cladding wetted and restoring a heat removal path in the event a fire interrupts or degrades heat removal to an

ultimate heat sink. Contingency plans are required because of the need for a reliable and readily available source of water to maintain the wetted fuel cladding.

(4) Make support systems available to provide reasonable assurance that the DHR function is accomplished and support systems such as the emergency onsite power, service water, and HVAC are operational during an outage.

(5) Develop a fire protection plan that includes the above measures.

A defense-in-depth philosophy is used in the design of the fire protection system to reduce the shutdown risk due to fire. The elements in the defense-in-depth philosophy are:

- Prevent a fire from occurring
- Promptly detect and suppress a fire
- Mitigate the consequences of a fire

These elements are also used to reduce the risk associated with a shutdown fire.

2.7.3 Description

The acceptance criteria in SECY-97-168 are described in the following subsections along with a description of the APR1400 fire protection features.

2.7.3.1 Administrative Controls

Prevention is the most important element in the defense-in-depth philosophy. When this element is successful, there is no need for the other elements. To facilitate the implementation of prevention, workplace procedures and guidelines are established by the owner-operator based on guidance provided by the plant designer.

Procedural guidance includes control of combustibles, housekeeping, and control of hotwork. Development of these procedures includes a consideration of the areas in which a fire during shutdown modes of operation could pose a risk. Areas that have increased combustible loading are listed in Table 2.7-1. The procedures include requirements to reduce the risk of fire ignition during shutdown. For example, the procedure for controlling combustibles may establish a maximum amount and configuration of combustible materials that may be left unattended in any of these areas. This is not based solely on an arbitrary “good engineering practice” approach but also considers the amount of combustibles necessary to result in a fire that could cause unacceptable damage. The procedures for controlling hotwork and for housekeeping are developed by the owner-operator and implemented to avoid unnecessary restrictions on shutdown maintenance activities, yet provide a high level of fire prevention.

2.7.3.1.1 Divisional Separation

The SCS components of each division are separated from each other by 3-hour-rated fire barriers with no communicating openings (see Figure 2.7-1). All penetrations in these barriers are sealed with assemblies that are qualified to maintain the integrity of the 3-hour rating. These measures provide reasonable assurance that a fire involving one division of SCS components does not affect the redundant division.

2.7.3.1.2 Interdivisional Separation

The containment spray pump and shutdown cooling pump within one division can be interchanged. With

valve manipulations guided by approved procedures. In 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 that allows switchover from one pump to the other is located in a separate fire area, which enables operators to make the switchover without being exposed to a fire involving either the CSS or SCS. Finally, the containment spray pump is powered from a safety bus separate from the SC pump. The safety buses are separated from each other with 3-hour-rated fire walls. For example, the Division I – Class 1E 4.16 kV switchgear A is located in Fire Area F078-A25A and the Division II – Class 1E 4.16 kV switchgear B is located in Fire Area F078-AEEB (see Figure 2.7-2).

This interdivisional mechanical and electrical separation provides reasonable assurance that the operation of shutdown cooling can be maintained if a fire occurs when the redundant division is out of service.

2.7.3.1.3 Control Room Fire

Controls for the SCS components are provided at the remote shutdown panel. This panel is available during reduced inventory and refueling conditions. The remote shutdown panel is physically and electrically isolated from the control room. Therefore, the SCS can be operated and controlled from the remote shutdown panel if there is a fire in the control room during reduced inventory or refueling conditions.

2.7.3.2 Limiting the Level of Fire Damage

2.7.3.2.1 Detection

As noted in Subsection 2.7.3.1, the divisional as well as interdivisional separation with 3-hour-rated fire barriers is maintained for the SCS and CSS components. These areas are equipped with full area coverage, ceiling-mounted, ionization smoke detectors that provide an early warning alarm at the central fire alarm console in the event of a fire. Detector location and spacing are based on engineering analysis to optimize detector effectiveness. This analysis is referenced in the APR1400 fire hazards analysis, which is completed later in the design process.

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

2.7.3.2.2 Suppression

Fixed automatic suppression in the form of automatic sprinklers is not warranted in the shutdown cooling and containment spray areas because of the minimal combustible loadings in these areas. This is verified in the APR1400 fire hazards analysis, which is completed later in the design process, by the plant designer before operations.

Portable fire extinguishers and fixed manual fire hose stations provide manual firefighting capability. The fire hoses are supplied from a dedicated fire protection water supply. Because of the fire barrier arrangement described earlier, manual firefighting 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.

2.7.3.2.3 Manual Fire Fighting

A fully trained and equipped onsite fire brigade provides firefighting activities for the APR1400 (see APR1400 DCD Tier 2, Subsection 9.5.1.9.3). The brigade is thoroughly familiar with the plant layout and conducts sufficient fire drills and fire pre-planning to effectively control and suppress any credible fire. A

documented pre-fire plan that outlines the necessary firefighting strategies is prepared prior to plant startup.

2.7.3.2.4 Level of Fire Protection

The APR1400 fire protection system is not degraded or reduced during plant shutdown. There is 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. Fire protection systems remain operable for all modes of operation including reduced inventory and refueling.

In particular, the interdivisional separation, including removable walls, removable slabs, and penetrations on the fire barriers, remain intact except during the defueled period in accordance with procedural guidance related to shutdown operations. When the fire doors are open, the procedural guidance requires maintenance of the integrity of the interdivisional separation (e.g., fire watch).

2.7.3.3 Developing and Implementing Contingency Plans

The safe shutdown systems needed for DHR are as follows:

a. For the APR1400, decay heat is removed by the CSS or SCS. The CSS is an alternative to the SCS. Only plant operating Modes 4, 5, and 6 are considered for the shutdown evaluation. During those modes, the RCS pressure and temperature are less than 28.83 kg/cm^2 (410 psig) and 176.7 °C (350 °F), respectively.

b. The RCS pressure control function is available during the shutdown cooling operation because the RCS has to be pressurized to maintain a certain amount of subcooled margin. When the RCS pressure is low during the shutdown cooling operation with a loss of offsite power (LOOP), the RCS pressure can be maintained by stopping the shutdown cooling pump to allow the RCS to be pressurized by the decay heat. When the RCS pressure is high, RG vents are used to depressurize the RCS.

All fire areas in the auxiliary building that contain these systems and equipment are identified in Table 2.7-2 based on the systems and equipment needed for DHR noted above. Table 2.7-2 shows that all fire areas except the MCR and the remote shutdown room are related to only one division or train of DHR systems. A fire in the MCR or remote shutdown room does not jeopardize the DHR capability as described in the fire hazard/safe shutdown analysis.

The containment building contains redundant trains of DHR systems and equipment. Table 2.7-3 lists the DHR equipment in the containment building. The concerns associated with a fire during shutdown modes of operation within containment building are noted below. However, a detailed review is performed at the final design stage when cable routing design information is available.

a. A fire in the containment building fire area could damage cables for the Division 1 and 2 shutdown cooling isolation valves SI-652/654 and SI-651/653. These valves are required to be open to reach cold shutdown. When both trains of these valves are damaged due to a fire in this fire area, enough time is available for the operator to manually open at least one pair of the valves after the fire is extinguished. However, re-entry into the fire area is not allowed per SECY-93-087. Therefore, cables for these valves are routed to minimize the likelihood of damaging both trains. Final review is required for cable separation and protection.

b. Cables for both channels of ex-core neutron monitoring are routed in this area. Cables are routed to maximize separation. Cable routing is reviewed at final design stage.

c. Cables for both channels of PZR pressure and level are routed in the containment building fire area. Instruments are located and cables are routed to maximize separation, which is reviewed at the final design stage.

d. The redundant trains of RG vent valves from the PZR and reactor vessel are located in the containment building fire area. At least one train of vent paths from the PZR and reactor vessel is available for RCS pressure control. Each train of valves is located and cables are routed to maximize separation, which is reviewed at the final design stage.

Based on the above review, a fire in the auxiliary building during the shutdown modes of operation has the potential to impair the DHR capability. However, inside containment, there is potential to jeopardize both divisions of DHR equipment due to a fire in the containment during shutdown operation. Therefore, separation between redundant equipment and associated cables is maximized as noted above. In addition, appropriate administrative controls are established and implemented to limit the effects of a fire during shutdown or maintenance within the containment to one division of equipment. The administrative control includes procedures restricting maintenance on redundant divisions of DHR and support systems at the same time.

2.7.3.4 Providing Support Systems

Major support systems needed for shutdown operation are as follows:

- Emergency diesel generator
- Component cooling water
- Essential service water
- Essential chilled water
- Class 1E electrical distribution

The major instrumentation needed to monitor the shutdown-related systems operation are as follows:

- PZR pressure
- Source range neutron flux monitor
- SC/CS HX inlet/outlet temperature

See Subsection 2.7.3.3 for an evaluation of support systems.

2.7.3.5 Developing a Fire Protection Plan

Reasonable assurance is provided that the findings from the shutdown risk analysis including the administrative controls and contingency plans are incorporated into the APR1400 fire protection plan.

2.7.4 Resolution

The fire protection features provided by the APR1400 design are consistent with the acceptance criteria in Subsection 2.7.2. These features significantly reduce the risk due to fire during shutdown operation to an acceptable level. The combination of fire protection features resulting from using the defense-in-depth philosophy minimizes the potential for fire damage to systems required for shutdown operations.

The fire protection issue is resolved by the APR1400 design features.

Table 2.7-1

Areas of Increased Outage-Related Combustible Material

Fire Area	Occupancy
F055-A30A	SC Heat Exchanger Room A
F055-A30B	SC Heat Exchanger Room B
F055-A01C	Containment Spray Heat Exchanger Room A
F055-A01D	Containment Spray Heat Exchanger Room B
F055-A18A	Pipe Chase and Valve Room
F055-A18B	Pipe Chase and Valve Room
F055-A14C	Pipe Chase and Valve Room
F055-A14D	Pipe Chase and Valve Room
F050-A01C	CS Pump and Miniflow Heat Exchanger Room A
F050-A01D	CS Pump and Miniflow Heat Exchanger Room B
F050-A04A	SC Pump and Miniflow Heat Exchanger Room A
F050-A04B	SC Pump and Miniflow Heat Exchanger Room B
F000-ACVU	Personnel Airlock Entrance Area
F100-A24A	SFP Cooling Heat Exchanger Room
F100-A32B	SFP Cooling Heat Exchanger Room
F156-A04B	Containment Entrance Area
F000-AFHU	Loading and Unloading Area
F000-ADGC	Diesel Generator Room C
F000-ADGD	Diesel Generator Room D

Table 2.7-2 (1 of 5)

Auxiliary Building Fire Areas Containing Decay Heat Removal Equipment

Fire Area	Area Description	Equipment	Division
F055-A01C	Containment Spray Heat Exchanger Room A	CS HX	I
F055-A01D	Containment Spray Heat Exchanger Room B	CS HX	II
F055-A02C	CCW Pump Room C	CCW Pump, Valves	I
F055-A02D	CCW Pump Room D	CCW Pump, Valves	II
F055-A04C	Seismic CAT-I Fire Water Tank Room A	Fire Water Tank	I
F055-A04D	Seismic CAT-I Fire Water Tank Room B	Fire Water Tank	II
F055-A14C	Pipe Chase and Valve Room	CS Valves, Piping	I
F055-A14D	Pipe Chase and Valve Room	CS Valves, Piping	II
F050-A01C	CS Pump and Miniflow Heat Exchanger Room A	CS Pump, Miniflow HX, Valves	I
F050-A01D	CS Pump and Miniflow Heat Exchanger Room B	CS Pump, Miniflow HX, Valves	II
F050-A04A	SC Pump and Miniflow Heat Exchanger Room A	SC Pump, Miniflow HX, Valves	I
F050-A04B	SC Pump and Miniflow Heat Exchanger Room B	SC Pump, Miniflow HX, Valves	II
F055-A18A	Pipe Chase and Valve Room	SC Valves, Piping	I
F055-A18B	Pipe Chase and Valve Room	SC Valves, Piping	II
F055-A30A	General Access Area A - 55′-0″	SC HX, VSI-310, 312	I
F055-A30B	General Access Area B - 55′-0″	SC HX, VSI-311, 313	II
F055-A02A	CCW Pump Room A	CCW Pump, Valves	I
F055-A02B	CCW Pump Room B	CCW Pump, Valves	II
F055-A22A	Pipe Chase	CC Piping	I
F055-A22B	Pipe Chase	CC Piping	II

Table 2.7-2 (2 of 5)

Fire Area	Area Description	Equipment	Division
F078-A01C	PNS Switchgear Rm	4.16 kV PNS Switchgear	I
F078-A01D	PNS Switchgear Rm	4.16 kV PNS Switchgear	II
F078-A02C	Class 1E Switchgear 01C Rm	4.16 kV Class 1E Switchgear	I
F078-A02D	Class 1E Switchgear 01D Rm	4.16 kV Class 1E Switchgear	II
F078-A03C	Class 1E Load Center 01C Rm	480 V Class 1E Load Center, MCCs	I
F078-A03D	Class 1E Load Center 01D Rm	480 V Class 1E Load Center, MCCs	II
F078-A05C	Train C DC and IP Equip. Room	125 Vdc Class 1E Battery Charger, Reg. Transformer, 120 Vac Class 1E Inverter, 125 Vdc Control Center	I
F078-A05D	Train D DC and IP Equip. Room	125 Vdc Class 1E Battery Charger, Reg. Transformer, 120 Vac Class 1E Inverter, 125 Vdc Control Center	II
F078-A56A	Train A DC and IP Equip. Room	125 Vdc Class 1E Battery Charger, Reg. Transformer, 120 Vac Class 1E Inverter, 125 Vdc Control Center	I
F078-A56B	Train B DC and IP Equip. Room	125 Vdc Class 1E Battery Charger, Reg. Transformer, 120 Vac Class 1E Inverter, 125 Vdc Control Center	II
F078-A07C	Train C Battery Room	125 Vdc Class 1E Battery	I
F078-A07D	Train D Battery Room	125 Vdc Class 1E Battery	II
F100-A11A	Train A Battery Room	125 Vdc Class 1E Battery	I
F100-A11B	Train B Battery Room	125 Vdc Class 1E Battery	II
F078-A19A	General Access Area A - 78′-0″	Cables for Ch. A and C SR Equipments	I
F078-A19B	General Access Area B - 78′-0″	Cables for Ch. B and D SR Equipments	II
F078-AGAC	General Access Area C - 78′-0″	Cables for Ch. A and C SR Equipments, CCW Makeup Pump	I
F078-AGAD	General Access Area D - 78′-0″	Cables for Ch. B and D SR Equipments, CCW Makeup Pump	II
F078-A11C	Essential Chiller Room A	Essential Chiller, Chilled Water Pump	I
F078-A11D	Essential Chiller Room B	Essential Chiller, Chilled Water Pump	II
F078-A12C	Essential water Chiller Room C	Essential Chiller, Chilled Water Pump	I

Table 2.7-2 (3 of 5)

Fire Area	Area Description	Equipment	Division
F078-A12D	Essential water Chiller Room D	Essential Chiller, Chilled Water Pump	II
F078-A20A	Motor-Driven AFW Pump Room A	Aux. Feedwater Pump	I
F078-A20B	Motor-Driven AFW Pump Room B	Aux. Feedwater Pump	II
F078-AAFC	Turbine-Driven AFW Pump Room C	Aux. Feedwater Pump	I
F078-AAFD	Turbine-Driven AFW Pump Room D	Aux. Feedwater Pump	II
F078-A21A	Pipe Chase	CC and SC Piping	I
F078-A21B	Pipe Chase	CC and SC Piping	II
F078-A25A	Class 1E Switchgear 01A Rm	4.16 kV Class 1E SWGR, 480V Class 1E LC	I
F078-AEEB	Class 1E Switchgear 01B Rm	4.16 kV Class 1E SWGR, 480V Class 1E LC	II
F078-A09C	HVAC Chase	HVAC Duct	I
F078-A09D	HVAC Chase	HVAC Duct	II
F100-AEEA	480V Class 1E MCC 01A Rm	480V Class 1E MCC	I
F100-AEEB	480V Class 1E MCC 01B Rm	480V Class 1E MCC	II
F000-ADGC	Diesel Generator Room C	Diesel Generator	I
F000-ADGD	Diesel Generator Room D	Diesel Generator	II
F100-A10A	General Access Area A - 100′-0″	Cables for Ch. A and C SR Equipment	I
F100-AGAC	General Access Area C - 100′-0″	Cables for Ch. A and C SR Equipment	I
F100-A10B	General Access Area B - 100′-0″	Cables for Ch. B and D SR Equipment	II
F100-A06D	General Access Area D - 100′-0″	Cables for Ch. B and D SR Equipment	II
F100-A13A	Mech. Penetration Room	Piping, Valves	I
F100-A13B	Mech. Penetration Room	Piping, Valves	II

Table 2.7-2 (4 of 5)

Fire Area	Area Description	Equipment	Division
F100-A24A	SFP Cooling Pump and HX Rm	SFP Cooling Pump/HX	I
F100-A32B	SFP Cooling Pump and HX Rm	SFP Cooling Pump/HX	II
F120-AGAC	General Access Area C - 120′-0″	Cables for Ch. A and C SR Equipment	I
F120-AGAD	General Access Area D - 120′-0″	Cables for Ch. B and D SR Equipment	II
F120-A09C	Electrical Penetration Room C	Cables for Ch. C SR Equipment	I
F120-A09D	Electrical Penetration Room D	Cables for Ch. D SR Equipment	II
F120-A16A	Mech. Penetration Room	Piping, Valves	I
F120-AMPB	Mech. Penetration Room	Piping, Valves	II
F120-A15B	Class 1E MCC 03B Room	480V Class 1E MCC	II
F137-A01C	Cable Spreading Area	Cables for Ch. C SR Equipment	I
F137-A01D	Cable Spreading Area	Cables for Ch. D SR Equipment	II
F137-A06D	Remote Shutdown Room	Remote Shutdown Panels	I / II
F137-A10C	Class 1E MCC 03C Room	480V Class 1E MCC	I
F137-A10D	Class 1E MCC 03D Room	480V Class 1E MCC	II
F137-A11C	Elec. Penetration Rm C	Cables for Ch. A and C SR Equipment	I
F137-A11D	Elec. Penetration Rm D	Cables for Ch. B and D SR Equipment	II
F137-AEPA	Elec. Penetration Rm A	Cables for Ch. A SR Equipment	I
F137-AEPB	Elec. Penetration Rm B	Cables for Ch. B SR Equipment	II
F137-A09C	General Access Area C - 137′-0″	Cables for Ch. A and C SR Equipment	I
F137-AGAD	General Access Area D - 137′-0″	Cables for Ch. B and D SR Equipment	II
F137-A23A	480V Class 1E MCC 03A Room	480V Class 1E MCC	I

Table 2.7-2 (5 of 5)

Fire Area	Area Description	Equipment	Division
F120-A15B	480V Class 1E MCC 03B Room	480V Class 1E MCC	II
F137-A15A	480V Class 1E MCC 04A Room	480V Class 1E MCC	I
F137-A15B	480V Class 1E MCC 04B Room	480V Class 1E MCC	II
F157-A19C	I&C Equipment Room C	SR Cabinets	I
F157-A19D	I&C Equipment Room D	SR Cabinets	II
F157-A25C	I&C Equipment Room A	SR Cabinets	I
F157-A01D	I&C Equipment Room B	SR Cabinets	II
F157-AMCR	Control Room Area	Main Control Boards, Cables for Ch. A, B, C, D SR Equipment	I / II
F157-ACPX	Computer Room Area	Cables for Ch. D SR Equipment	II
F156-A20C	I&C Equip. Room	Cables for Ch. A SR Equipment	I
F156-A20D	I&C Equip. Room	Cables for Ch. B SR Equipment	II
F137-A35C	Reactor Trip Switchgear Room	Reactor Trip Switchgear	II
F137-A36C	Reactor Trip Switchgear Room	Reactor Trip Switchgear	I
F137-A37C	Reactor Trip Switchgear Room	Reactor Trip Switchgear	II
F137-A38C	Reactor Trip Switchgear Room	Reactor Trip Switchgear	I
F174-A24C	Control Room Area Supply AHU/ACU Room	Control Room Area Supply AHU/ACU,	I
F174-A24D	Control Room Area Supply AHU/ACU Room	Control Room Area Supply AHU/ACU,	II
F174-AGAC	General Access Area C - 174′-0″	Cables for Ch. A and C SR Equipment, CCW Surge Tank, Essential Compression Tank	I
F174-AGAD	General Access Area D - 174′-0″	Cables for Ch. B and D SR Equipment, CCW Surge Tank, Essential Compression Tank	II

Table 2.7-3

Decay Heat Removal Equipment Inside Containment (Fire Area : F000-CNB)

System	Equipment	Division
SI	Valves SI-652, 654, 651, and 653 ; cables	I and II
NR	Excore neutron monitoring (two channels: JNR-001A, 001B); cables	I and II
RC	RCS hot leg loop temperature element (JRC-TE-132HA, 132HB, 133HA, 133HB)	I and II
	RCS cold leg loop temperature element (JRC-TE-142CA, 142CB, 143CA, 143CB)	I and II

TS

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Figure 2.7-1 Nuclear Island Fire Barrier Locations, Plan at El. 55'- 0"

TS

Figure 2.7-2 Nuclear Island Fire Barrier Locations, Plan at El. 78'- 0"

2.8 Instrumentation

2.8.1 Issue

Operators have had difficulty in many cases in determining plant parameters and equipment status during depressurized, shutdown conditions. The difficulty is due to the amount and quality of available information that is marginally adequate or inadequate for prevention, recognition, and mitigation of abnormal conditions in a timely manner. This information includes in particular the RCS water level, reactor core exit temperature, and SCS performance.

Loss of shutdown cooling can be partially attributed to misleading, inaccurate, or erroneous RV level indication, particularly when RCS coolant level is lowered to within the hot leg between the level required for SG nozzle dam installation and the level required to prevent vortexing in the SC suction line. Refer to Figure 2.8-1. Vortex formation in the SCS suction lines is a function of SCS flow rate and hot leg level. Providing an adequate fluid level in the hot leg above the level at which vortexing occurs provides reasonable assurance that the shutdown cooling fluid does not entrain air. To avoid these undesired consequences, mid-loop operations are not recommended at RCS levels lower than 0.58 m (23 in) in the hot legs at a SCS flow rate of 14,384.57 ~ 17,791.44 L/min (3,800 ~ 4,700 gpm). Thus, during mid-loop operations with the RCS level maintained at 0.58 m (23 in) or above in the hot leg, vortex formation and air entrainment into the SCS are avoided.

The NRC recommends that advanced reactor designs include an enhanced instrumentation package that provides reasonable assurance that:

a. Reduced inventory operations can be accurately and continuously measured. For example, accurate instrumentation can establish RCS coolant level any time during the drain-down process. Accurate level measurement can assist in differentiating between the anticipated dynamic effects of the drain-down process and additional, unintended inventory losses

b. A loss of shutdown cooling event during reduced inventory operations can be readily detected. Detection provides reasonable assurance of a timely response to a loss of shutdown cooling event. The instrumentation should "provide reliable indication of parameters that describe the state of the RCS and the performance of systems normally used to cool the RCS for both normal and accident conditions."

The NRC has specified that instrumentation for reduced inventory conditions provide both visible and audible indications of abnormal conditions in RV temperature and level and SCS performance.

2.8.2 Acceptance Criteria

The instrumentation provided for reduced inventory operations in the APR1400 design reduces the safety risks associated with shutdown modes of operation. Instrumentation is 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 if one occurs.

The recommendations in Enclosure 2 of Generic Letter No. 88-17 for minimum indications and alarms in the MCR are as follows:

a. Two independent RCS level indications when the RV head is on the vessel

b. At least two independent temperature measurements representative of the core exit whenever the RV head is located on the top of the RV

c. Capability of continuously monitoring SCS performance whenever an SCS is being used to cool the RCS

d. Visible and audible indications of abnormal conditions in temperature, level, and SCS performance.

Enclosure 2 of Generic Letter No. 88-17 also includes the following NRC concerns and suggestions on meeting the recommendations:

> … 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.
>
> We expect each licensee … to provide sufficient information be provided to the operators that an approaching SCS malfunction is clearly indicated.
>
> 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.
>
> … The low limit of the range of level indication must be below the level necessary for operation of DHR systems. … 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 mitigation of a loss of DHR accident.

Subsection 2.8.3.2 contains a description of the APR1400 instrumentation package for reduced inventory operations, including:

- Monitored parameters
- Instrumentation ranges and accuracies
- Alarm setpoints
- Instrument availability
- Display and monitoring capability
- Quality assurance

A summary of the APR1400 design features that meet the above NRC recommendations for instrumentation is provided in the following subsections.

2.8.3 Description

2.8.3.1 Instrumentation Design Basis

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

- Operations from a solid plant to mid-loop conditions
- Instrumentation features that 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 SCS, including those that contribute to vortexing
- Plant response to losses of shutdown cooling due to various initiators
- 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 shutdown cooling operation
- Mitigation –timely response to a loss of shutdown cooling

These goals are achieved by the design features of the APR1400 instrumentation described in the following subsections.

2.8.3.2 Instrumentation Description

Table 2.8-1 describes the instrumentation package for reduced inventory operations included in the APR1400 design. Additional details are provided below.

2.8.3.2.1 Level

Four unique sets of instruments are provided for the measurement of level during RCS drain down and reduced inventory operations. These instruments make up the refueling water level indication system (RWLIS).

The first set of instruments is a pair of wide-range, pressure differential (dP)-based level sensors. These sensors are provided to measure the level between the PZR and the bottom of the hot leg during drain-down operations. Another pair of dP-based level sensors is used to determine RCS water level once it is within the RV. These narrow-range level sensors function to measure level between the DVI nozzle and the bottom of the hot leg.

The permanent refueling water level indication system (PRWLIS) consists of one wide-range and one narrow-range dP instrument. The lower taps of the wide-range PRWLIS are connected to the SCS suction line for RCS loop 1 and hot leg drain line for RCS loop 2, respectively. The lower taps of the narrow range PRWLIS are connected to SCS suction lines for RCS loop 1 and loop 2. Separate lower level taps are provided for each RCS loop. The upper tap of the loop 1 and 2 wide-range PRWLIS is provided at the PZR pressure instrument line in order to minimize the steam condensing effect, while the upper taps of the loop 1 and 2 narrow-range PRWLIS are provided at each DVI nozzle line (see Figure 2.8-2). Because of the location of the upper level taps, each dP instrument operates with or without the RV head in place.

In addition to the dP-based instruments described above, a heated junction thermocouple (HJTC) system may also be available for RV level measurement during Mode 5 reduced inventory operations. This HJTC system displays the output from the two inadequate core cooling monitoring probes that are located inside the RV. The range of these probes extends from the RV head to the fuel alignment plate (see Figure 2.8-3). The measurement of RCS water level via these probes is limited to the periods when the RV head is installed.

The local refueling water level indication system (LRWLIS) has two trains (loop 1 and loop 2), and each train has a sight glass that has a minimum visible span of 3.81 m (150 in) above the bottom of the hot leg. The lower taps of the loop 1 and 2 LRWLIS are provided at the SCS suction line for RCS loop 1 and the

hot leg drain line for RCS loop 2, respectively. The upper tap of the loop 1 and 2 LRWLIS is provided at another PZR pressure instrument line in order to minimize the steam condensing effect (see Figure 2.8-2).

The ultrasonic level measurement system (ULMS) is installed temporarily on both hot legs to monitor the water level of the hot leg during mid-loop operation. ULMS sensors are installed at a distance greater than 1 m (39.4 in) from SCS suction nozzle toward the RV or at a proper location to provide reasonable assurance of a correct level measurement.

The use of both wide-range and narrow-range dP instruments, a pair of local sight glass systems, a pair of ULMS, and a pair of HJTC probes for refueling water level monitoring provides highly reliable, redundant, and independent indication of RV water level. Overlapping instrument ranges provide continuous drain-down measurement from the PZR to a level below that necessary for SCS operation. Since this level instrumentation is independent, common mode misoperation or failures due to dynamic effects are not masked.

Each independent level instrument provides a suitable measurement and is accurate for its intended range of use. For mid-loop operations, the narrow-range dP instruments, local sight glass system, and ULMS provide accurate level measurements. Accurate measurements are critical since there is a narrow margin between the RCS water level necessary for nozzle dam installation and that required to prevent SCS pump cavitation. The refueling water level indication is displayed and alarmed in the MCR because of its importance to plant safety.

2.8.3.2.2 Temperature

Several instruments are available for continuous temperature measurements during reduced inventory operations. These include:

- Core exit thermocouples (CETs)
- Shutdown cooling heat exchanger (HX) inlet and outlet line temperature sensors
- Hot-leg resistance temperature detectors (RTDs)
- Refueling water level instruments temperature sensor (HJTC probe only)

The CETs provide representative indications of the core exit temperature when the SCS is operational. If the shutdown cooling is lost, inputs from the CETs and HJTC probes are available to track the response to the loss of shutdown cooling or the approach to boiling.

Per Enclosure 2 to Generic Letter No. 88-17, temperature measurement is provided with the RV head off. The temperature instruments operable during this mode are the hot leg RTDs and, prior to fuel shuffle, the CETs. Each RCS hot leg has a total of seven resistance temperature detector (RTD) channels, which are located in the hot leg at the junction of the SCS suction nozzle. In relation to the hot leg horizontal centerline, two RTD channels are located above the centerline, one RTD channel is at the centerline, and two dual RTDs, which consist of two narrow-range and two-wide range RTD channels, are below the centerline. Only the lowest two wide-range RTD channels in each hot leg provide input to the temperature reading for mid-loop operations since they are the only ones in full contact with reactor coolant. The lowest probes penetrate the inside diameter of the hot leg pipe at approximately 0.3 m (12 in) below the mid-loop fluid level, thus providing reasonable assurance that accurate readings are provided.

All temperature sensors have associated alarms in the MCR to be used as aids in determining the response to a loss of shutdown cooling and tracking the approach to boiling

2.8.3.2.3 Shutdown Cooling System Performance

As stated in Enclosure 2 of Generic Letter No. 88-17, sufficient information is available to the MCR operator to indicate an approaching SCS malfunction. Indications of sufficient pump suction pressure and possible vortexing include unsteady pump current (as indicated by SCS / CSS pump motor current), loss or reduction in shutdown cooling flow (as indicated by the SCS flow rate), insufficient pump net positive suction head (NPSH) (as indicated by the pump suction pressure sensor), and indication of rising RCS level (as water is displaced by the air and vapor in the SCS). If a pump gives indications of air ingestion or cavitation, alarms prompt the operator to stop the pump immediately. As detailed in Subsection 2.8.3.2.5, the shutdown cooling panel displays include valve lineup information for critical shutdown cooling flow paths.

2.8.3.2.4 Quality Assurance

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

- Core exit thermocouples (CETs)
- Hot-leg RTDs
- Heated junction thermocouple probe assembly
- Shutdown cooling flowmeter
- Shutdown cooling HX inlet and outlet 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 Korea Hydro & Nuclear Power Co., Ltd. (KHNP) Quality Assurance Program designates items that are safety-related as Quality Class 1 equipment and are therefore subject to the highest level of quality activity.

The KHNP Quality Assurance Program designates items that 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 implementation of the Quality Assurance Program that is commensurate with the intended use . In the procurement of the instrumentation, appropriate technical requirements and quality requirements are specified in the purchase order to this end. The following Quality Class 2 instruments identified in Table 2.8-1 are classified as non-safety related:

- Refueling water level indicators (wide- and narrow-range dP design)
- Shutdown cooling (SC) / containment spray (CS) pump suction and discharge pressure sensors
- SC/CS pump motor current sensors

2.8.3.2.5 Display and Monitoring Capability

Details of the APR1400 MCR evaluation are described in APR1400 DCD Tier 2, Chapter 18. In addition to the following summary, refer to Chapter 18 for detailed or supplementary explanation of MCR presentation.

The operator obtains plant information from the following sources in the APR1400 MCR:

a. Large display panel (LDP)

b. Alarms and associated alarm messages

c. Qualified indication and alarm system (QIAS) flat panel displays (FPDs), which provide operator-selected information

d. Information processing system (IPS) display formats containing essentially all-power plant information

e. Component and process control indications via the soft control and operator modules

There are a number of APR1400 design features in Items 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 Large Display Panel

The large display panel (LDP) is used to quickly assess plant status, organize operational concerns, and establish priorities for operator action. Information provided on the LDP includes:

a. Major system and component status shown on an overview schematic that is representative of the current operating heat transport systems

b. Alarms that identify deviations in process parameter values, component operation, and process states

c. Alarms that indicate the status of the critical plant safety functions and indicate the status of the success paths that are associated with the critical functions

d. Key representative parameters (e.g., RCS temperature, RV level)

e. A variable display area (VDA) in which operator-selected cathode ray tube (CRT) display pages may be output

Alarm tiles are provided for the following plant critical functions:

- Reactivity control
- RCS inventory control
- RCS pressure control
- Vital auxiliaries
- Core heat removal
- RCS heat removal
- Containment temperature and pressure control
- Containment isolation

- Containment combustible gas control

APR1400 alarms are mode dependent and equipment dependent to provide reasonable assurance of 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 process parameter input. In either case, the operator is quickly made aware of the affected critical function(s). For example, a high core exit temperature alarm state is used as an input to the core heat removal critical safety function alarm during a loss of shutdown cooling.

The systems represented on the LDP are the major heat transport pathways and the systems that are required to support the heat transport process. These systems include those that require availability monitoring per Regulatory Guide 1.47 (Reference 16) and all major success paths that support plant critical functions.

System information presented on an LDP 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 inform an operator about possible underlying causes of critical function alarms.

2.8.3.2.5.2 Alarms and Associated Alarm Messages

Alarms are indicated on the LDP, individual information processing system (IPS) display pages, and the dedicated alarm CRT. The alarm CRT provides detailed alarm messages associated with each alarm. The qualified indication and alarm system (QIAS) flat panel display (FPD) can also be used to obtain detailed alarm lists associated with each alarm. Alarms are functionally grouped together on the LDP and IPS display pages. SCS alarms are available on the LDP and the IPS display pages. Detailed shutdown cooling alarm lists are available on the dedicated IPS alarm and the QIAS FPDs. Individual alarm inputs to the shutdown cooling grouped alarm or each train include:

- Low shutdown cooling pump header pressure
- Low shutdown cooling flow
- High shutdown cooling HX outlet temperature
- Shutdown cooling pump motor current deviation

In addition, alarms for RCS conditions are provided, with individual inputs for the following shutdown, depressurized conditions:

- Low RCS water level
- High core exit temperature
- Low refueling cavity level

To provide reasonable assurance of alarm validity, all APR1400 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 provide reasonable assurance of a true “dark board” when alarms do not exist. These features enhance operator diagnosis of alarms when they exist.

2.8.3.2.5.3 QIAS Flat Panel Displays

QIAS FPDs are provided on the MCR. These devices allow the operator to select certain parameters. The selected parameters are indicated on the FPDs after display selection is made by the operator.

QIAS FPDs 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 are composed of different ranges of information, QIAS automatically shifts to the appropriate range and indicates to the operator that a range change has occurred.

QIAS FPDs can be used to support shutdown cooling by displaying, per operator selection, a variety of key shutdown cooling parameters as:

- Shutdown cooling system (per train)
  - Return line temperature
  - Pump header pressure
  - Flow
  - HX inlet temperature
  - HX outlet temperature
  - Pump motor current
- Reactor coolant system
  - PZR level
  - RCS level
  - Pressure
  - Core exit temperature
  - Refueling cavity level

2.8.3.2.5.4 Information Processing System Display Pages

Information processing system (IPS) display pages contain, in a structured hierarchy, all of the APR1400 plant information that is available to the operator. The IPS display 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, IPS display formats are designed to aid operational activities of the plant by providing trends, categorized listings, messages, and operational prompts, as well as to alert the operator of abnormal processes.

The CRT displays are provided by the IPS. Any display page is available at any information CRT (except the alarm CRT, which is dedicated to alarm messaging). Operator acknowledgment of alarms (via the alarm CRT) also acknowledges the same alarm in QIAS (and vice versa). The alarm actuation message indicates the cause of the alarm on the information CRT, similar to the QIAS.

The SCS is shown the display pages. These displays include all necessary information to clearly

describe the status and performance of the system. Information 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 displays include RCS level and core exit temperature to integrate the shutdown cooling and RCS status for this display. The RCS is also presented on a separate IPS display. Any of these IPS displays may, at the option of the operator, be output on the variable display area (VDA) of the large display panel.

2.8.3.2.5.5 Component and Process Control Indicators

APR1400 component control is normally achieved through the soft control FPDs that are located on the control work stations. Via the soft control, the operator accesses and actuates equipment and systems. The SCS controls are accessible via the soft controls. In addition, the minimum inventory control as provided on the safety console provides for shutdown cooling control as specified in the APR1400 DCD Tier 2, Subsection 18.7.6.

2.8.3.2.5.6 Alarm Characteristics

There are a number of special features in the design of the APR1400 alarm that support operator diagnosis of alarm conditions and that are particularly supportive of depressurized, shutdown operations. The features are:

a. Mode and equipment status dependency

b. Audible alarm information

c. Stop flash feature

d. Operator-established alarms

e. Flags

By providing for critical function alarms, success path alarms, personnel hazard alarms, and equipment damage alarms, the operator can rapidly determine the type and relative significance of alarms. For example, an inventory control critical function alarm without a concurrent volume control success path performance alarm immediately suggests that inventory may be decreasing due to a non-success-path cause, such as a coolant leak. In a coolant leak, the indications of the IRWST level and the containment temperature, pressure, and humidity are immediately checked by the operator. Similar distinctions can be made by the operator for single or multiple alarm conditions to assist the operator in quickly establishing needs and priorities for operator action.

2.8.4 Resolution

The issue of instrumentation for shutdown operation is resolved in the APR1400 by the instrumentation and MCR displays described in previous sections of this report. This instrumentation meets or exceeds the recommendations of Generic Letter No. 88-17 and significantly reduces risk associated with operations during shutdown, particularly when the reactor is in a reduced inventory condition, as long as the reduced inventory instrumentation is placed into operation prior to the start of the drainage process.

The APR1400 MCR provides an LDP mimic display section, QIAS FPD displays, IPS displays, and alarms that meet the acceptance criteria described in Subsection 2.8.2. Indication and alarms are made available for RCS level and temperature. In addition, SCS status and performance are monitored on information CRTs. Shutdown cooling system performance is alarmed on the LDP and on the alarm 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 (1 of 2)

Reduced Inventory Instrumentation Package

Monitored Parameter	Instrument Type	Instrument Function	Range	Indication and Alarm Location
RCS water level	Permanent refueling water level indication system (dP-based design)	Continuous, redundant wide range RCS water level indication during drain-down operations	Wide range from the bottom of hot leg to PZR. An upper tap of loop 1 and 2 instruments is located at the PZR pressure instrument line, while each lower tap of loop 1 and 2 instruments is located at the SCS suction line and the hot leg drain line, respectively.	MCR with indication
RCS water level	Permanent refueling water level indication system (dP-based design)	Continuous, redundant narrow range RCS water level indication during reduced inventory operations	Narrow range from the bottom of hot leg to the DVI nozzle. An upper tap of loop 1 and 2 instruments is located at each DVI nozzle line, while each lower tap of loop 1 and 2 instruments is located at the SCS suction lines.	MCR with indication and high, low, and low-low alarms
RCS water level	Local refueling water level indication system (sight glasses)	Continuous, redundant RCS water level indication during reduced inventory operations	Visible span of 3.81 m (150 in) from the bottom of hot leg. An upper tap of loop 1 and 2 instruments is located at the PZR pressure instrument line, while each lower tap of loop 1 and 2 instruments is located at the SCS suction line and the hot leg drain line, respectively.	MCR with indication and high, low, and low-low alarms
RCS water level	HJTC system	Independent continuous level indication in the RV	Top of the vessel down to the fuel alignment plate	MCR with indication and low alarms
RCS water level	Ultrasonic level measurement system	Independent, continuous, narrow range level indication in the hot leg during mid-loop operation	Level range from 20 % to 100 % of the hot leg	MCR with indication and high, low, and low-low alarms

Table 2.8-1 (2 of 2)

Monitored Parameter	Instrument Type	Instrument Function	Range	Indication and Alarm Location
RCS temperature	Core exit thermocouples	Two independent temperature measurements of coolant exiting core (two channels).	0 ~ 1,260 °C (32 ~ 2,300 °F)	MCR with indication and high temperature alarm
RCS temperature	Heated junction thermocouples	Continuous, independent temperature measurement inside the RV.	0 ~ 1,260 °C (32 ~ 2,300 °F)	MCR with indication
Hot leg temperature	Resistance temperature detectors	Measures core exit temperature in the hot leg at both SCS suction line regions.	Wide range with a higher accuracy	MCR with indication
SCS flow rate	Flowmeter	SCS performance	Bounds SCS pump flow range	MCR with indication and low flow alarm
SCS pump / CSS pump discharge pressure	Pressure sensor	Measures individual pump discharge pressures.	0 to system design pressure	MCR with indication and low pressure alarm
SCS pump / CSS pump motor current	Current sensor	Measure current drawn by pump motor.	Range of motor current	MCR with indication
SCS pump / CSS pump suction pressure	Pressure sensor	Measures pump suction pressure in each pump.	0 to system design pressure	MCR with indication
SC HX inlet and return line temperature	Temperature sensor	Measures temperature in the suction and discharge lines of the SC HX.	4.4 ~ 200 °C (40 ~ 392 °F)	MCR with indication
SCS valve position indication	Valve position indication open/closed or throttled	Status of valve positions in the SCS.	Open / closed / throttled position indication	MCR with indication

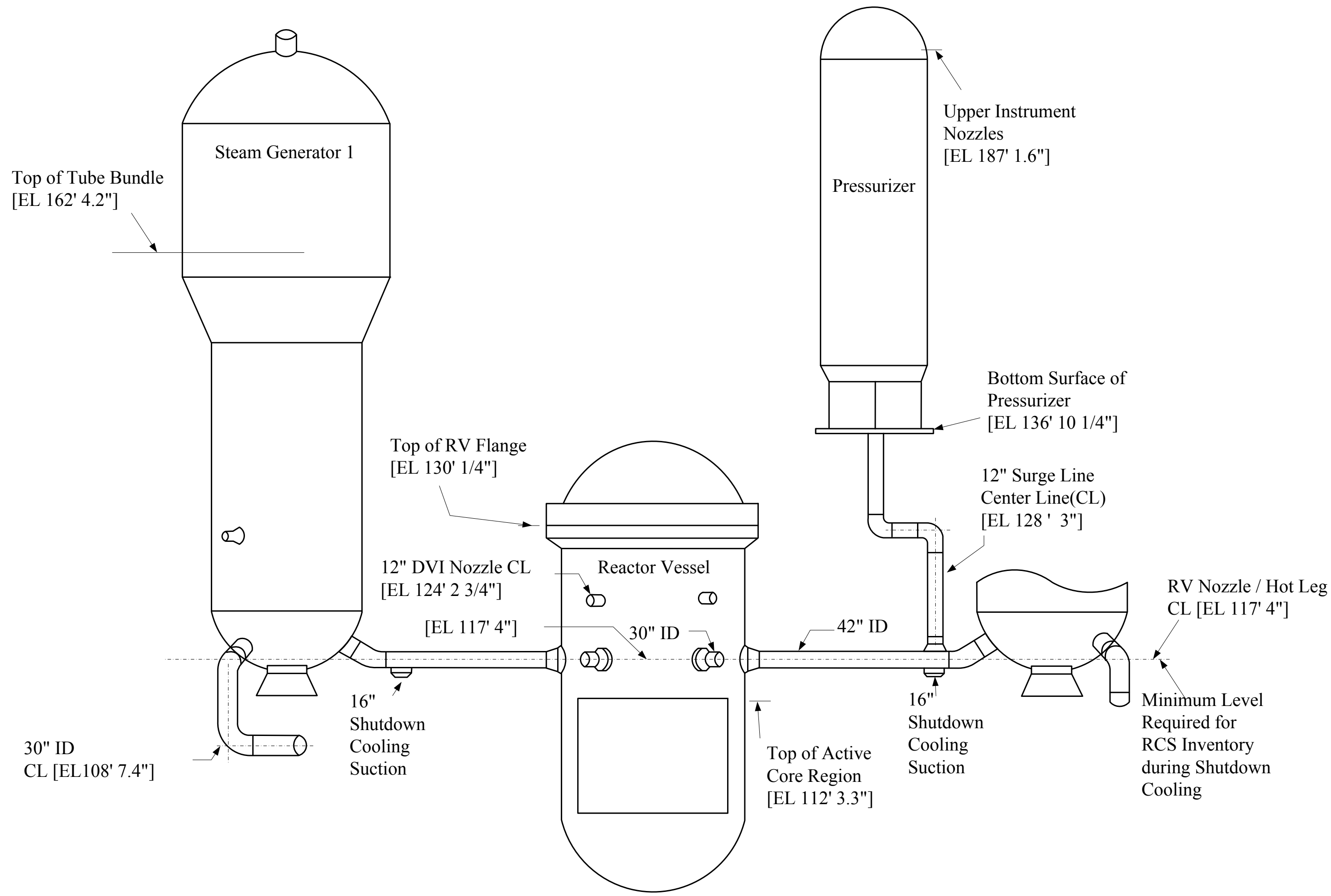

Figure 2.8-1 Reactor Coolant System Arrangement (Elevation)

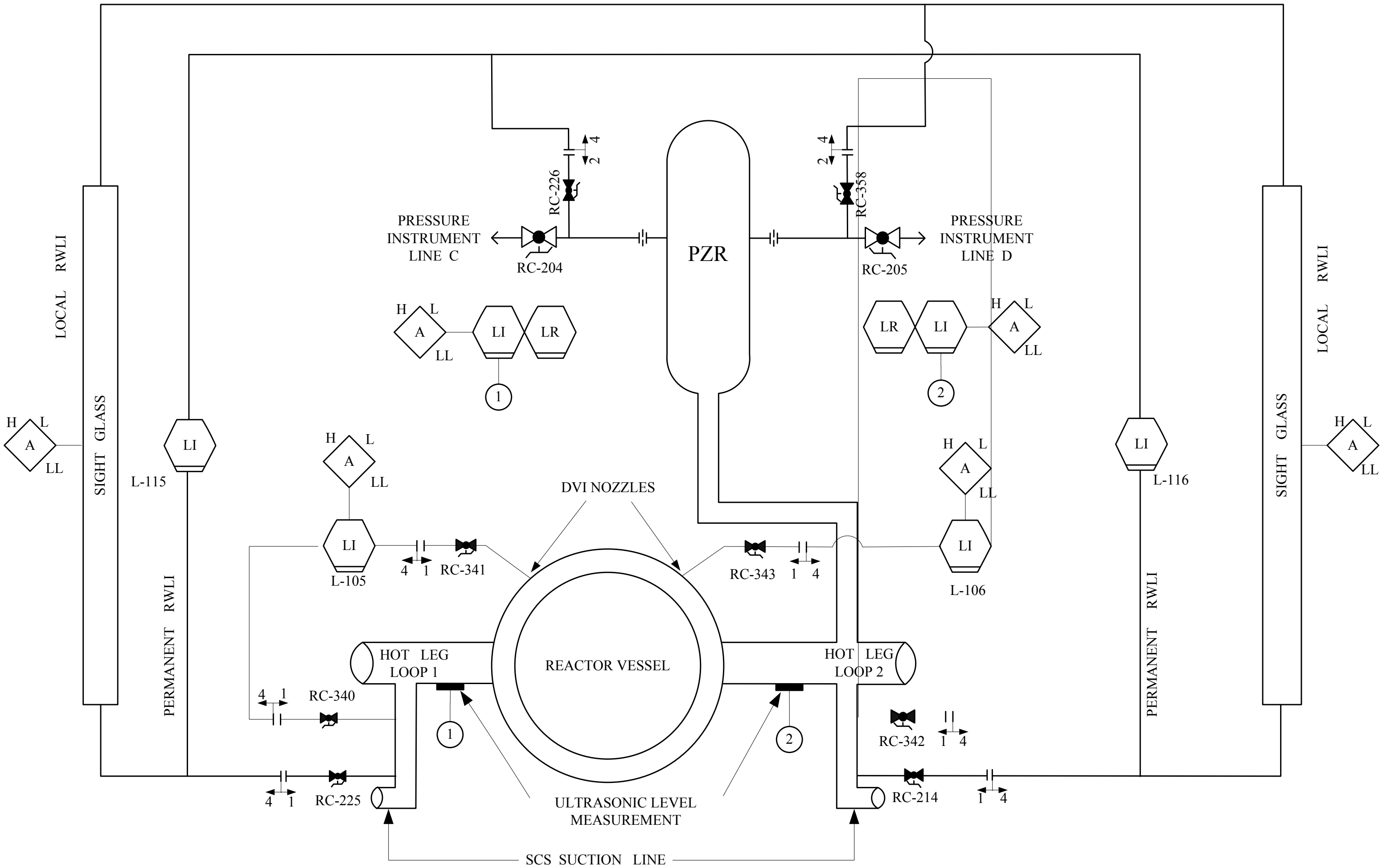

Figure 2.8-2 Refueling Water Level Indication System Schematic Diagram

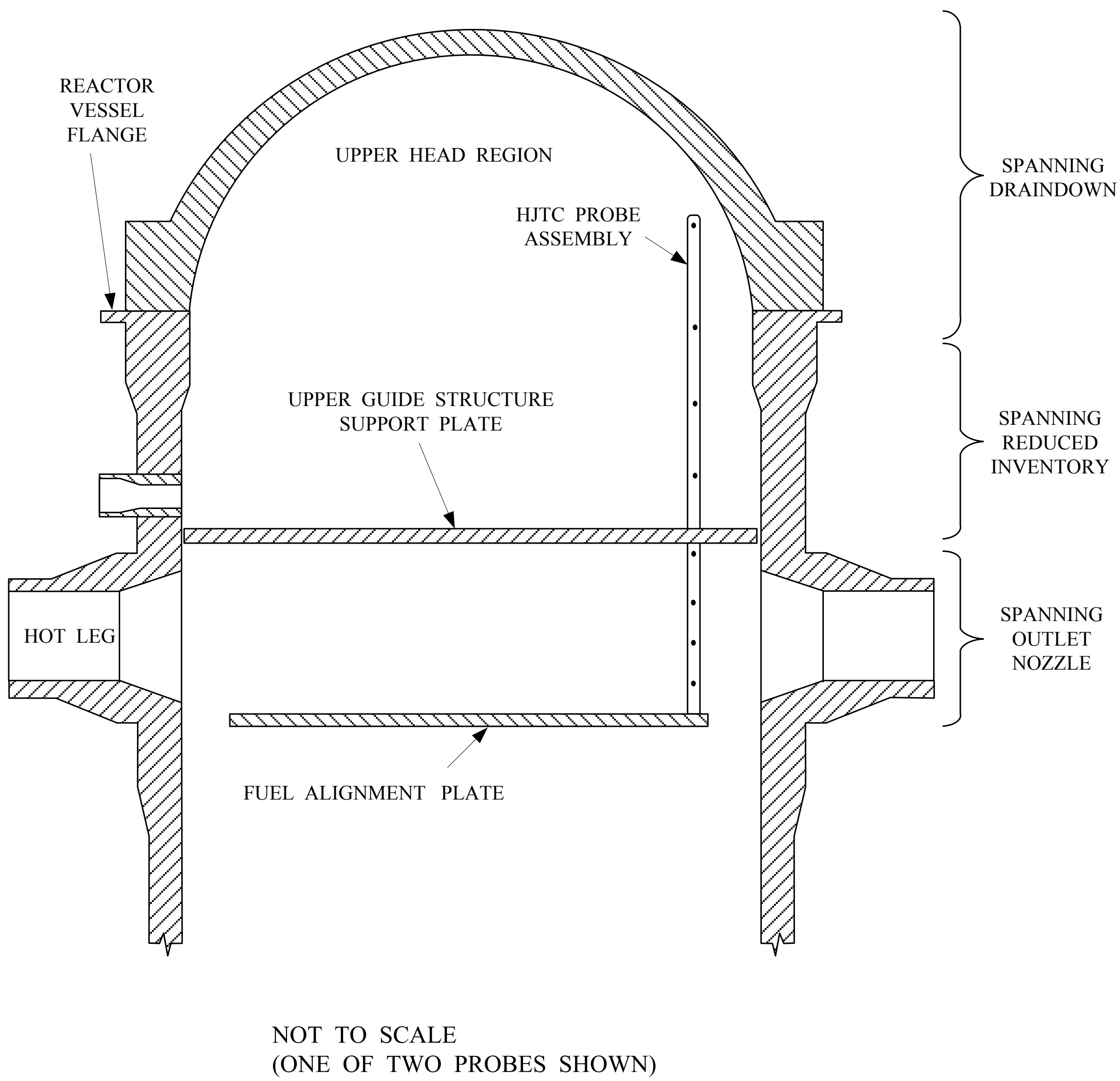

Figure 2.8-3 HJTC Probe Assembly Schematic Diagram

2.9 Emergency Core Cooling System 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, which include:

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

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

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

2.9.2 Acceptance Criteria

The design of the APR1400 in-containment refueling water storage tank (IRWST) and holdup volume tank (HVT) and their associated debris-blocking devices conforms with the requirements of General Design Criterion (GDC) 35 of 10 CFR 50, Appendix A (Reference 17). GDC 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 CS water and emergency core cooling (ECC) water to the HVT and then to the IRWST.

The safety injection system (SIS) meets the acceptance criteria in NUREG-0800 Subsection 6.3, “Emergency Core Cooling System,” Rev. 3 (Reference 18). In particular, Subsection 6.3 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 NUREG-0800, Subsection 6.2.2, “Containment Heat Removal Systems,” Rev. 5 (Reference 19). These criteria address the drainage of CS water and emergency core cooling water to the recirculation suction points (sumps) and the screen assemblies surrounding these suction points. RG 1.82, “Water Sources for Long-term Recirculation Cooling Following a Loss-of-Coolant Accident,” Rev. 4 (Reference 21), 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,” Rev. 1 (Reference 20).

2.9.3 Description

During a postulated accident, water in containment draining back to the IRWST passes through a large trash rack before entering the HVT. Major openings such as hatches and stairwells are also available to return water to the screened entrance to the HVT. The HVT serves as an effective solids trap for high density debris. Lower density debris that makes its way into the IRWST encounters debris screens that filter fine particles from the IRWST to the emergency core cooling (ECC) pump suction. Each of the four SIS pumps has separate IRWST suction lines, and each of the two CSS pumps takes suction from one of these four lines.

a. General

Water introduced into the APR1400 containment from an RCS break or from containment sprays drains into the HVT. This tank serves as a containment sump and is therefore the low collection point in containment. The contents of the tank are directed to the IRWST through the two IRWST spillways (see Figures 2.9-1 and 2.9-2). The IRWST also serves as the single water source of long-term recirculation for emergency core cooling and containment heat removal.

In the APR1400 design, the IRWST does not serve as the containment sump. The IRWST serves as a storage tank for refueling water, a reliable source of clean water for safety injection, and a heat sink for condensing steam discharged from the PZR.

b. Trash rack

Following an accident, water introduced into containment drains to the HVT. Debris that may exist in containment may be transported to the HVT with this fluid. Debris with a diameter of greater than 3.8 cm (1.5 in) 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 about 2.1 m (7 ft) height and about 6.4 m (21 ft) total width. A drain trench exists in front of the 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 helps impede the deposition of debris buildup on the screen surface. Particles that are smaller than the trash rack mesh enter the HVT.

c. Holdup volume tank

The HVT 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 accumulates in the bottom of this tank. The IRWST spillways are located at a high enough elevation to provide reasonable assurance that much of the higher density debris (and debris that tends to sink slowly) settles to the bottom of the HVT before spilling over into the IRWST. Debris that remains in suspension makes its way to the IRWST spillways.

d. Spillway

Multiple spillways are available to return water from the upper containment elevations to the IRWST. The drain pathways are fully redundant to provide reasonable assurance of recirculation capability. The spillways are shown in Figures 2.9-1 and 2.9-2.

e. Debris screen

The fine debris that is introduced into the IRWST is prevented from entering the ECC pump suction by a debris screen. These screens are located at each entrance of the four ECC pump suctions (see Figure 2.9-2). The spillway screens have the capability of removing particles with diameter of greater than 2.3 mm (0.09 in). The screen size is consistent with the screens used in currently operating units.

Blockage of the debris screens is a major concern with respect to recirculation. The APR1400 screens have a vertical orientation to prevent debris from settling on the screen surfaces. This helps in keeping the screens clear. The design takes into consideration the types and quantities of insulation used for the APR1400 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 are also considered. The effective areas of the screens are determined according to the guidelines in RG 1.82, Rev. 4 (Reference 21), Appendix A, “Guidelines for Review of Sump Design and Water Sources for Emergency Core Cooling”.

f. Vortex suppression and NPSH

Protection against air ingestion by SIS pumps is also a major concern with respect to recirculation and is considered in the APR1400 design. The location and size of the suction lines in the IRWST are 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 is calculated conservatively to account for blockage that may result, per RG 1.82, Appendix A. The minimum water level in the IRWST is conservatively calculated to be 26.2 m (86 ft). This water level allows for sufficient NPSH for the CS pumps and SI pumps operating at runout flow. A conservative margin is 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 RG 1.82, Appendix A, yields zero air ingestion at normal pump flow rates and less than 2 percent air ingestion at pump runout flow rates. The IRWST suction lines are also provided with vortex suppressors to aid in minimizing air ingestion by the SIS pumps. The guidelines in RG 1.82, Appendix A, regarding the design of these vortex suppressors are considered.

g. Separation

The arrangement of the IRWST within containment meets the multi-sump requirement of RG 1.82, "Water Sources for Long-Term Recirculation Cooling following a LOCA" (Reference 21). 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 (see Figure 2.13-1) 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 2.9-2) and each of the two CS pumps shares one of these four connections.

h. Inspection

Each screen used in the APR1400 design is provided with an access opening to allow for inspection of the racks or screens. The screens are visually examined periodically to detect any corrosion or structural degradation during refueling outage periods.

i. Insulation

The design considered the types and quantities of insulation used for the APR1400 components, since post-LOCA deterioration of this insulation is the major potential source of debris in containment. The locations of insulation with respect to the HVT and IRWST as well as the possible location of breaks have also been considered.

The type of the insulation used for the piping and equipment except the RV in the containment is fiberglass insulation with stainless steel jacket. It is assumed that the insulation material is low density material (32 to 64 kg/m3 (2 to 4 pounds per ft3) such as NUKON). The potential maximum quantities of insulation debris generated are conservatively estimated as about 42.5 m3 (1500 ft3).

j. Debris screens design

The debris screens are 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 IRWST transient loads due to an open pilot-operated safety relief valve (POSRV).

Consideration is also given to the materials used for the debris screens. Materials are selected to avoid degradation during periods in which the screens are fully submerged.

k. Pressure relief dampers

Overpressure protection for the IRWST is provided by four pressure relief dampers in the IRWST cover. These dampers are self-actuated by overpressure or vacuum in the IRWST. The location of the dampers (equipment numbers 1305 ~ 1308) are shown in Figure 2.9-3. The height of debris curbs surrounding the dampers is high enough to prevent water and other debris from passing through the damper directly to the IRWST pool.

l. ECCS pumps flow test

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. The testing can verify the satisfactory hydraulic performance of the IRWST by running the pumps at runout flow.

2.9.4 Resolution

The design of the APR1400 IRWST and HVT provides reasonable assurance that a clean and reliable source of borated water is available for ECCS recirculation. The arrangement of the IRWST within the APR1400 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 encounters debris screens that filter fine particles from the IRWST to ECC pump suction. Each of the four SIS pumps has separate IRWST suction lines and each of the two CSS pumps takes suction from one of these four lines.

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

Consideration is 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 design. The performance of the design is deemed acceptable with respect to these considerations.

This issue, the potential for loss of flow to the CS and SI pumps during the accident conditions, is resolved by the design features of the APR1400.

TS

Figure 2.9-1 IRWST and HVT Upper Plan View

TS

Figure 2.9-2 IRWST, HVT and Cavity Flooding System Section View

TS

Figure 2.9-3 IRWST and HVT Lower Plan View

2.10 Effects of Pressurized Water Reactor Upper Internals

2.10.1 Issue

Events with the potential for loss of decay heat removal (DHR) have initiated from plant configurations with the RV head removed, the refueling pool filled with water, and the reactor upper internals still in place. Under these conditions, the RV 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 RV head is off and the core and upper internals are in the vessel, one or more of the following conditions are satisfied:

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

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

2.10.3 Description

In Mode 6 configurations with the vessel head removed, loss of shutdown cooling events could be of concern if the upper internals inhibit natural circulation cooling of the core via the heat sink in the upper refueling pool. An analysis to predict the extent of natural circulation flow through the upper guide structure (UGS) and alignment keyholes is described below. Resolution of this issue is based on the results of the analysis.

The entry flow path through alignment keyholes in the core support barrel flange to the upper downcomer from the refueling pool admits very small flow in view of its small flow area and high hydraulic resistance. Flow areas for the down-flow and up-flow at the bottom of the core are calculated from the geometrical relationship and the assumptions of constant density and no-slip condition. Flow areas for the down-flow and up-flow at the UGS plate are assumed to be the same. A Runge-Kutta method to solve ordinary differential equations, based on the principles of conservation of mass, momentum, and energy, is used to determine the transient natural circulation flow after loss of shutdown cooling and the time to saturation at the core outlet.

Key assumptions in the analysis are one-dimensional flow with no transverse momentum and energy exchange, no conduction loss through the vessel, no convection from the surface of the refueling pool, and no heat storage within the bounds of the upper internals. The decay heat generation is conservatively assumed to be constant, corresponding to 2 days after shutdown, with 2 sigma uncertainty. Other simplifications include a uniform refueling pool temperature together with fluid temperature in inner core flow region that actually exhibit a linear variation axially. All metals in the RV and refueling pool are neglected for conservatism.

This analysis indicates that natural circulation flow makes the water in the upper refueling pool available for cooling the core and significantly extends the time required for the core outlet temperature to reach saturation under these circumstances. Fluid temperature versus time variation results as a function of initial coolant temperature are shown in Figure 2.10-1. The curves in the figure for the maximum Mode 6 initial coolant temperature of [ ]TS indicate that the core outlet temperature does not reach saturation until [ ]TS after the loss of DHR. The calculated natural circulation flow rate increases throughout this time period with a value of [ ]TS of the rated design core flow rate at 1 minute after

loss of DHR rising to [ ]TS where the outlet temperature reaches saturation at [ ]TS For an initial coolant temperature of [ ]TS, the time to saturation increases significantly to [ ]TS.

These results make it possible to conclude that the presence of the upper internals does not prevent natural circulation communication with the refueling pool. Depending on the initial fluid temperature, the time to boil exceeds [ ]TS and may last approximately 1 hour. Ample time therefore exists for the implementation of operating procedures to deal with the restoration of forced circulation DHR and/or to begin the process of containment closure, which requires 10 minutes in a Mode 6 configurations.

2.10.4 Resolution

The results described above indicate the availability of a natural circulation flow path through the upper internals in the vessel in the event of a loss of both SCS trains during Mode 6 when the refueling pool is full. The heat removal effected by transference of core-generated decay heat to the upper refueling pool allows plant operators a minimum of at least [ ]TS to terminate the event before the core outlet temperature reaches saturation. Substantially more time is available before possible uncovery of the core. This time period is of sufficient duration that the issue can be considered resolved on the basis of two cases (27.8 °C (100 °F) and 57.2 °C (135 °F) of initial coolant temperature) of the previously stated in Subsection 2.10.2. [ ]TS and [ ]TS are clearly longer than the minimum Mode 6 containment closure time of 10 minutes and sufficiently long to allow plant operators to restore shutdown cooling. In order to provide reasonable assurance that the operators are aware of the potential for loss of DHR during Mode 6 with the upper internals in place and to provide recommended recovery guidelines, procedural guidance is provided to the COL applicant.

TS

Figure 2.10-1 Temperature vs. Time during Mode 6 Natural Circulation

2.11 Fuel Handling and Heavy Loads

2.11.1 Issue

The issue is the potential for damage to fuel and safety related equipment as a result of dropping heavy loads during plant shutdown. Related issues involve the transport of heavy loads in the containment building and the fuel handling area of the auxiliary building are as follows:

- Dropping the integrated head assembly (IHA) and the RV internals
- Accidental release of a fuel assembly
- 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

The two major acceptance criteria of the shutdown heavy load analysis are as follows:

a. For a heavy load drop accident anywhere in the plant, redundant and/or diverse equipment remains operable to provide reasonable assurance that the capability to remove decay heat from the core exists independent of equipment or systems affected by the load drop.

b. Fuel assemblies in the reactor or storage racks remain subcritical during and following postulated load drop accidents.

2.11.3 Description

The acceptance criteria and the relevant APR1400 design features are described in the following subsections.

2.11.3.1 Impacts on the Capability to Remove Decay Heat

Dropping a heavy load during shutdown modes of operation may cause loss of DHR capability. The potential for heavy load drop accidents increases during shutdown modes because of the maintenance activities that occur during shutdown.

Measures that are used to prevent heavy load drop accidents are using safe load paths, following load-handling procedures, training crane and hoist operator, implementing lifting device designs; and inspecting, testing and maintaining load handling equipment as specified in NUREG-0612 (Reference 7). In addition, reasonable assurance is provided that a heavy load drop anywhere in the plant does not impair DHR capability.

The safe shutdown systems and functions needed for DHR are identified as follows:

a. In the APR1400 design, decay heat is removed by the CSS or SCS. The CSS is an alternative to the SCS. Only plant operating Modes 4, 5, and 6 are considered for the shutdown evaluation. During those modes, the RCS pressure and temperature are less than 28.83 kg/cm^2 (410 psig) and 176.7 °C (350 °F), respectively.

b. The RCS pressure control function is available during the shutdown cooling operation because the RCS is pressurized in order to maintain a certain subcooled margin. When the RCS

pressure is low during the shutdown cooling operation with a LOOP, the RCS pressure can be maintained by stopping the SC pump, which allows the RCS to be pressurized by the decay heat. When the RCS pressure is high, the RG vents are used to depressurize the RCS.

c. The major support systems and equipment that are needed for the operation of the safe shutdown systems for the DHR are as follows:

(1) Emergency diesel generator

(2) Component cooling water

(3) Essential service water

(4) Essential chilled water

(5) Class 1E electrical distribution

d. The major instrumentation needed to monitor the operation of the safe shutdown systems needed for DHR are as follows:

(1) PZR pressure

(2) Source range neutron flux monitor

(3) SC/CS HX inlet/outlet temperature

The overhead load-handling systems, if dropped, that can damage the DHR-related equipment listed above include the containment polar crane, the auxiliary building trolley beams Nos. 5, 6, 7, 8, 11, 12, 13, 18, 19 and 30, and the gantry crane. Table 2.11-1 shows the load-handling equipment in the APR1400 along with the equipment that could be damaged by a load drop. Among the overhead load-handling systems in the containment building, the containment polar crane is considered as the only load-handling system that can affect DHR capability. If the other load-handling systems listed above are damaged by postulated drop, only one train of equipment associated with the equipment under maintenance is affected. Therefore, no further review is required. The containment building contains redundant trains of DHR systems and equipment. The areas and loads in the containment that can affect DHR capability are as follows:

a. The two channels of PZR pressure and level instrumentation and associated cabling in the containment building are routed close to each other at El. 136 ft 6 in. A load drop by the containment polar crane above the operating floor would not damage this instrumentation because the floor at El. 156 ft 0 in is made of concrete.

b. The redundant trains of RG vent valves from PZR and RV are located on the operating floor of the containment building. At least one train of vent paths from the PZR and RV are available for RCS pressure control. In addition, both trains are protected from a potential load drop since the load drop may cause loss of RCS pressure boundary. The load-handling system with concerns associated with this issue is the containment polar crane. Among the loads listed in Table 2.11-1, the IHA and the main/aux hook load blocks have potential to damaging the RG valves and piping. Therefore, reasonable assurance is provided to avoid carriage of these loads above or near the RG valves and associated piping to the maximum extent possible. This is confirmed at the final design stage.

Based on the above review, a load drop in any location during the shutdown modes of operation dose not impair DHR capability. However, inside containment, there is potential to jeopardize DHR capability due to

a load drop in the containment during shutdown operation. Therefore, appropriate load handling procedures and load-handling paths are established as noted above.

2.11.3.2 Impacts on the Capability to Maintain Subcriticality Condition

2.11.3.2.1 Fuel and Load Handling within the Containment Building

The transport of heavy loads within the containment building and the fuel handling area of the auxiliary 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 provide reasonable assurance that heavy loads are restricted to pre-assigned travel zones. Equipment interlocks and procedures are also used to provide reasonable assurance that load transport is accomplished in a predictable manner.

Issues associated with the transport of heavy loads within the containment building include movement of the IHA, reactor internals, 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 IHA as provided in APR1400 DCD Tier 2 Subsection 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 IHA as it is raised from the reactor. This provides reasonable assurance that the containment building remains isolated from the spent fuel pool building during transport of the IHA. In addition, by isolating the containment building from the spent fuel building, the spent fuel pool is protected against drain down that could occur as a result of a postulated drop accident.

In the case of a postulated IHA drop, though the RV and internals may sustain damage, the RV remains filled and the fuel remains covered and in a subcritical configuration. A drop accident involving the internals would be less severe than the postulated IHA drop accident.

Travel paths for the heavy loads within the containment building except the IHA, leading from the RV to the respective storage stands, are arranged so the transported loads do not pass directly over the in-core instrumentation (ICI) seal table (see Figure 2.11-1). But even if it is postulated that portions of these structures affect 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 would remain at the flange level. In addition, if the reactor cavity pool seal is damaged to the extent that there is significant pool drainage, the RV would remain filled.

The refueling machine is structurally designed to withstand the effects of design basis seismic motions. In addition, this machine is provided with interlocks that 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 cannot pass over the top of the ICI seal table. This precludes the possibility of a load drop accident involving a fuel assembly falling onto the ICI seal table. In addition, 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 it withstands 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 that would provide reasonable assurance that it remains submersed in water. The assembly may be inserted into the RV or lowered into the deep end of the refueling pool, adjacent to the fuel transfer system. In order to inform the operator of equipment capabilities and of available response time for emergency actions if a cavity seal failure occurs during

refueling operations, procedural guidance is provided and, is included in the appendix on shutdown in the EOG for the APR1400 design.

2.11.3.2.2 Fuel and Load Handling within the Fuel Handling Area

As with the containment building, special consideration is given to the transport of heavy loads and fuel assemblies in the fuel handling area of the auxiliary building. Restrictions on the transport of heavy loads over fuel storage racks, and movement of the fuel shipping cask also apply (see Figure 2.11-2).

Transport of the fuel shipping cask within the fuel handling area of the auxiliary building is accomplished using a special high capacity hoist. The cask is transported using a staggered lift from the cask decontamination fit to the cask loading pit, 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 are designed to withstand a postulated cask drop accident.

The spent fuel pool is connected to the cask loading pit by a gate, that is 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 remains submerged following a postulated pool drain down through the gate.

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

New fuel enters the fuel handling area of the auxiliary building through a designated unloading area. It is handled and transported to new fuel storage racks by an intermediate capacity hoist. The lift height of the hoist is restricted in order 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 design of the fuel handling machine used in the fuel building is similar to the refueling machine and is structurally designed to withstand seismic excitations. It is also provided with interlocks to control the movement of fuel within the pool.

New fuel and spent fuel storage racks are both 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 result of an evaluation of the consequences of dropping a spent fuel assembly in the spent fuel pool are presented in the APR1400 DCD Tier 2 Subsection 15.7.4. The results show 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 APR1400 by the equipment design and building layout, which satisfy the applicable acceptance criteria and provide physical limitations to movement and by the administrative limitations in Chapter 9 of the APR1400 DCD Tier 2. The design and layout of the APR1400 plant incorporate features that have proven to be successful in other plants. Appropriate improvements that provide an additional margin of safety during handling operations have been introduced.

Table 2.11-1

APR1400 Overhead Load Handling Systems

A. Containment Building

System Designation	System Elevation	System Capacity	HEC (Note)	System Components Lifted	System Components Underneath	Impact Area
Containment polar crane	241'-0"	408/54 (M ton) (900,000 /120,000 lb)	1&3	Integrated Head Assembly	- RV - CEDM and CEDM AHU - PZR, RCP - Piping for CV and SI	- RV - Top of PZR enclosure wall
			3	RV head assembly and lift rig assembly	- RV - I&C seal table - UGS lift rig	- RV/UGS laydown area - RV head storage area
			3	Single stud tensioner and lifting frame	- RV - Fuel transfer system	- RV/CSB laydown area - fuel transfer system - MST laydown area
			3	RV closure head lifting frame assembly	- RV - ICI seal table - UGS lift rig	- Refueling pool - RV head storage area - MST laydown area
			3	UGS and lift rig assembly	- RV	- RV/UGS laydown area
			3	CSB and lift rig assembly	- RV	- RV/UGS laydown area
			3	CEA/ICI transport container	- RV	- CEA elevator/ transport container rack
			2&3	ICI cable tray assembly	- ICI seal table	- ICI seal table
			2&3	ICI instrumentation holding frame	- ICI seal Table	- ICI seal table
			2&3	RCP Motors	- Fuel transfer system - Piping for MS system	- Fuel transfer system
			2&3	RCFC Motors	- Fuel transfer system - Piping for MS, WL system	- Fuel transfer system
			3	Main/Aux. hook load blocks	- RV - SG - RCP, PZR, RCFC - SITs, ICI seal table - Piping inside containment	- All the zone within the main/aux hook radius
			3	CEA Rod shipping container	- Fuel transfer system	- Fuel transfer system
			2&3	Fuel transfer tube blind flange	- Fuel transfer system	- Fuel transfer system
Refueling Machine	156'-0"	1180 kg (2600 lb)	3	Spent fuel and handling tool New fuel and handling tool	- RV - Fuel transfer system	- Fuel transfer system - Refueling pool

Table 2.11-1 APR1400 Overhead Load Handling Systems (Con't)

B. Fuel Handling Area in Auxiliary Building

System Designation	System Elevation	System Capacity	HEC (Note)	System Components Lifted	System Components Underneath	Impact Area
Fuel Handling Area Overhead Crane	190'-0"	113.4/4.5 M ton (250,000 /10,000 lb)	2&4	New fuel container	- Piping for AS, WL, FP System	- Truck bay - New fuel container storage area
			2&4	New fuel assembly	- Piping for AS, WL, FP System	- New fuel container storage area - New fuel inspection area
			2&4	Spent fuel cask	- Piping for WL, FP system	- Cask loading pit - Cask decontamination pit - Truck bay
Spent Fuel Handling Machine	156'-0"	907 kg (2000 lb)	2	Fuel handling area emergency exhaust ACU /fuel handling area normal exhaust ACU	- Piping for AS, WL, FP System	- New fuel storage area - Truck bay
			2&4	Main hoist load block	- New fuel assembly - Spent fuel cask - Piping for AS, WL, FP system	- Cask loading pit - Cask decontamination pit - Truck bay
			4	Spent fuel assembly and handling tool	- Spent fuel and racks - Fuel transfer system - Piping for FC system	- Spent fuel pool - Fuel transfer system - Cask loading pit
			4	New fuel assembly and handling tool	- Spent fuel and racks - New fuel elevator - Fuel transfer system - Piping for FC system	- New fuel inspection area/ New fuel storage pit/New fuel elevator spent fuel pool
Hoist Beam No. 1 Monorail	156'-0"	5 M ton (11023 lb)	4	CEA/ICI transport container and handling tools	- Fuel transfer system - Spent fuel - Piping for FC system	- Fuel transfer system - Spent fuel pool - Cask loading pit
			4	Failed fuel storage canister assembly	- Spent fuel and racks - Fuel transfer system - Piping for FC system	- Spent fuel pool - Fuel transfer system
			4	Spent fuel inspection device Spent fuel pool swing gate	- Spent fuel and racks - Fuel transfer system - Piping for FC system	- Spent fuel pool - Cask loading pit - Cask decontamination pit

Notes for HEC (hazard elimination category)

1. Special station procedures for heavy loads or operation.
2. No safe shutdown equipment exists below the overhead handling system.
3. Refer to Subsection 2.11.3.2.1 for detail and the overhead handling system.
4. Refer to Subsection 2.11.3.2.2 for detail and the overhead handling system.

TS

Figure 2.11-1 Equipment Removal Path of Containment Building Operating Floor (El. 156')

TS

Figure 2.11-2 Equipment Removal Path of Auxiliary Building Fuel Handling Area (El. 156')

2.12 Potential for Draining the Reactor Coolant System

2.12.1 Issue

The issue is the risk of losing primary coolant from the RCS during Modes 2 through 6 (startup, hot standby, hot shutdown, cold shutdown, and refueling). The safety significance of draining the coolant from the RCS during shutdown is that such an event can lead directly to voiding in the core and to eventual core damage. Draining the RCS may also lead to a loss of shutdown cooling capability that could lead to core uncovery.

2.12.2 Acceptance Criteria

The criteria used to evaluate the adequacy of the APR1400 design with respect to the potential for draining the RCS are prevention, detection, and mitigation. Prevention is the preferred criterion, but detection and mitigation are provided in some instances.

2.12.2.1 Prevention Acceptance Criteria

The prevention acceptance criteria are as follows:

a. The design prevents or inhibits the draining through the use of isolation valves, interlocks, and system alignment restrictions during the various modes of plant operation.

b. The design minimizes the potential for component failure, inadvertent action, or human/operator error that results in the rapid draining of the RV. Redundant components are provided as appropriate. The design provides instrumentation, overview displays, and alarms to clearly supply the operator with equipment status that is specific to shutdown modes.

c. Technical Specifications and procedural guidance are provided to the operator to assist in identifying plant conditions and configurations in Modes 2 through 6 that could result in a potential primary coolant drainage event.

2.12.2.2 Detection Acceptance Criteria

The design provides the capability to detect and monitor a drainage event. Drainage pathways that could lead to a loss of shutdown cooling or core uncovery are considered. Appropriate instrumentation, displays, and alarms are provided. The adequacy of APR1400 instrumentation to detect drainage events are confirmed.

2.12.2.3 Mitigation Acceptance Criteria

The mitigation acceptance criteria are as follows:

a. The design provides the capabilities to mitigate the loss of primary coolant from the RCS including isolation of a drain path and provide a source and path for sufficient makeup.

b. Technical Specifications and procedural guidance are provided to identify potential makeup water injection sources and paths in the event a drainage path occurs. Recovery actions are specified.

2.12.3 Description

Primary coolant can drain from the RV because there is a path directly from the RCS or by way of paths

through systems interfacing with the RCS. Plant modes of operation characterize the potential for draining the RCS (the alignments and conditions, such as pressure, temperature, and flow that can exist within and between the RCS and interfacing systems).

The causes of RCS draining (initiators) are divided into two groups. The first group includes components and equipment that fail to operate as intended, which could be the result of equipment malfunction (e.g., stuck open relief valve). The second group includes operator error such as misoperation of valves or pumps.

The key factors in the probability and consequence (i.e., risk) associated with initiators are the plant configuration, the ability to respond to the event, and the characterization of the initiator. Consideration of these factors can aid in the development of procedures for prevention, detection, identification, and mitigation or termination of such events. The key risk factors and subfactors are presented in Table 2.12-1.

Plant configuration is most notably characterized by system alignments with the RCS during various modes of shutdown operation. These alignments define the possible drain paths from the RV. The initial water level in the RCS influences operator response time. The use of temporary seals in the RCS (e.g., nozzle dams) and interfacing systems during maintenance and refueling activities can be the source of the initiator (e.g., the dam fails) or preclude mitigating actions by defeating injection paths (e.g., if an in-place dam blocks circulation).

Plant configuration is also characterized by the availability of mitigating systems. Mitigating systems must have a sufficient source of borated water and the ability to deliver the water to the RV at a rate that is equal to or greater than the rate at which water is draining from the RCS. Finally, maintenance activities must not preclude mitigating systems from being used to respond to the event.

Another risk factor is the ability to quickly determine that a loss of coolant is occurring (detection) and the source of the loss (identification).

The analysis presented in this subsection was conducted by examining design drawings and P&IDs for the APR1400 to define potential drain paths from the RV and to consider these drain paths in the context of the risk factors listed in Table 2.12-1. A potential drain path is any opening in the RCS (seal, manway) or any interfacing system piping path that can take primary coolant away from the RCS. Many openings and interfacing systems are designed to allow fluid to leave the RCS or an interfacing system (e.g., relief valves) or to circulate primary coolant for letdown, purification, charging (chemical and volume control system (CVCS)), sampling system (SS) activities, or SCS.

A shutdown risk drain path results in an unplanned loss of primary coolant from the RCS with the accompanying lowering of the water level in the RV. The path could be short (seal leakage) or relatively long (via piping of an interfacing system). The driving head for the draining flow depends on the relative conditions between the coolant in the RV and the conditions at the end of the drain path. The conditions include RCS pressure, elevation head, pump head, and back pressure.

For purposes of this analysis, a shutdown risk drain path is categorized as major or minor. A major drain path is one that, based on analysis for certain specified plant configurations, could result in a rapid drain flow rate. The preferred recovery is to isolate the drainage source before the RCS water level reaches the break level and to add makeup to the RCS. It is possible, however, for such a major path to drain the RCS to the bottom of the hot-leg elevation too rapidly (i.e., before the operator is able to take mitigative action). Such an occurrence could result in a loss of SCS due to insufficient water level. The identification of major drain path does not imply that such a path is likely or probable for the APR1400, but only that it is possible and that its potential consequence requires special attention to be directed to prevention and recovery procedures.

Minor drain paths are those that could result in a drain flow rate that can be compensated for by using available makeup systems or are otherwise insignificant. A drain path that is categorized as minor requires no further action.

Drain paths that are categorized as major require the procedural guidance that aids the operator in avoiding these paths and providing the operator the means and guidance to recover from such an event. The means to detect, mitigate, and recover from such postulated major drainage events are described in Subsections 2.3, 2.4, and 2.8.

2.12.3.1 Potential Drain Paths Directly from the Reactor Coolant System

Potential drainage paths directly from the RCS are associated with the RCPs, SGs, ICI seals, and the reactor cavity seal. The reactor cavity seal leakage is not an actual RCS leakage, but it is a form of inventory loss during Mode 6. Table 2.12-2 identifies the major paths (i.e., paths with potential to rapidly drain the RCS to a critical water level) and minor paths (i.e., paths that can be controlled by available systems).

2.12.3.2 Major Drain Paths Directly from the Reactor Coolant System

The failure of temporary SG dams and the opening or leakage through SG manways could lead to a rapid loss of primary coolant and perhaps to an RCS water level at the bottom of the hot legs. The APR1400 design includes a requirement to establish a mid-loop vent pathway before operating in reduced inventory. When opened to the containment atmosphere, the pathway provides sufficient venting capacity to prevent RCS pressurization and subsequent dam failure. A description of SG nozzle dam integrity and guidance on the safety and risk aspects of nozzle dam installation timing are provided in Subsection 2.3.3.3. The APR1400 procedural guidance precludes manway openings that would lead to a loss of reactor coolant.

2.12.3.3 Minor Drain Paths Directly from the Reactor Coolant System

The APR1400 reactor coolant pump (RCP) has three stage seals, and each seal is capable of operating at full RCS pressure. However, there is potential for the RCP suffer some leakage from pump seals during any mode. Seal leakage is detected by an RCP-controlled bleed-off flow alarm (F-156, -166, -176, -186). The pump that is the source of leakage can be identified. Drainage from RCP seals is manageable and can be compensated for by available systems.

The APR1400 ICI system design uses instrument tubes that terminate in the refueling cavity at an elevation that is several feet above the RV flange. The APR1400 design does not use temporary thimble tube seals. Even so, according to Section 6.7.2 of NUREG-1449 (Reference 2), there is concern that evolutions could exist that would provide a potentially significant flow path between the bottom of the RV and the top of the seal table, particularly if the RCS is pressurized. Table evolutions are prohibited by procedural guidance (see Subsection 2.3.3.5 of this report) while the vessel head is on and mid-loop evolutions are in progress, thus preventing seal leaks. An evaluation of heavy load handling for the APR1400 relative to the ICI seal table is presented in Subsection 2.11. Travel paths for the closure head and internals from the RV to the respective storage stands are arranged so that the transported loads do not pass directly over the ICI seal table. This precludes the possibility of a load drop accident falling on the ICI table.

An SG can leak primary coolant through tube failure during any mode when the RCS is pressurized. An evaluation of SG tube rupture initiated in a shutdown mode is presented in Subsection 3.6.3. The leakage from a ruptured tube during shutdown is minor relative to reaching an RCS water level at the bottom of the hot legs.

Reactor cavity seal failure could be postulated during Mode 6 when the refueling pool is full. Although such a drainage could be a concern relative to any fuel being transported, the drainage would be self-

limiting to the level of the RV flange, and the SCS would be uninterrupted.

2.12.3.4 Potential Drainage Paths through Interfacing Systems

Potential RCS interfacing system drainage paths from the SCS, the safety injection system (SIS), the chemical and volume control system (CVCS), and the sampling system (SS) are illustrated in Figures 2.12-1, 2.12-2, 2.12-3, and 2.12-4. The originating points of the drainage paths are located on the RCS. The paths include piping, valves, pumps, HXs, and orifices. In the figures, the Greek symbol "φ" represents the piping nominal diameter. Inside containment (IC) and outside containment (OC) are used to describe whether a path segment is inside or outside the containment. The "letter-number" number designation in the boxes generally refers to system valves that are motor operated, manually operated, check, or relief. The letter "F" followed by a number refers to a flow measuring device. The paths represented by the darker/bold lines identify major drain paths. The end points of these paths are various tanks or systems assumed to be open for maintenance.

An assessment of these paths related potential flow rates and the time it would take for the RCS water level to reach the bottom of the hot legs.

2.12.3.5 Potential Drainage Paths from the Reactor Coolant System through the Shutdown Cooling System

This subsection is specific to train 1 but also applies to train 2.

The drainage path from the RCS through the SCS represents one of the two normal shutdown cooling trains. As long as this path is isolated from the RCS during Modes 2, 3, and part of 4, loss of coolant from the RCS through the SCS is prevented. While the SCS is in operation during part of Mode 4 and during Modes 5 and 6, there exists potential paths from the SCS through which the primary coolant passing through the SCS could be drained. The primary initiators to open a drain path are an overpressurization leading to the lifting of a relief valve (which fails to reseat) and/or operator error opening a valve or series of valves to an open system. A valve failure is another possible initiator. An open system is a system that has been drained for maintenance activities.

The SCS suction piping contains two motor-operated isolation valves (SI-651 and SI-653) in series inside the containment (see Figure 2.12-1). These valves are closed during Modes 1, 2, and 3. There is also a motor-operated valve (SI-655) outside the containment. All three valves can be operated from the MCR and have position indication in the MCR. An alarm exists to notify the operator if the two motorized valves inside containment are not fully closed coincident with high RCS pressure. Operator actions require that the RCS be depressurized below the maximum pressure for SCS operation in order to clear the permissive SCS interlock. Therefore, drainage from the RCS through the SCS to interfacing systems in the normal flow path direction is prevented during Modes 2 and 3. Flow occurs through the SCS train during part of Mode 4 and throughout Modes 5 and 6.

The potential drain paths from the SCS, as shown in Figure 2.12-1, are summarized in Table 2.12-3. The 23 paths shown in Figure 2.12-1 are divided into groups with similar drain path characteristics.

Group A through F paths are minor paths and include relief valves, paths through small piping (1/2 in to 3 in outside diameter [OD]) to assumed "open" systems, SC pump seal leakage, SC mini-flow HX tube leakage, SC HX tube leakage, and a postulated path to the SITs.

The paths of Groups A through F, if they occurred and led to a loss of primary coolant, could be mitigated with available equipment and makeup sources during Modes 4, 5, and 6. The discharge would be slow enough that a loss of primary coolant to the bottom of the hot legs is not likely to occur before detection and mitigation have been accomplished. No new design features, Technical Specifications, or procedural guidance are identified for the paths in these groups.

Groups G through I represent major paths that, if established, could result in a rapid loss of reactor coolant during Modes 5 and 6. Depending on the availability of equipment and systems to perform mitigative action during these modes, such a rapid discharge may lead to an RCS water level at the bottom of the hot legs with a concurrent loss of both SCS trains. Procedural guidance to aid the operator in addressing these paths is specified in Subsection 2.1. The recovery action for this scenario is described in Subsection 2.4.3.1.

2.12.3.6 Potential Drainage Paths from the Reactor Coolant System to the Safety Injection System / Shutdown Cooling System from Direct Vessel Injection Nozzles 1A, 2A, 1B and 2B

The potential drain paths from the RCS through the SIS/SCS, originating at the DVI nozzles, are shown in Figure 2.12-2.

The potential drain paths presented in Figure 2.12-2 are reverse paths relative to the normal safety injection flow direction. As a result, in order to assume RCS drainage from the DVI nozzles, it is postulated in this analysis that multiple failures of check valves occur.

The DVI nozzles in APR1400 are located 2.1 m (6.9 ft) above the hot leg and cold legs. Therefore, loss of coolant through a DVI nozzle would be self-limiting. The level of primary coolant would stabilize at the level of the DVI nozzle and would not result in a loss of shutdown cooling. No new design features, Technical Specifications, or procedural guidance are identified for these paths associated with flow from DVI nozzles.

2.12.3.7 Potential Drainage Paths from the Reactor Coolant System to the Chemical and Volume Control System

The potential drain paths from the RCS through the CVCS are shown in Figure 2.12-3. The potential drain paths presented in Figure 2.12-3 are the normal letdown, charging, RCP seal injection, RCP seal leak off, RCP seal bleed off, and drain paths associated with the CVCS design. A major opening in the letdown or charging line needs to occur for any appreciable drainage.

The paths defined for the CVCS, if established, would be manageable with available makeup sources during Modes 2, 3, 4, 5, or 6. This discharge would be slow enough that a loss of primary coolant to the hot leg level is not likely to occur before detection and mitigation have been accomplished. No new design features, Technical Specification, or procedural guidance are identified for paths associated with the CVCS.

2.12.3.8 Potential Drainage Paths from the Reactor Coolant System to the Sampling System

The potential drain paths from the RCS through the SS are shown in Figure 2.12-4. The potential drain paths presented in Figure 2.12-4 are normal sampling paths. A major opening in the sampling lines would need to occur for a net loss of primary coolant to occur.

The paths defined for the SS, if they occurred, would be manageable with available makeup sources during Modes 2, 3, 4, 5, or 6. The discharge would be slow enough that a loss of primary coolant to the hot leg level is not likely to occur before detection and mitigation have been accomplished. No new design features, Technical Specifications, or procedural requirements are identified for paths associated with the SS.

2.12.4 Resolution

The shutdown risk issue of the potential for draining the APR1400 RCS is resolved primarily by design

features, Technical Specifications, and procedural guidance to prevent a drainage event from occurring and to allow the operator to recover in a timely manner if such an event occurs.

Most of the potential paths that were reviewed were judged to be minor such that the drain flow rate could be compensated using available detection and mitigating systems or were otherwise insignificant. APR1400 design features, Technical Specifications, and procedural guidance are sufficient for such paths.

An examination of the potential drainage paths for various APR1400 plant arrangements and operating configurations has provided candidate paths that if assumed to be opened, could lead to a rapid loss of primary coolant. The candidate paths involve primarily those opened by misoperation or misalignment of one or multiple valves by the operator. The importance of such potential major drainage paths to a shutdown risk scenario has led to the specification of procedural guidance (see Subsection 2.1) to aid the operator in addressing these paths. Procedural guidance is provided to the operator via the emergency operating guidelines in order to provide reasonable assurance of the availability of the SCS, CSS, and boric acid makeup pumps to respond to a reactor cavity seal failure event.

Table 2.12-1

Factors Which Affect the Risk Associated with an Initiator

Key Factor	Subfactor
Plant Configuration	• System alignment with the RCS. • Initial water level in the RCS. • Availability of mitigating systems. • Use of temporary seals (e.g., nozzle dams)
Ability to Respond	• Detection of a draining event. • Identification of the initiator. • Termination or mitigation of the event.
Characterization of the Initiator	• Probability of the initiator. • Rate of drainage from the RCS

Table 2.12-2

Potential Drain Paths Directly from the Reactor Coolant System

Drain Path	Drain Detail
Major Drain Paths	• Steam Generator Nozzle Dam Failure • Steam Generator Manway Opening
Minor Drain Paths	• Reactor Coolant Pump Seal Leakage • ICI Seal Table/Housing Leakage • Steam Generator Tube Rupture • Reactor Cavity Seal Leakage

Table 2.12-3

Grouping of Primary Coolant Drainage Paths from the SCS

Group	Paths (Figure 2.12-1)
Minor Drain Paths	
A. Thermal Relief Valve Discharge	1, 3, 14, 15, 16, 20, 22
B. Paths to the Sampling System	9, 19
C. Paths to the CVCS	12
D. Paths to the CCWS	7, 8, 11
E. Paths to SIT #1	17
F. Paths to IRWST, RDT, and EDT through manual valves and small piping	18, 23, 24, 25
Major Drain Paths	
G. Paths through motor operated valves and large piping to the IRWST	4, 13
H. SCS suction relief valve discharge	2
I. Paths through large motor operated valves to CS pump suction and discharge	6, 10

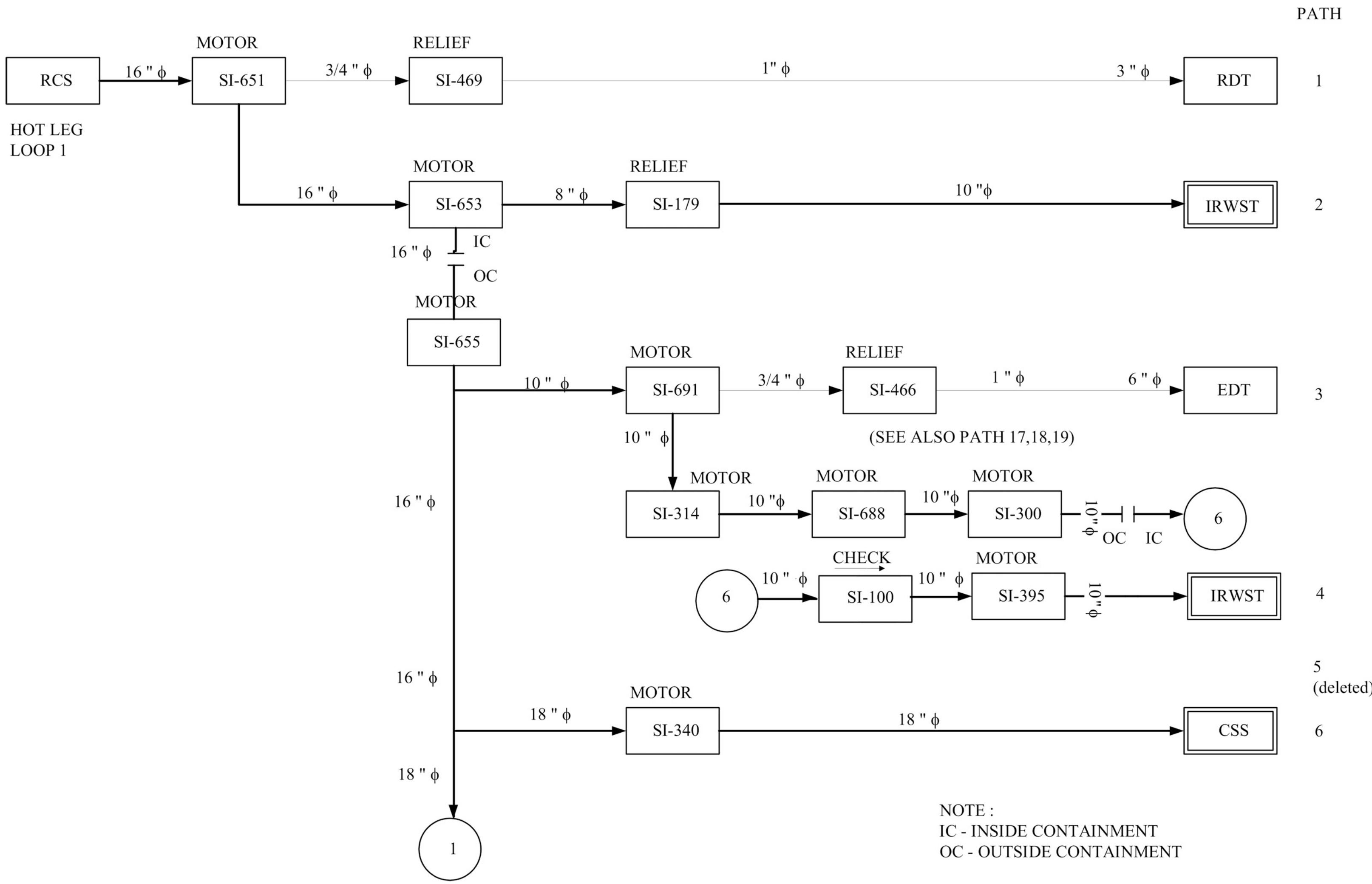

Figure 2.12-1 RCS Drain Paths through SCS Train 1 (1 of 5)

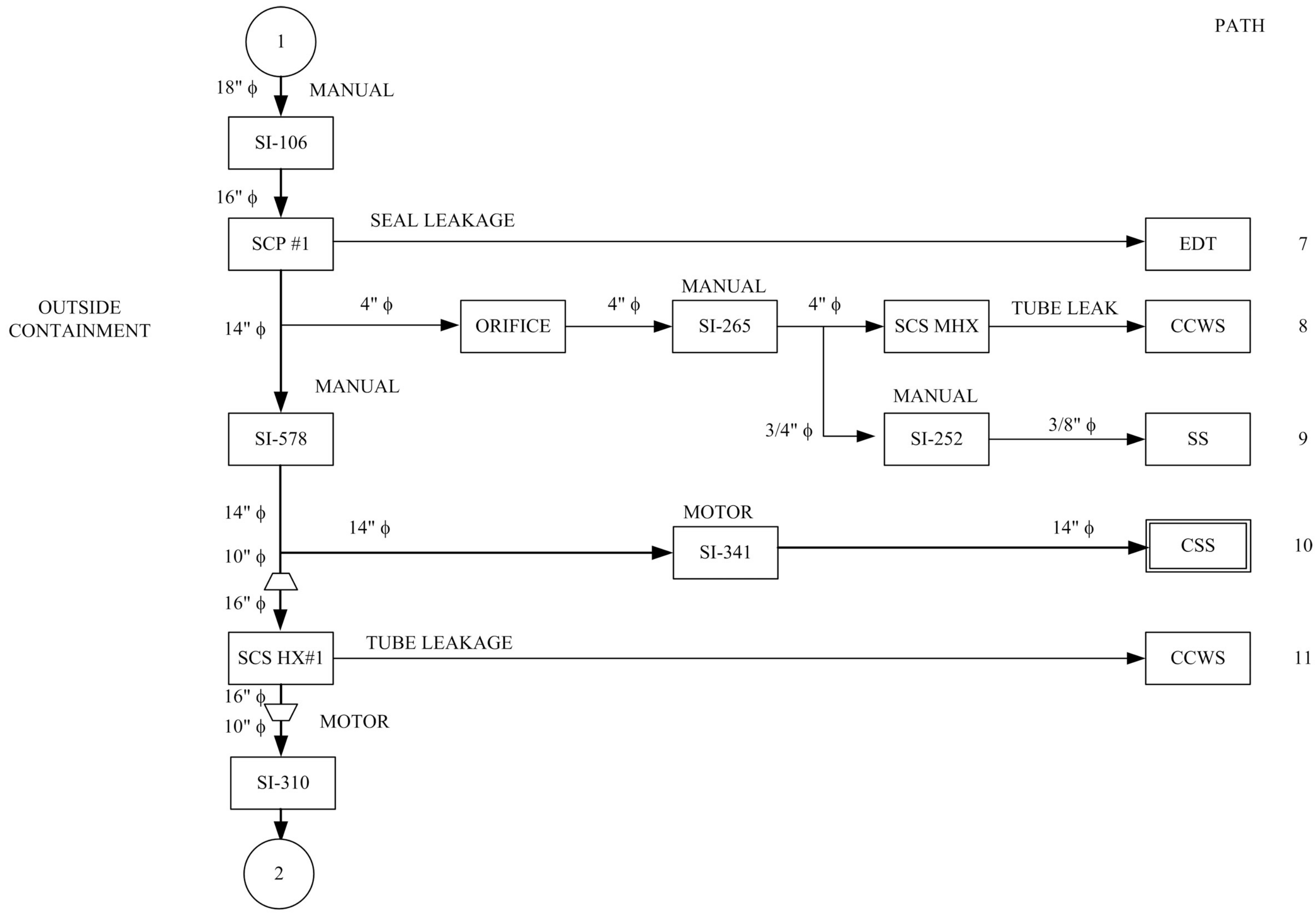

Figure 2.12-1 RCS Drain Paths through SCS Train 1 (2 of 5)

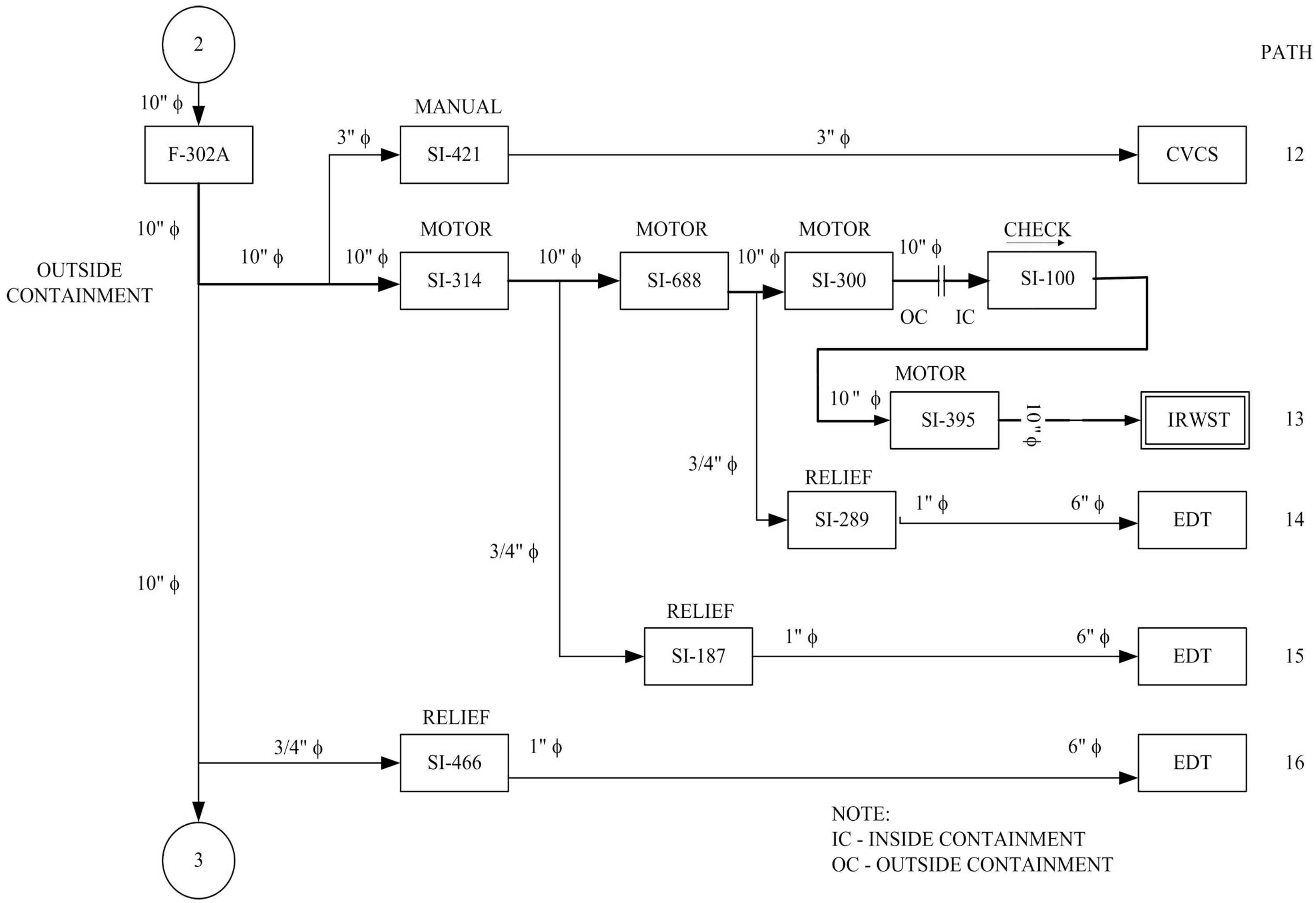

Figure 2.12-1 RCS Drain Paths through SCS Train 1 (3 of 5)

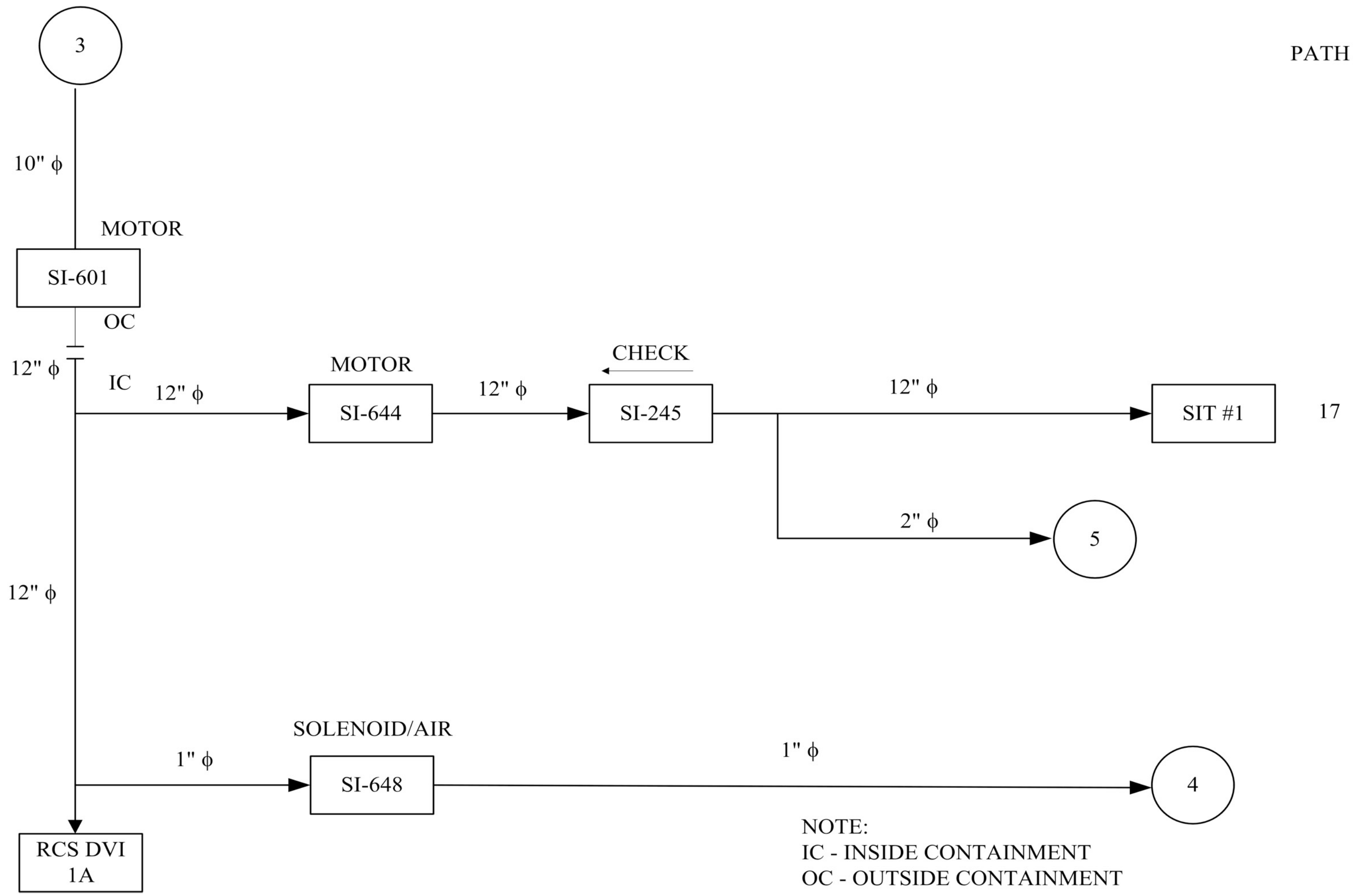

Figure 2.12-1 RCS Drain Paths through SCS Train 1 (4 of 5)

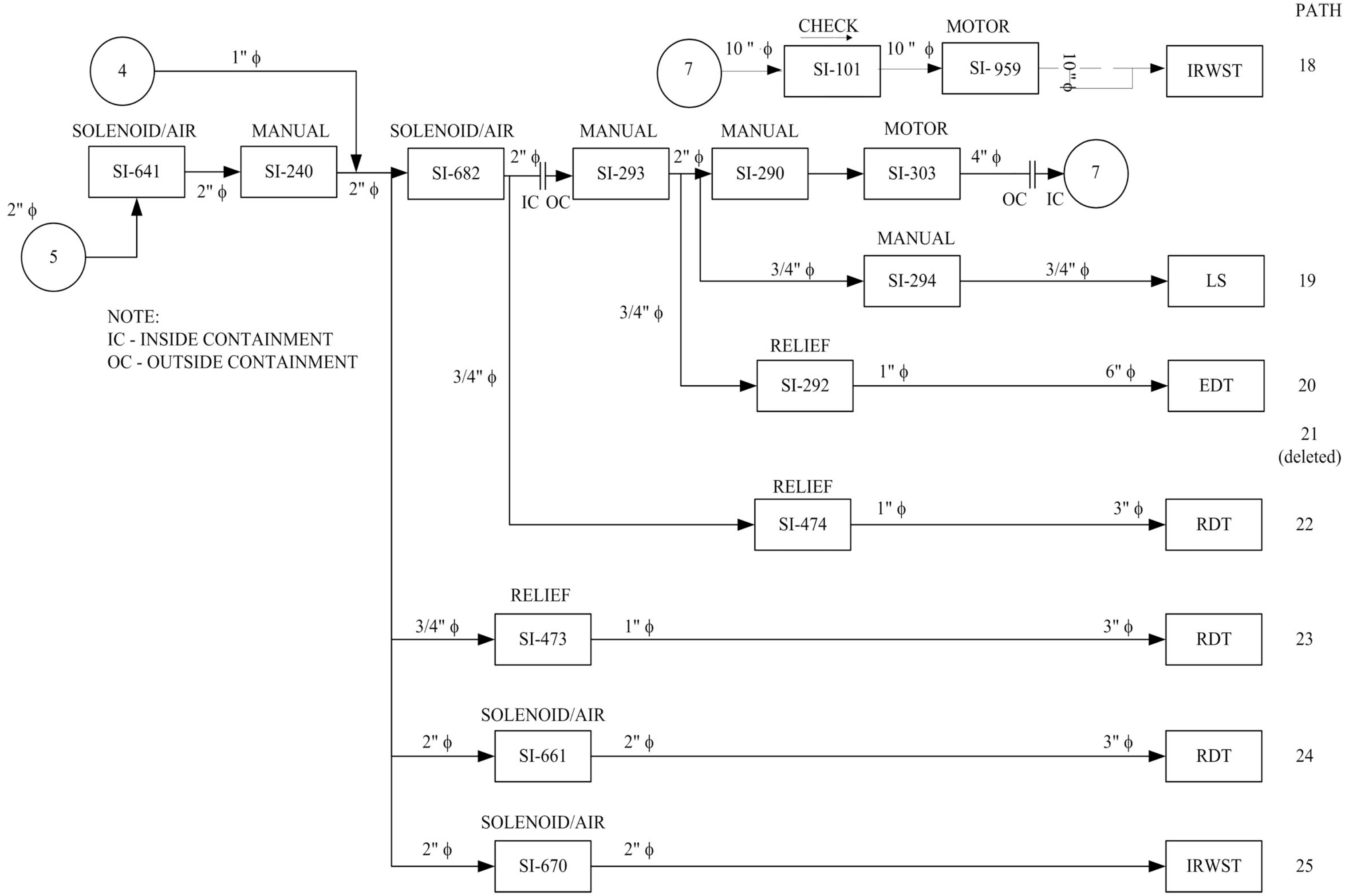

Figure 2.12-1 RCS Drain Paths through SCS Train 1 (5 of 5)

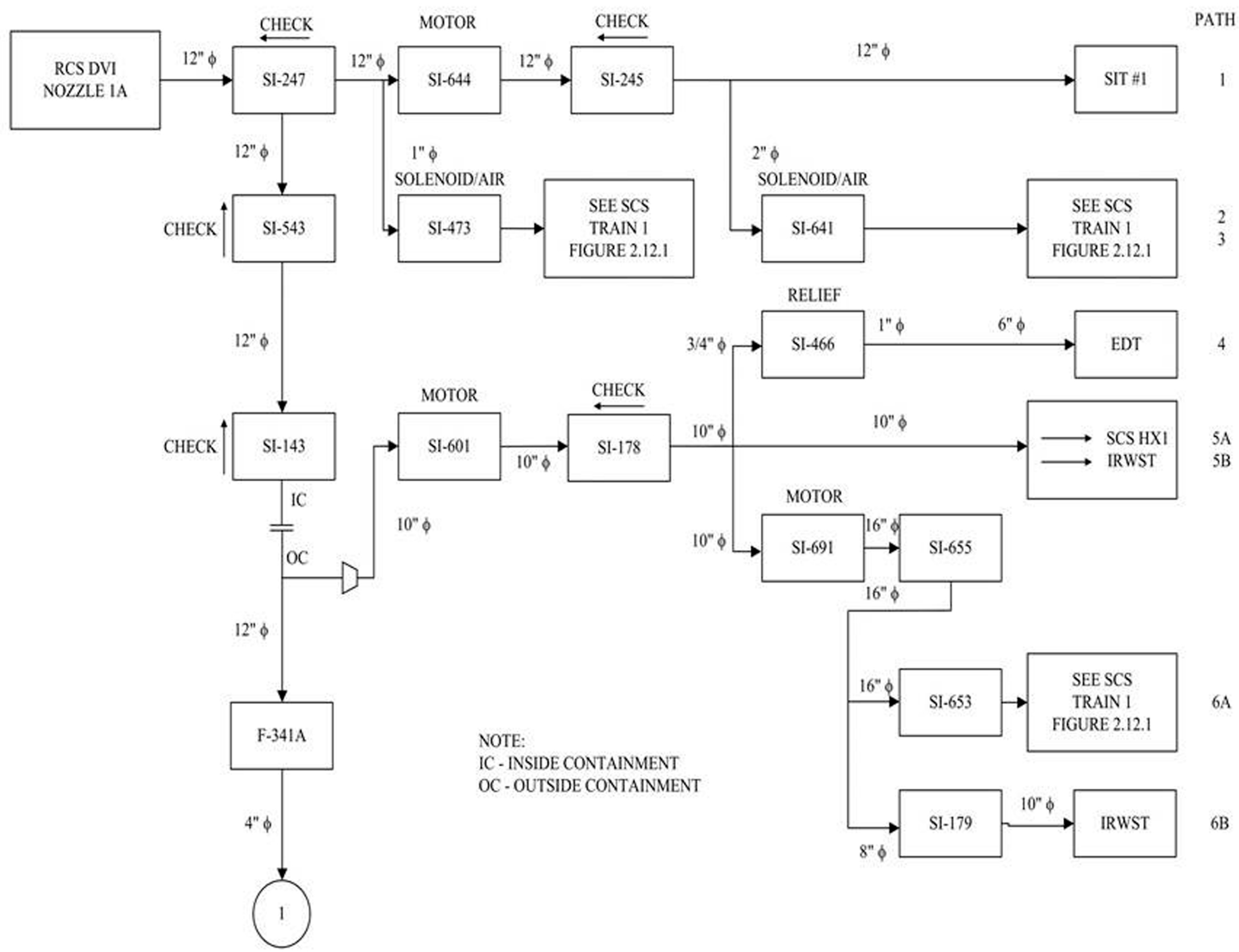

Figure 2.12-2 RCS Drain Paths through SIS/SCS DVI Nozzle 1A (1 of 2)

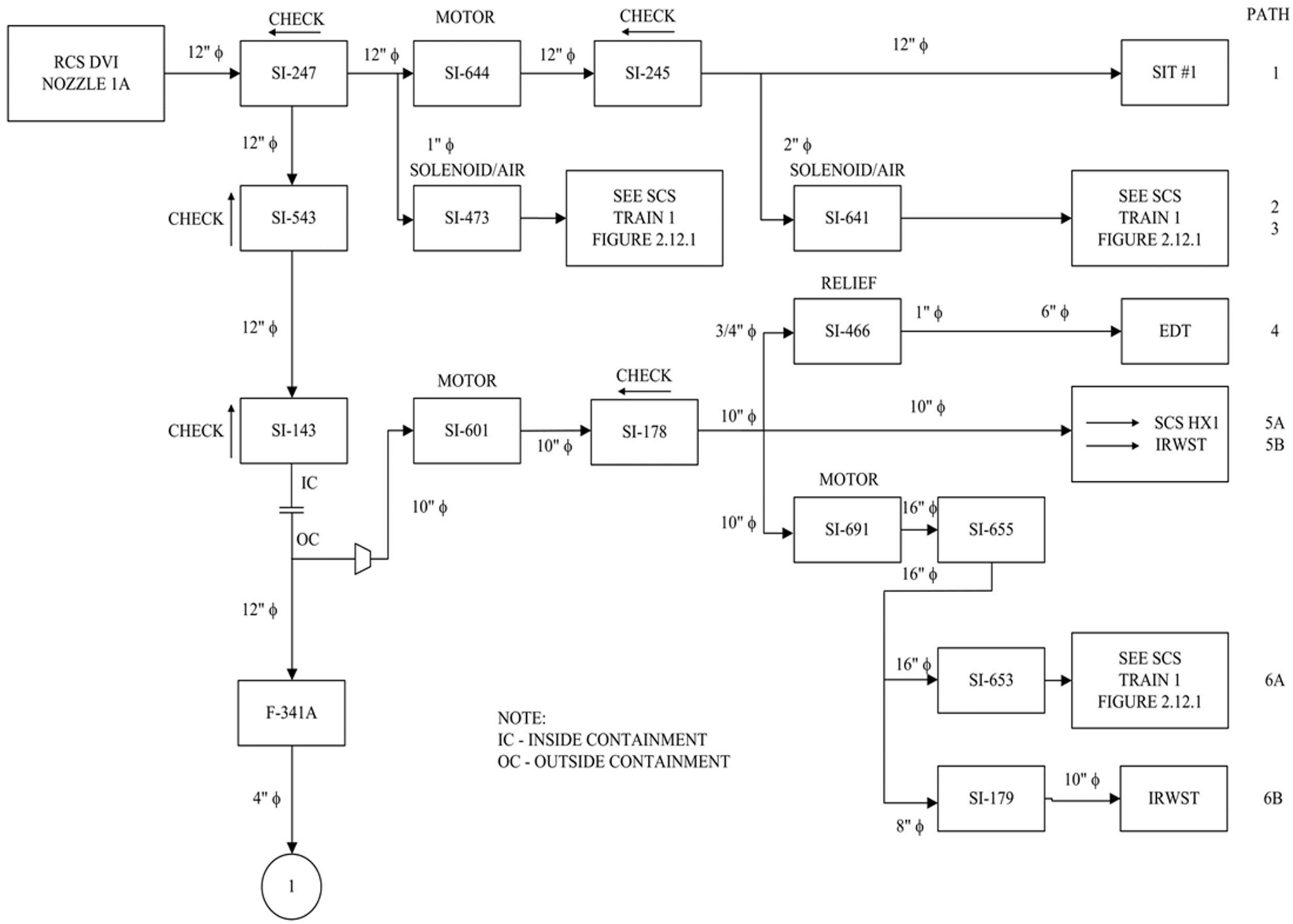

Figure 2.12-2 RCS Drain Paths through SIS/SCS DVI Nozzle 1A (page 2 of 2)

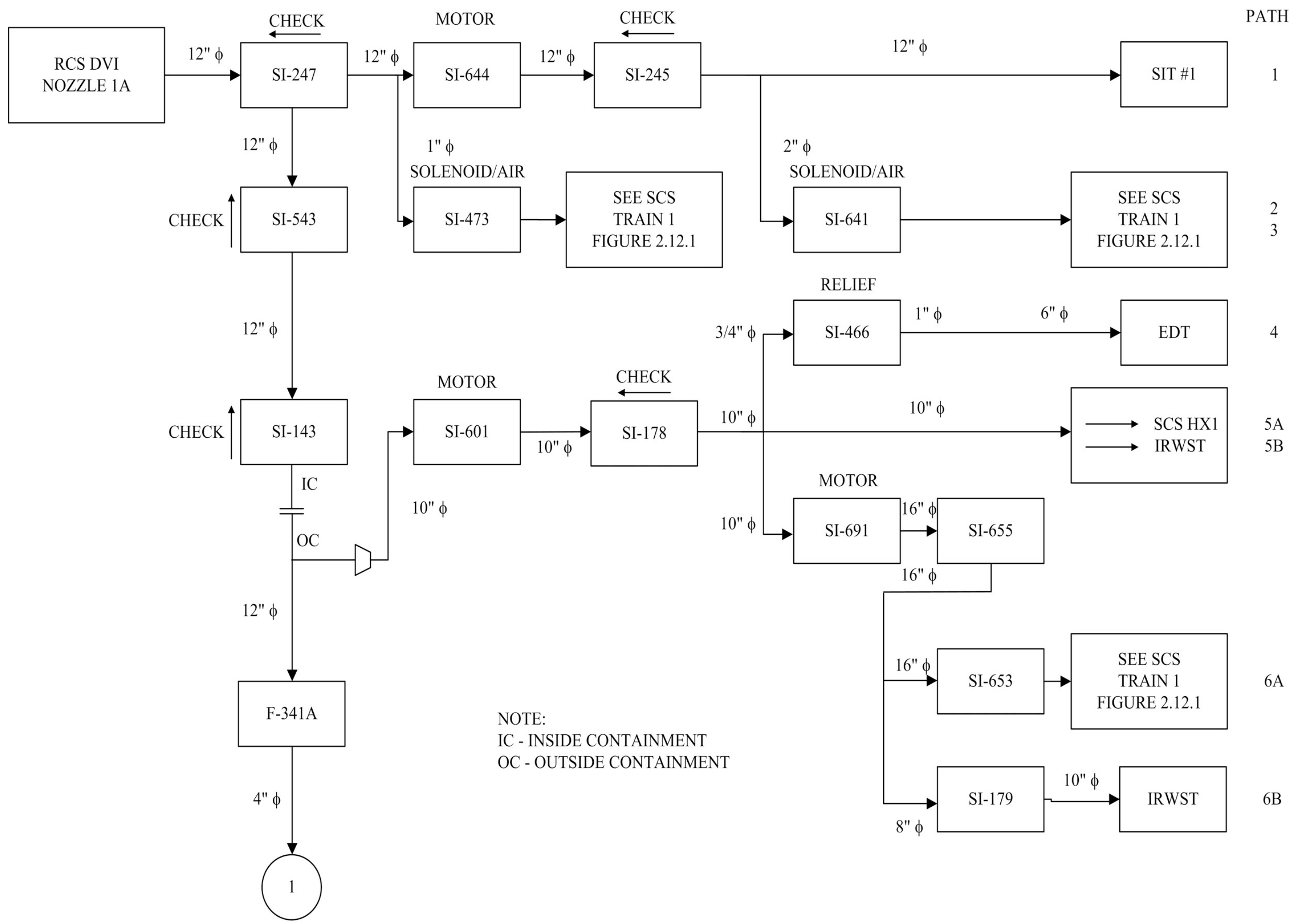

Figure 2.12-3 RCS Drain Paths through CVCS (1 of 4)

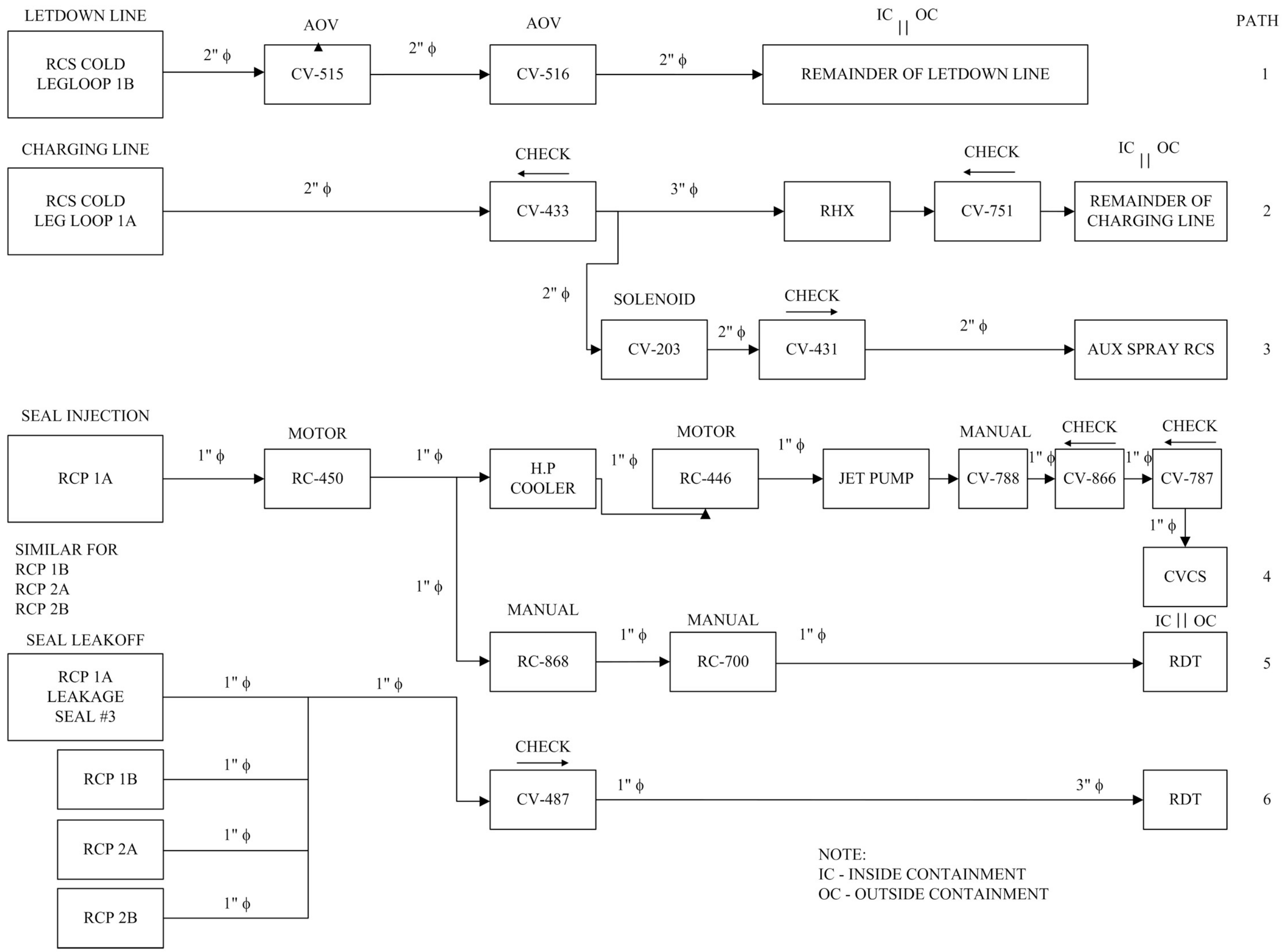

Figure 2.12-3 RCS Drain Paths through CVCS (1 of 4)

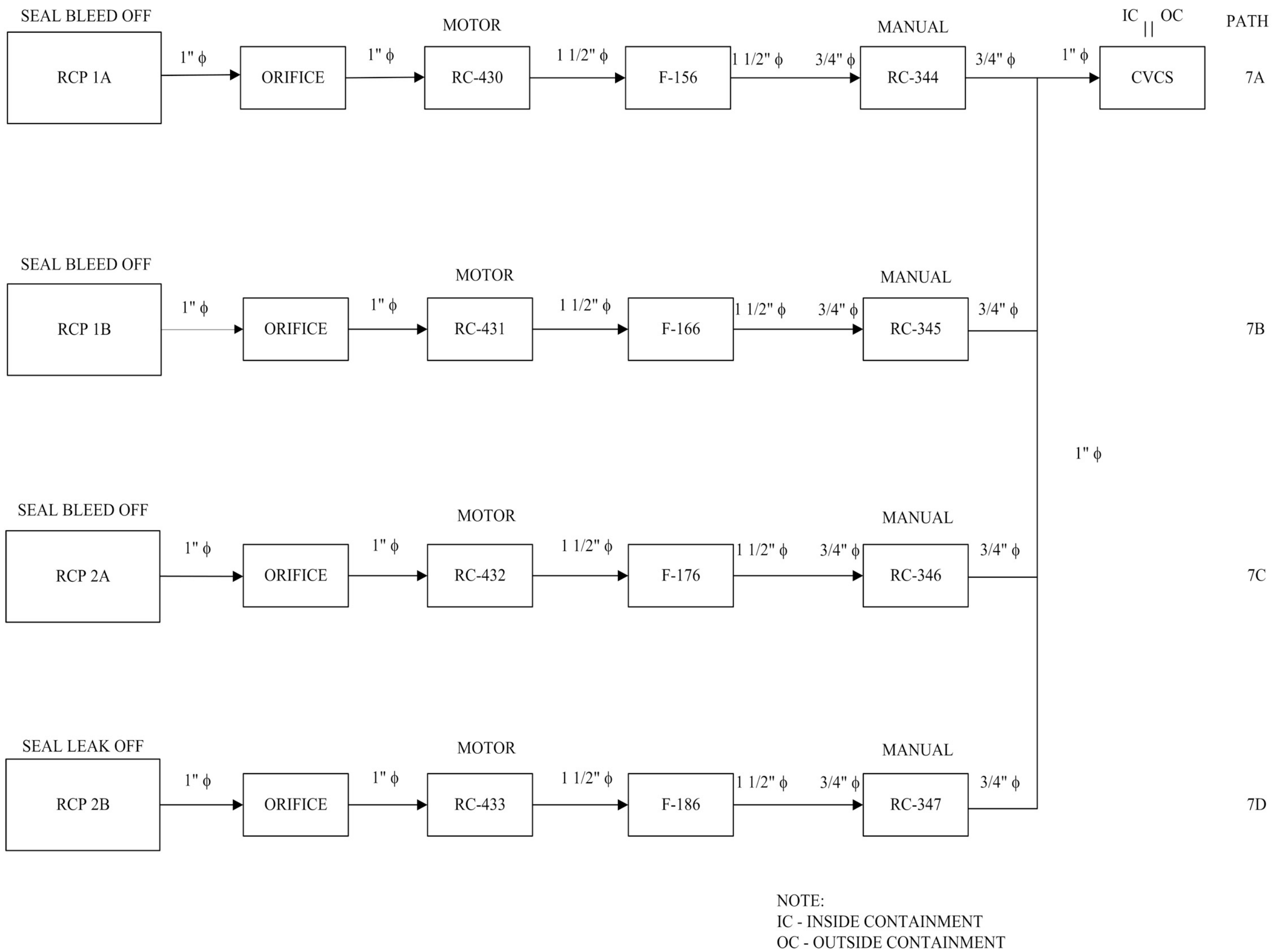

Figure 2.12-3 RCS Drain Paths through CVCS (page 2 of 4)

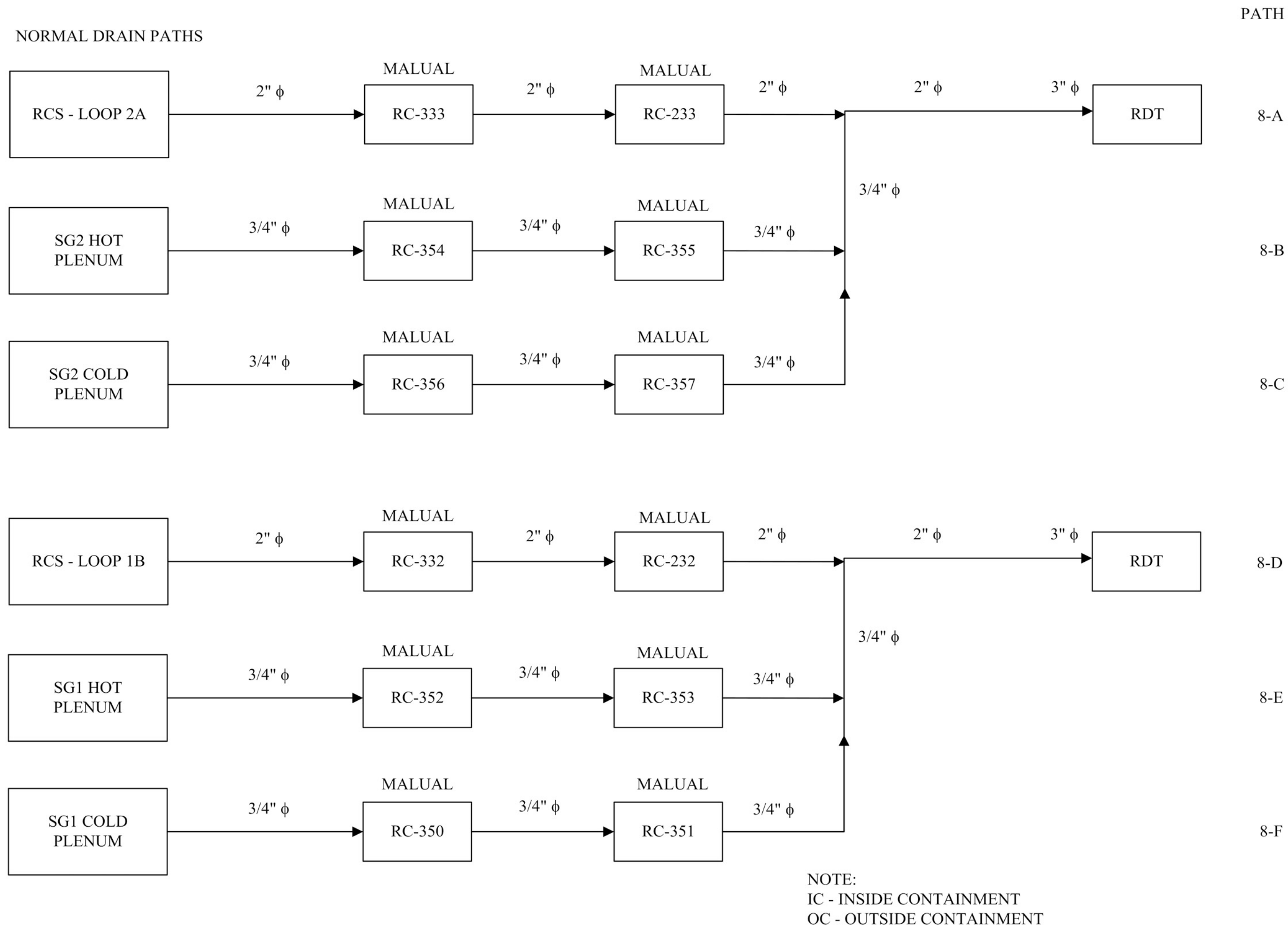

Figure 2.12-3 RCS Drain Paths through CVCS (page 3 of 4)

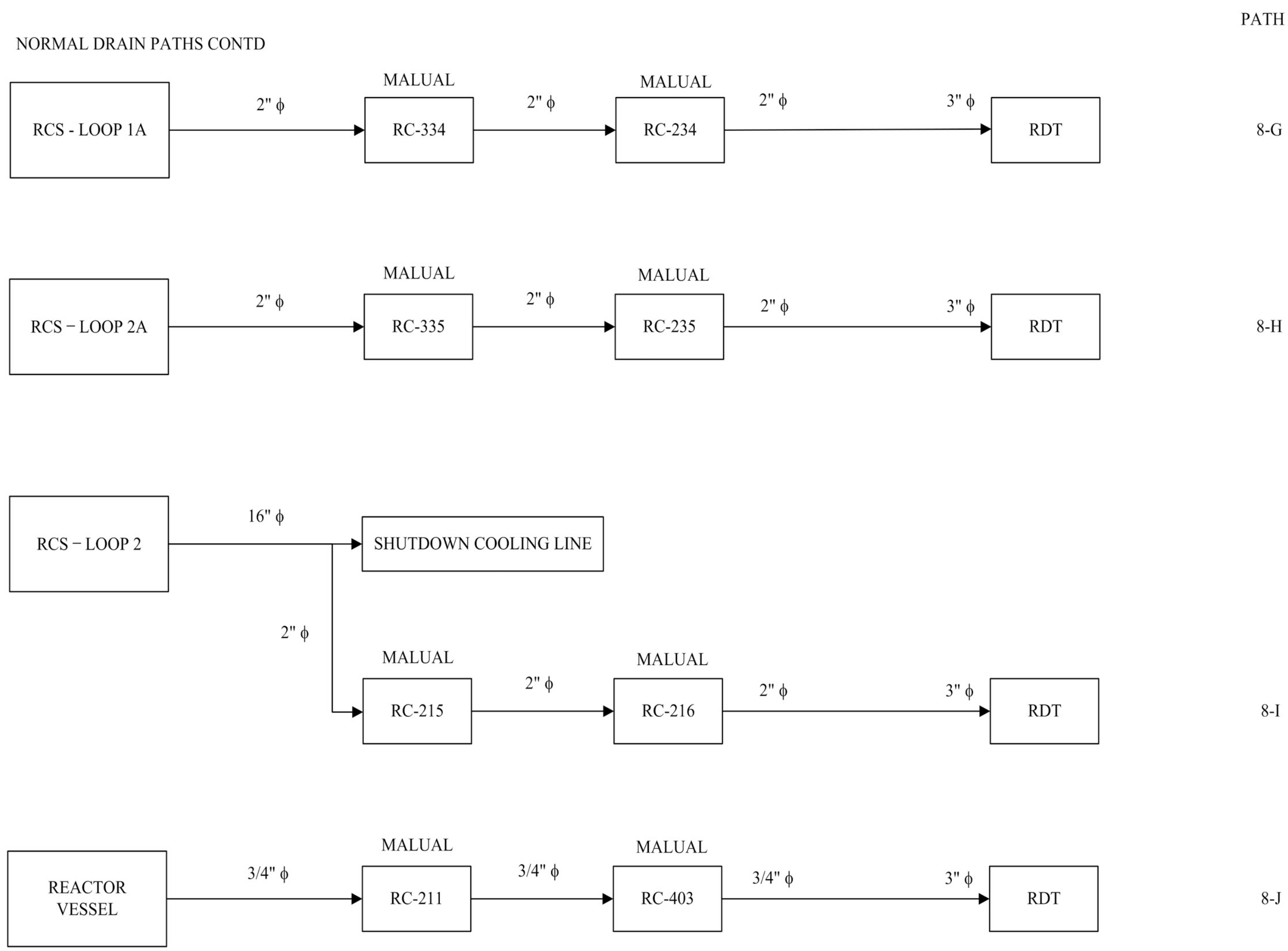

Figure 2.12-3 RCS Drain Paths through CVCS (page 4 of 4)

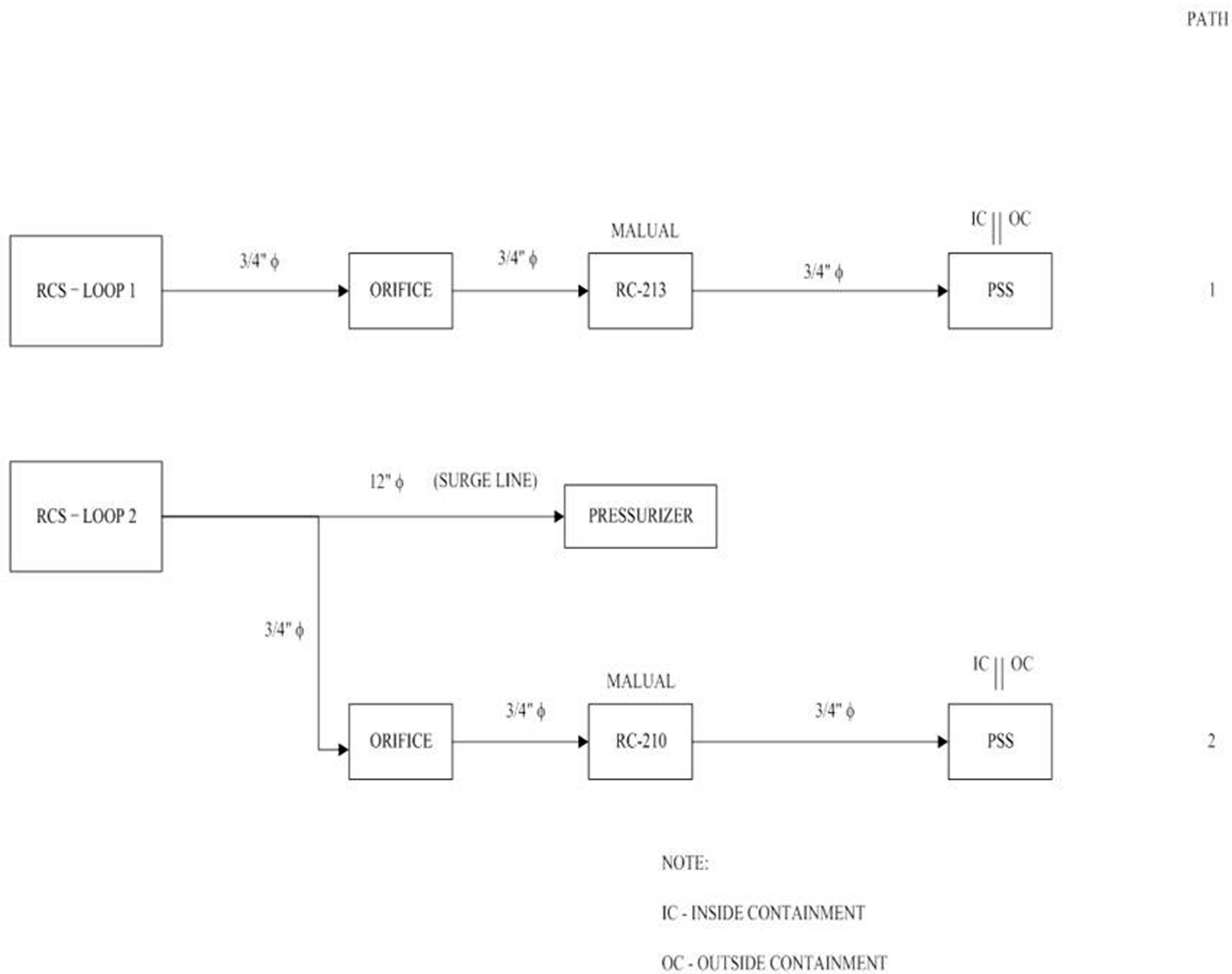

Figure 2.12-4 RCS Drain Paths through Primary SS

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 CCWS, service water system, condensers, and refueling pool seals. The issue addressed here is the potential for loss of DHR as a consequence of spills and internal flooding that may disable components of the SCS.

2.13.2 Acceptance Criteria

The flood protection design provides separation of redundant equipment to provide reasonable assurance that the availability and capability of DHR systems are not precluded due to flooding and spills.

In Generic Letter No. 88-17, "Loss of Decay Heat Removal," and SECY 94-176 "Issuance of Proposed Rulemaking Package on Shutdown and Low-Power Operation," NRC recommends that all PWR license holders implement certain expeditious actions before operating their plants in a reduced inventory condition, and implement, as soon as practical, program enhancements concerning operations during shutdown cooling, to reduce the potential for core damage and/or release of radiation during shutdown cooling.

In NUREG-1449, "Shutdown and Low-Power Operation at Commercial Nuclear Power Plants," NRC staff provides the scopes and methodologies for shutdown evaluation, with operating experience related to shutdown and low-power operation, probabilistic risk assessment of shutdown and low-power conditions, and utility programs for planning and conducting activities during periods in which the plant is shutdown.

2.13.3 Description

The APR1400 design emphasizes the elimination and minimization of potential flood sources within safety-related areas as a means of flood protection. It includes a number of features to minimize the risk associated with flooding and spills during shutdown (e.g., divisional and quadrant separation, floor drainage system and emergency flow paths separated by quadrant, the cooling water system located outside the auxiliary building).

2.13.3.1 APR1400 Design Features

2.13.3.1.1 Divisional and Quadrant Separation

Flood barriers are integrated into the design to provide further flood protection while minimizing the impact on maintenance accessibility. The primary means of flood control in the auxiliary building is provided by the structural wall that serves as a barrier between redundant divisions of safe shutdown systems and components. At the lowest elevation, this structural wall contains no doors or passages, and the limited penetrations through the wall are sealed. This design confines flood water to one division up to elevation of El. 78 ft. Thus, the other division is unaffected.

Half of each division is compartmentalized to separate redundant safe shutdown components to the extent practical, while maintaining accessibility requirements. The bottom of auxiliary building, which houses the front line safety systems, is compartmentalized into quadrants with two quadrants on either side of the divisional structural wall. Flood barriers (flood doors) provide separation between the divisions, while maintaining equipment removal capability. This design confines flood water to one quadrant up to El. 78 ft. Thus, the other adjacent quadrant is unaffected. The flood barrier drawing on divisional and quadrant separation is shown in Figure 2.13-1.

Flood doors are provided with open and close sensors and are alarmed in the control room. Flood doors withstand the static pressure from the maximum flood elevation as determined in the flood analysis.

2.13.3.1.2 Main Control Room

The APR1400 main control room (MCR) is protected from flooding in that no water lines are routed above or through the control room or computer room. Water lines routed to HVAC air handling units and similar components around the control room are contained in rooms with curbs, which prevents any potential water leakage from entering the control room or computer room.

2.13.3.1.3 Safety-Related Electrical Components

At higher elevations electrical equipment is elevated above the floor so that flooding events do not affect components. Additional barriers (e.g., curbs, ramps, sealed penetrations) are provided, or safety-related electrical components are elevated to mitigate the effects of postulated pipe rupture. Elevated equipment pads also prevent equipment from being inundated in the event of flooding.

2.13.3.1.4 Cooling Water System

Component cooling water, auxiliary feedwater, and essential chilled water are separated by division with no open cross connections, thus eliminating the possibility of a single pipe break from flooding one division and the other division being lost due to loss of pressure boundary integrity. Additionally, auxiliary feedwater pumps are located in separate compartments within the quadrants with each compartment protected by flood barriers.

2.13.3.1.5 Floor Drainage Systems

Flood protection is also integrated into the floor drainage systems. Floor drainage systems are separated by quadrant. The auxiliary building also has its own quadrantal separated floor drainage system, and there are no common drain lines between quadrants. Floors are gently sloped to allow good drainage to the quadrantal sumps. Floor drains are routed to the lowest elevation to prevent flooding of the upper elevations. The lower elevation in each quadrant has adequate volume to collect water from a break in any system without flooding the other quadrant. In addition, potential discharge of fixed fire-suppression systems and fire hoses is considered in sizing floor drains to preclude flooding of areas if fire protection systems are initiated.

2.13.3.1.6 Emergency Flow Paths

In the auxiliary building, normal floor drains do not have sufficient capacity to accommodate design basis flooding, and emergency drain paths are provided. The radioactive drain sump areas consist of four sections, one for each quadrant. Emergency drain paths from the upper to lower elevations within a quadrant are routed to the section of the radioactive drain sump area dedicated to that quadrant. These paths may be dropout panels, open gratings, or wall slots that are appropriately sized.

In the bottom of the auxiliary building, overflow provisions between the ESF area and non-ESF area (fire pump area or CVCS area) in each quadrant are provided to hold the maximum flood source within each quadrant.

2.13.3.1.7 Turbine Building

The turbine building includes engineered design features such that flooding equivalent to full flow discharge of the circulating water system can be accommodated without jeopardizing safe plant shutdown. These features include a watertight wall between the turbine and the auxiliary building from the turbine

building basement to a sufficient height above grade to allow outflow to the outside while preventing cross-building flooding.

A door leading from the turbine building to the auxiliary building is located above the maximum turbine building flood elevation.

Design basis flooding in the turbine building is accommodated by appropriately sized openings or other equivalent discharge paths.

2.13.3.1.8 Emergency Diesel Generator Building

The EDG building is divisional separated by walls such that a flood in one division cannot flood the other division.

2.13.3.1.9 Component Cooling Water Heat Exchanger Building and Essential Service Water Intake Structure

Flood protection is also incorporated into the CCW HX building and ESW intake structure. These structures are divisional and separated by walls such that a flood in one division cannot flood the other division.

2.13.3.2 Evaluations

The internal flood sources that were identified during power operation are also the sources within the auxiliary building during shutdown operations. The following internal flood sources are determined to have the potential for release within the auxiliary building:

- Component cooling water system (CCWS):

  The internal flood source is the estimated volume of water contained in one division of the CCWS. It is assumed as 757 m^3 (200,000 gal or 26,734 ft^3). After a postulated pipe rupture, this volume is released to a quadrant of the auxiliary building within 24 hours.

- Auxiliary feedwater system (AFWS):

  The internal flood source is based on the volume of water (minimum usable volume) contained in one AFW tank, 2,315 m^3 (611,627 gal or 81,763 ft^3). After a postulated pipe rupture, this volume is released to a quadrant of auxiliary building within 24 hours.

- Fire protection system (FPS):

  The internal flood source is assumed to be a potential source of water that could enter a quadrant of auxiliary building through the FPS piping from the two fire protection water supply tanks with a capacity of 3,293 m^3 (870,000 gal). After a postulated pipe rupture (through-wall crack for moderate energy line), it is estimated that the water of the two fire protection tanks drained to a quadrant with discharge flow rate of 1070 L/min (0.63 ft^3/sec or 283 gpm). Therefore, the total discharged volume is 1,541 m^3 (54,432 ft^3) for 24 hours.

- In-containment refueling water storage tank (IRWST):

  The internal flood source is based on the normal operating water volume contained in the IRWST with a capacity of 2,540 m^3 (671,114 gal or 89,175 ft^3). After a postulated pipe rupture, this volume is released to a quadrant of the auxiliary building within 24 hours.

- Chemical and volume control system (CVCS):

  The internal flood source is assumed to be a potential source of water that could enter a quadrant of the auxiliary building through CVCS piping from combined CVCS tanks (holdup tank : 318 m^3 [84,000 gal], boric acid storage tank : 899 m^3 [237,500 gal], and reactor makeup water tank : 1,420 m^3 [375,250 gal]. After a postulated pipe rupture, it is conservatively assumed that the volume of these three largest tanks is combined and drained to the CVCS area with a discharge flow rate 1,893 L/min (1.11 ft^3/sec or 500 gpm). Therefore, the total discharged volume is 2,637 m^3 (696,750 gal or 93,142 ft^3) for 24 hours.

  The net floodable volume of each quadrant is calculated by conservatively assuming that 75 percent of each quadrant gross volume is taken up by internal structures and equipment (e.g., heat exchanger, pump, tank, valve, etc.). The calculated volume of each quadrant is as follows:

  - Net floodable volume of the Quadrant A or B – 3205.3 m^3 (113,193 ft^3)

  - Net floodable volume of the Quadrant C or D – 3004.5 m^3 (106,104 ft^3)

Comparing the free volume of the each quadrant to each of the potential internal flood source volumes shows that the flood source volumes are less than the free volume of each quadrant. Therefore, flood water from either of the applicable flood sources is contained within the flood zone below El. 78 ft 0 in, and the equipment in the adjacent quadrants is unaffected by the flood due to the integrity of the division wall and flood door.

During shutdown, essential systems may be at higher risk for failure due to flooding and spills because of the varied and interrelated maintenance activities that may be in progress simultaneously. In other words, during shutdown conditions there is an increased probability that maintenance activities will affect a redundant train of DHR. In the APR1400, the SCS design provides two completely redundant divisions for DHR capability (i.e., they do not share components or equipment). The containment spray pumps are interchangeable with the SC pumps and used as an alternate DHR flow path. Even if a flooding event may be assumed simultaneously with a maintenance activity during shutdown operations, one train of DHR is operable. Therefore, it is considered that a flooding and spills in the auxiliary building during the shutdown modes of operation do not impair the DHR capability.

2.13.4 Resolution

The APR1400 flood protection design features are consistent with acceptance criteria outlined above in Subsection 2.13.2. These features resolve the issue of flooding and spills during shutdown operations in the APR1400 by providing separation of redundant equipment required for DHR. The separation provides the availability of DHR when a flood has occurred within the auxiliary building.

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Figure 2.13-1 Divisional and Quadrant Separation (Nuclear Island, El. 55´-0´´)

3.0 Applicability of the APR1400 Design Certification Chapter 15 Transient and Accident Analyses to the Shutdown Evaluation

The purpose of this section is to present the results of analyses that confirm that the transient and accident analyses in APR1400 DCD Tier 2, Chapter 15, conservatively bound the outcome of transients and accidents initiated during shutdown modes (Mode 2 subcritical and Modes 3 through 6) for the APR1400 design. The analyses presented in this section were performed at a design core power level of 3,983 MWt.

The analyses of events presented in Chapter 15 are based on the most adverse consequences. As a result, events postulated to occur in Mode 1 or Mode 2 critical are referenced more than shutdown modes, which are referenced only in cases that intrinsically involve shutdown modes (e.g., startup of an inactive reactor coolant pump [RCP]). The purpose of this section is to provide reasonable assurance that all operating modes are examined in the documentation of Chapter 15 events. The analysis models that were used are the same as those used in Chapter 15 and were approved by the NRC and the Korea Institute of Nuclear Safety.

Table 3.0-1, together with Figure 3.0-1 and Table 3.0-2, present the complete range of initial conditions used for the analyses of Chapter 15 events for all modes (Modes 1 through 6).

The structure of the remainder of Section 3 is parallel to the structure of Chapter 15. For example, Subsection 3.1.1 addresses the same group of initiating events that are addressed in Chapter 15, Subsection 15.1.1.

Table 3.0-1

Range of Initial Conditions

Parameter	Units	Range
Core power	% of 3983 MWt	0 - 102
Axial shape index	-	-0.3 < ASI < + 0.3 (1)
Reactor vessel inlet coolant flow rate	% of 1,689,429 L/min (% of 446,300 gpm)	95 - 116
Pressurizer water level	% distance (between upper tap and lower tap) above lower tap	21 - 60
Core inlet coolant temperature < 90 % Power 90 % - 100 % Power	 °C (°F) °C (°F)	 285.0 - 295.0 (545 - 563) 287.8 - 295.0 (550 – 563)
Pressurizer Pressure	kg/cm^2A (psia)	152.9 - 163.5 (2,175 – 2,325)
Steam generator water level Low level High level	 % Wide Range (2) % Narrow Range (3)	 40.7 95.0

Notes:

(1) ASI = (A - B) / C
where; A = Core power in lower half of core
B = Core power in upper half of core
C = Total core power
For power less than 20 %, -0.6 < ASI < +0.6 is used.

(2) Percent of distance between the wide range instrument taps.

(3) Percent of distance between the narrow range instrument taps.

Table 3.0-2

Reactor Vessel Inlet Coolant Flow Rate

Mode	System	Flow Rate (%)
3	RCP	33 to 116
4	RCP	33 to 116
	SCS	0.94 to 2.24
5 (loops filled)	RCP	33 to 116
	SCS	0.94 to 2.24
5 (loops not filled)	SCS	0.22 to 2.24
6	SCS	0.94 to 2.24

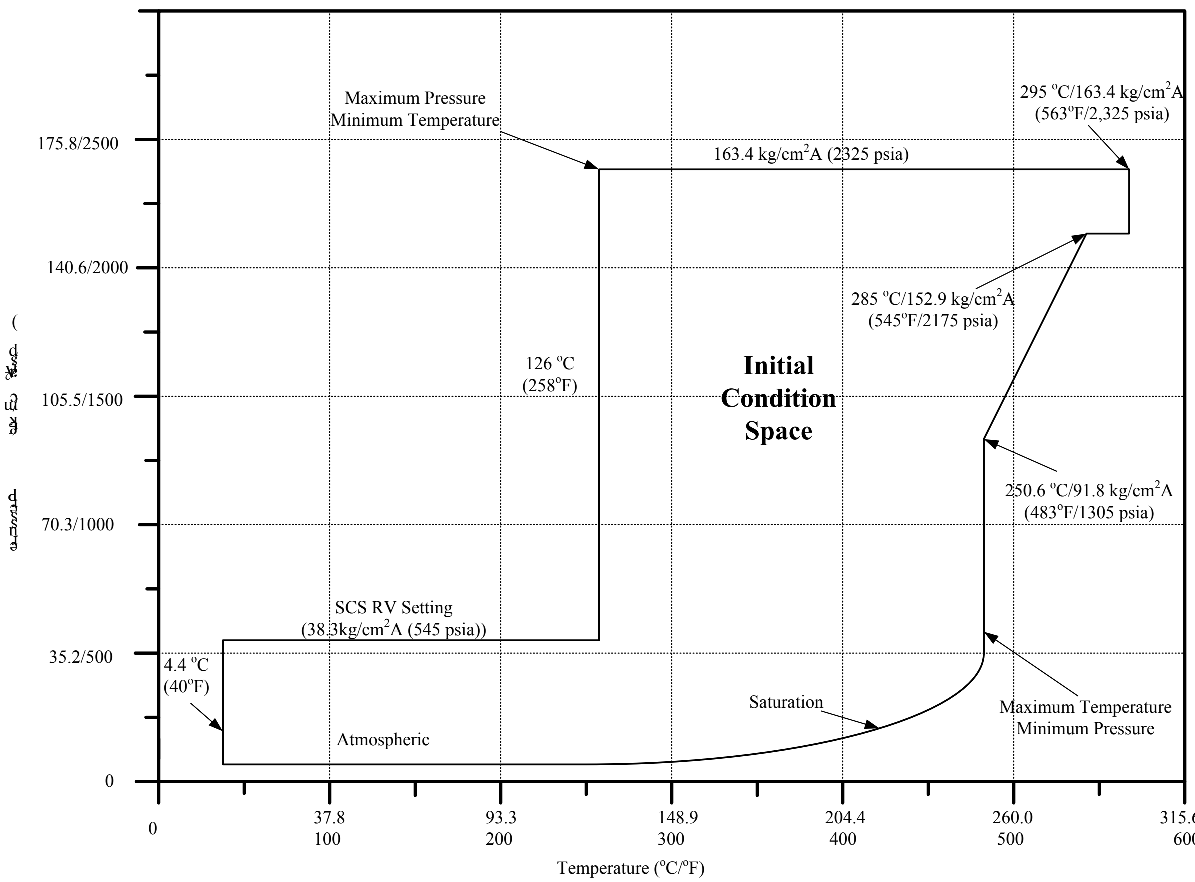

Figure 3.0-1 Initial Conditions for Pressure and Temperature in Mode 2 through 6

3.1 Increase in Heat Removal by the Secondary System

The purpose of this subsection is to present evaluations that confirm that all increases in heat removal by the secondary system events postulated to be initiated in a shutdown mode have acceptable consequences for the APR1400 design.

APR1400 DCD Tier 1, Chapter 15, Subsection 15.1, documents results that show that all increases in heat removal by the secondary system events have acceptable consequences if they are postulated to occur in Mode 1 or Mode 2 critical for the APR1400 design. This is demonstrated for steam system piping failures inside and outside containment (Chapter 15, Subsection 15.1.5) by analyses for both Mode 1 and Mode 2 critical. The choice of initial conditions for the other analyses of Chapter 15, Subsection 15.1, to minimize the transient departure from nucleate boiling ratio (DNBR) for any operating condition provides reasonable assurance that the results presented are bounded for all conditions of Mode 1 and Mode 2 critical.

3.1.1 Decrease in Feedwater Temperature

The consequences of a decrease in feedwater temperature event initiated in a shutdown mode are bounded by the consequence of increase in feedwater flow events presented in Subsection 3.1.2. The increase in feedwater flow events of Subsection 3.1.2 are initiated at the coldest possible feedwater temperature and the highest feedwater flow rate. This extreme combination of conditions results in the most severe reactor coolant system overcooling transient and consequently the greatest potential for a return to criticality with subsequent power generation and potential for DNB.

3.1.2 Increase in Feedwater Flow

An evaluation of the consequences of increase in feedwater flow events postulated to be initiated in a shutdown mode shows that the results are no more adverse than those of the full power event documented in Chapter 15, Subsection 15.1.2. For the most adverse combination of conditions, the minimum transient value of DNBR is predicted to be greater than that of Subsection 15.1.2, and fuel integrity is therefore not challenged. All increase in heat removal events analyzed in this section are characterized by decreasing steam generator (SG) pressures and RCS pressure, and there is therefore no approach to 110 percent of the design pressure of the SGs or RCS.

An increase in feedwater flow event is applicable only when the SGs are being used as a heat sink, which could occur in Mode 2 subcritical, Mode 3, or Mode 4 when the SC system (SCS) is not being used for decay heat removal (DHR). The turbine-generator would be offline. Steam would be exiting the SGs via the atmospheric dump valves or the steam bypass control valves. Normal feedwater flow would be entering the SGs from the startup feedwater system via the main feedwater piping. An increase in feedwater flow could occur via the main or auxiliary feedwater systems.

Administrative controls require main feedwater pumps to be unavailable for inadvertent delivery of feedwater to the SGs in shutdown modes. There are therefore two increase in feedwater flow event scenarios that can be postulated as potentially bounding events for shutdown modes:

a. Inadvertent auxiliary feedwater actuation signal (AFAS) actuation: The initiating event is assumed to be a spurious AFAS for one SG. Each SG is fed from a separate auxiliary feedwater (AFW) train, and each AFW train is actuated by a separate AFAS. The postulated initiating event results in starting both the turbine-driven and motor-driven AFW pumps to the affected SG. The cavitating venturi in the AFW supply pipeline limits the flow to less than 3,096.1 L/min (950 gpm). The minimum temperature from the AFW storage tanks is 4.4 °C (40 °F).

b. Maximum startup feedwater flow: The initiating event is assumed to be one control valve failing open, diverting all flow from the startup feedwater (SUFW) pump to one SG. One SUFW pump supplies both SGs, and there is a separate control valve for each SG. A bounding value of the feedwater flow is 15,141.2 L/min (4,000 gpm) (more than twice the specified 5 percent of main feedwater flow and greater than the expected runout flow). The water source is assumed to be the condensate storage tank, which is conservatively assumed to be at 0 °C (32 °F).

To cover these postulated scenarios with extreme bounding assumptions, a spectrum of cases were investigated using feedwater flows rates from less than 3,096.1 to 15,141.2 L/min (950 to 4,000 gpm), all at 0 °C (32 °F), and all initiated from the most adverse set of initial conditions.

The most adverse initial conditions for excess feedwater flow events initiated in a shutdown mode occur in Mode 3. The maximum core inlet temperature of 296.1 °C (565 °F) is conservatively assumed to maximize the positive moderator and Doppler reactivity addition caused by the RCS cooldown. (Parametric cases demonstrated that the most adverse results occurred with the maximum initial temperature.) Initiating at a lower core inlet temperature results in less positive moderator and Doppler reactivity coefficients and in a lower rate of RCS overcooling. Further, fewer than four RCPs can be operating in Mode 3 (as opposed to Mode 2 subcritical). This parameter has the most significant effect on the minimum DNBR. All cases were therefore initiated in Mode 3.

The maximum k_{eff} allowed by Technical Specifications in Mode 3 provides the greatest potential for reaching criticality and consequent power generation due to the cooldown event prior to encountering a reactor trip on low SG pressure or high SG level. The excess feedwater flow events were therefore initiated at k_{eff} = 0.99. It was assumed that one or more regulating CEAs was inserted and that all remaining regulating CEAs and shutdown CEAs were fully withdrawn.

To maximize the cooldown rate and the time required to reach the high SG water level trip setpoint, the events were assumed to be initiated at the minimum Mode 3 Technical Specification SG water level. For additional conservatism, it was assumed that the decay heat production was zero at the start of each event. Consequently, all supplied feedwater was excess and contributed to the RCS overcooling.

Decreasing core coolant flow reduces the minimum DNBR for a given level of power generation. However, the maximum rate of RCS overcooling - and the consequent potential for power generation - for these events occurs with four RCPs operating. The low reactor coolant flow trip prevents operation with the reactor trip breakers closed and fewer than one RCP operating in each loop for Modes 3 or 4. Further, with fewer than all RCPs in operation, a reactor trip is generated two decades lower in power due to the automatic removal of the core protection calculator (CPC) bypass than when all RCPs are in operation. In addition, any postulated occurrence that would result in tripping f operating RCPs after event initiation (e.g., loss of offsite power [LOOP]) would cause a reactor trip on low flow and terminate any approach to criticality earlier than that calculated without the loss of flow. It has been determined via parametric studies that the net effect of these factors is that the minimum transient DNBR occurs for postulated events for which all four RCPs are in operation.

These cases included the effect of the most reactive CEA stuck in the fully withdrawn position. No single failures were found that could adversely affect these events with respect to reducing the transient minimum DNBR. The calculation of the maximum post-trip reactivity included the effect of a failure of one EDG to start, run, or load (resulting in the unavailability of two safety injection pumps for conservatism).

The rate of cooldown for the excess feedwater flow events is less dependent on the SG temperature (and resultant pressure) than it is for steam line break events. Further, the maximum cooldown rate attainable with the spectrum of potential flows is such that all of the excess feedwater flow events were mitigated at low core powers by a high logarithmic power trip or CPC trip. Consequently, the most adverse results

were calculated to occur for the highest flow rate and the highest initial RCS temperature.

The investigation of this spectrum of cases with the assumed conservative initial conditions found that the minimum transient DNBR for increased feedwater flow events postulated to be initiated in a shutdown mode was greater than that for the events addressed in Chapter 15, Subsection 15.1.2. Further, the maximum reactivity following reactor trip was also more negative than that for the events addressed in Chapter 15, Subsection 15.1.2. There was, therefore, no post-trip power generation or approach to DNB.

Since the minimum transient DNBR for the events to which the events of Chapter 15, Subsection 15.1.2, are referenced was greater than 1.29 throughout the transient, the consequences of events in shutdown modes are no more adverse than those of the events presented in the APR1400 DCD.

3.1.3 Increased Main Steam Flow

As noted in Chapter 15, Subsection 15.1.3, 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 an SG relief or safety valve event. Further, there are no other differences between these events that affect the consequences. Therefore, the conclusions of Chapter 15, Subsection 3.1.4, also apply to an increased main steam flow event.

3.1.4 Inadvertent Opening of a Steam Generator Relief or Safety Valve

An evaluation of inadvertent opening of an SG relief or safety valve events postulated to be initiated in a shutdown mode shows that the results are less adverse than those of the events documented in Chapter 15, Subsection 15.1.4.

This evaluation was completed as integral to the study of steam system piping failures postulated to be initiated in a shutdown mode. The transient caused by an inadvertent opening of an SG relief or safety valve is identical to that caused by a steam line break with an area that is equal to that of a relief or safety valve. The study presented below covers a full spectrum of break sizes and initial conditions, including all or fewer than all RCPs running. As documented below, the minimum transient DNBR for a steam system piping failure in a shutdown mode is greater than that for the events of Chapter 15, Subsection 15.1.4. The minimum transient DNBR for inadvertent opening of a SG relief or safety valve events postulated to be initiated in a shutdown mode is, therefore, also greater than that for the events of Chapter 15, Subsection 15.1.4. Since the minimum transient DNBR for the events of Chapter 15, Subsection 15.1.4, was greater than 1.29 throughout the transient, the consequences of the events presented in Chapter 15 bound those of the events in shutdown modes.

3.1.5 Steam System Piping Failures Inside and Outside Containment

An evaluation of steam system piping failures inside and outside containment, postulated to be initiated in a shutdown mode, shows that the results are less adverse than those of the events documented in Chapter 15, Subsection 15.1.5.

The evaluation presented in this subsection focuses on fuel performance as measured by the DNBR for verification of fuel integrity. Steam system piping failures are characterized by decreasing reactor coolant system (RCS) and SG pressures. Therefore, the RCS pressure remains well below 110 percent of design pressure and the pressure-temperature (P-T) limits for brittle fracture, providing reasonable assurance that the integrity of the RCS is maintained and that the SG pressure remains below 110 percent of design pressure, providing reasonable assurance that the integrity of the secondary system is maintained. The 2-hour inhalation dose at the exclusion area boundary (EAB) is also examined to confirm that if such an event were postulated to be initiated in a shutdown mode, the value would meet the NUREG-0800 acceptance criteria referenced in NUMARC 91-06 (Reference 5).

For shutdown modes, the reactor core is subcritical with power being generated only by decay heat.

Substantial margins to DNB exist at the time of a postulated event initiation. Steam system piping failures would cause a decrease in the temperature of the reactor coolant and in the RCS and SG pressures. The decrease in the reactor coolant temperature would result in an increase in core reactivity due to the negative moderator temperature coefficient. If the cooldown were sufficiently large, a return to criticality followed by an increase in core power may occur and could create a potential for degrading the margins to DNB. The two possibilities to be considered for a return to criticality for these events are as follows:

a. Cooldown sufficient to reach criticality prior to reactor trip

b. Extended cooldown after reactor trip of sufficient magnitude to reach criticality if there is insufficient safety injection boration to prevent a return to criticality

The initial reactor core and SG temperatures have particularly significant effects for steam system piping failures. The total potential magnitude of the possible RCS cooldown rises with increasing initial SG temperature. In addition, the initial rate of cooldown would be greater if the initial SG temperature were higher. Further, both the potential rate and the total potential magnitude of the reactivity increase caused by cooldown rise with increasing reactor core temperature. Steam system piping failures would therefore be expected to have the most adverse consequences if they were assumed to be initiated at the highest possible initial temperatures, if event mitigation were not affected by the initial conditions.

Event mitigation could, however, be delayed by decreasing the rate and extent of cooldown. The evaluations presented here therefore have explored the entire range of conditions from the maximum possible initial temperatures to those for which the changes in temperature induced by the cooldown would be insufficient to result in a return to criticality. Further, a full range of break sizes have been explored, up to and including a double-ended guillotine break of a main steam line upstream of a main steam isolation valve (MSIV). Table 3.0-1 presents a complete list of the ranges of initial conditions that were considered.

As noted above, one of the possibilities to be considered for a return to criticality for these events is a cooldown sufficient to cause criticality prior to reactor trip, assuming the core is initially at the maximum allowed reactivity for a shutdown mode. Exploration of a spectrum of break sizes shows that for the more rapid transients, caused by larger breaks and higher initial temperatures, a reactor trip would be initiated by a low SG pressure signal prior to reaching criticality, thus precluding any fission power generation. For slower transients (smaller breaks and lower initial temperatures), the low SG trip pressure would not be reached before the core was calculated to reach criticality and some fission power was generated, which would result in the generation of a reactor trip signal at a low power level by one of two reactor trip functions: a CPC trip or a high logarithmic power level trip.

Reactor trip is generally due to a high logarithmic power level signal for smaller break sizes and a low SG pressure trip signal for larger break sizes. For events initiated under conditions that would result in a CPC trip (e.g., fewer than four RCPs operating), the trip signal would be generated when the reactor power was two orders of magnitude lower. The core powers reached for the shutdown mode case are, however, all substantially lower than the core power of the limiting Mode 1 case. Further, the minimum transient DNBR for the shutdown mode cases was greater than that for the events of Chapter 15, Subsection 15.1.5. There is therefore no approach to loss of fuel integrity for steam system piping failures postulated to be initiated in a shutdown mode. The minimum DNBR of the events documented in Chapter 15, Subsection 15.1.5, bounds those of events in shutdown modes.

The second potential possibility for a return to criticality for steam system piping failures postulated to be initiated in a shutdown mode is extended cooldown after reactor trip of sufficient magnitude to reach criticality if there is insufficient safety injection boration. Consideration of the maximum possible integral reactivity change due to cooldown shows that the Technical Specification shutdown margin (6.5%Δρ) is sufficient to prevent a return to criticality following reactor trip for postulated steam system piping failures for which the initial RCS average temperature is less than 260 °C (500 °F), even if no safety injection

boration occurred. A series of cases was run to show that for initial RCS average temperatures between 260 °C (500 °F) and 296.11 °C (565 °F) (the maximum temperature for shutdown modes) and with other initial conditions within the limits of Table 3.0-1, a safety injection actuation signal would be generated on low PZR pressure and sufficient boration would occur to prevent a return to criticality. These cases included the effect of the most reactive CEA stuck in the fully withdrawn position and a single failure. Single failure considered is either one emergency diesel generator to start, or run, or load (resulting in unavailability of two safety injection pumps) or one MSIV failure to close. The maximum post-trip reactivity for these cases was significantly more negative than the limiting case documented in Chapter 15, Subsection 15.1.5. No fission power was generated, and there was no approach to DNB.

The mass of steam released to the environment for steam system piping failures postulated to be initiated in a shutdown mode is bounded by that for the Mode 2 analysis. Further, no loss of fuel integrity is calculated to occur for the shutdown mode events. The 2-hour inhalation dose at the EAB is therefore bounded by the does presented in Chapter 15, Subsection 15.1.5.

3.2 Decrease in Heat Removal by Secondary System

The purpose of this subsection is to present evaluations that 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 APR1400 design.

Chapter 15, Subsection 15.2, provides the results that show that all decrease in heat removal by the secondary system events have acceptable consequences if they are postulated to occur in Mode 1 or in general in Mode 2 critical for the APR1400 design. However, if there was a question of whether an event has already been considered in Mode 2 critical, this mode was also included in the evaluation.

The focus of the evaluations presented in this subsection is on providing reasonable assurance that the peak primary and secondary pressures are less than 110 percent of their design pressures, that the P-T limits for brittle fracture of the reactor coolant system (RCS) are not violated, and that fuel integrity is maintained for the events that are considered. Fuel performance, as measured by the DNBR, is used for verification of fuel integrity.

3.2.1 Loss of External Load

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

3.2.2 Turbine Trip

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

3.2.3 Loss of Condenser Vacuum (LOCV)

A loss of condenser vacuum (LOCV) event postulated to occur in a shutdown mode would result in less severe consequences than the event documented in Chapter 15, Subsection 15.2.3, since only decay heat is being generated by the reactor core.

The LOCV event is applicable only when the condenser is being used as a heat sink, which could occur in Mode 2 subcritical, Mode 3, or Mode 4 when the SCS is not being used for DHR. The turbine-generator would be offline, and steam would enter the condenser via the turbine bypass valves. Loss of DHR in Modes 4 through 6, when the SCS is being used, is addressed in Subsection 2.4.

The dominant factor that determines the peak primary and secondary pressure following a LOCV event is the magnitude of the energy mismatch between the primary and secondary systems. There is much less mismatch for events postulated to occur in a shutdown mode than for the event of Chapter 15, Subsection

15.2.3, even when an instantaneous termination of steam flow and feedwater flow is assumed. The maximum peak primary and secondary pressure are therefore less for a LOCV postulated to occur during a shutdown mode than for this event at full power. Consequently, there is no approach to 110 percent of the design pressure of the SGs or RCS. The LOCV event would not challenge the P-T limits for brittle fracture of the RCS since the condenser is not online (shutdown cooling conditions have been established) when low-temperature overpressure protection (LTOP) is active.

Likewise, fuel integrity is maintained for LOCV events initiated in a shutdown mode due to the relatively low reactor core heat fluxes associated with decay heat. Decreased coolant flow is the only parameter that can significantly reduce the minimum DNBR since reactor power does not increase and RCS pressure increases for this event. A LOOP is the only basic assumption that can affect coolant flow. Because the plant is in a shutdown mode, there would be no turbine trip after the LOCV event, and consequently, there would be no perturbation to the electrical grid that could cause a LOOP. However, analyses have shown that even when the LOCV event is initiated with no RCPs operating at the highest decay heat flux value (just after reactor shutdown), the minimum value of DNBR for this event is greater than that of Chapter 15, Subsection 15.2.3. For colder conditions, the minimum DNBR is even higher. For example, when the LOCV event is initiated just prior to establishing shutdown cooling conditions in Mode 4 and natural circulation conditions exist (i.e., no RCPs are operating), the minimum value of DNBR is more than sufficiently high to preclude any fuel failure.

3.2.4 Main Steam Isolation Valve Closure

The comparison between main steam isolation valve (MSIV) closure and LOCV events in Chapter 15, Subsection 15.2.4, also applies to shutdown modes. The evaluation of the LOCV event presented in Subsection 3.2.3 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 Subsection 3.2.3.

3.2.5 Steam Pressure Regulator Failure

A steam pressure regulator failure event does not apply to the APR1400 design and is therefore not evaluated.

3.2.6 Loss of Offsite Power to the Station Auxiliaries

The results of the loss of offsite power (LOOP) to the station auxiliaries event are the same as those for the loss of reactor coolant flow event presented in Subsection 3.3.1.

3.2.7 Loss of Normal Feedwater Flow

A postulated loss of normal feedwater flow (startup feedwater flow) during a shutdown mode would be less adverse than a 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.

3.2.8 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. The heatup event that is considered in this section is based on the same arguments in Chapter 15, Subsection 15.2.8.1, The cooldown potential would be worse for a steam line break, which is described in Subsection 3.1.5. A heatup event is mitigated by the PZR safety valves or the shutdown cooling relief valves when RCS temperatures are above or below, respectively, the LTOP enable/disable temperatures,

main steam safety valves, and auxiliary feedwater (AFW) system.

A feedwater system pipe break postulated to occur in a shutdown mode would result in less severe consequences than the event documented in Chapter 15, Subsection 15.2.8, 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 AFW system is capable of removing decay heat event with only one AFW pump in operation. The dominant factor that determines peak primary and secondary pressures is the magnitude of the energy mismatch between the primary and secondary systems. There is much less mismatch for events postulated to occur in shutdown modes than for the event in Chapter 15, Subsection 15.2.8. There is therefore no approach to 110 percent of SG design pressure because the maximum pressure is less than that for the full power event. For the same reason, there is no approach to the criterion of 110 percent 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 provides reasonable assurance that the P-T limits for brittle fracture of the RCS are not violated.

3.3 Decrease in Reactor Coolant System Flow Rate

The purpose of this subsection is to present evaluations that confirm that all decreases in reactor coolant flow rate events postulated to be initiated in a shutdown mode have acceptable consequences for the APR1400 design.

Chapter 15, Subsection 15.3, documents results that show that all decreases in reactor coolant flow rate events have acceptable consequences if they are postulated to occur in Mode 1 or Mode 2 critical. This is demonstrated for total loss of reactor coolant flow (Chapter 15, Subsection 15.3.1) by analyses. The choice of initial conditions for the other analyses of Chapter 15, Subsection 15.3, to minimize the transient DNBR for any operating condition provides reasonable assurance that the results that are presented are bound those for all of Mode 1 and Mode 2 critical.

A decrease in the reactor coolant flow rate in Modes 4 through 6 when the SCS is being used for DHR is addressed in Subsection 2.4 as integral to the evaluation of loss of DHR capability. Therefore, this section addresses events postulated to be initiated in Mode 2 subcritical, Mode 3, or Modes 4 and 5 when the SCS is not being used.

3.3.1 Loss of Forced Reactor Coolant Flow

An 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 Chapter 15, Subsection 15.3,1 full-power event.

A LOOP 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 that is much less than at full power. Therefore, a 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 the minimum DNBR 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, Subsection 15.3.1, 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 SG pressures that are lower than 110 percent 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 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 suction line relief valves provides reasonable assurance that the P-T limits for brittle fracture are not violated.

3.3.2 Flow Controller Malfunctions

A flow controller malfunction event is categorized as a boiling water reactor (BWR) event and is therefore not evaluated.

3.3.3 Reactor Coolant Pump Rotor Seizure

The major parameter of concern for the single RCP rotor seizure with a LOOP event documented in Chapter 15, Subsection 15.3.3, is the minimum hot channel DNBR. This is minimized by higher power conditions. Lower power modes would therefore not produce more adverse conditions.

The second parameter of concern is the peak RCS pressure that is attained. The full-power single RCP pump rotor seizure with a LOOP event produces RCS pressures that are less than 110 percent 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 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 provides reasonable assurance that the P-T limits for brittle fracture of the RCS would not be violated.

In addition, at these low-power levels, a concurrent turbine trip that results in a LOOP is not an issue since the turbine is not in operation at less than 5 percent power.

3.3.4 Reactor Coolant Pump Shaft Break

Since a postulated RCP shaft break transient results in a less rapid flow coastdown than a rotor seizure event, the results of the event are bounded by the results of the evaluation in Subsection 3.3.3.

3.4 Reactivity and Power Distribution Anomalies

The purpose of this subsection is to present evaluations that confirm that all reactivity and power distribution anomaly events postulated to be initiated in a shutdown mode have acceptable consequences for the APR1400 design.

3.4.1 Uncontrolled Control Element Assembly Withdrawal from a Subcritical or Low Power Condition

An evaluation of the consequences of an uncontrolled CEA withdrawal event postulated to be initiated from a shutdown mode shows that the results are less adverse than those of the event documented in Chapter 15, Subsection 15.4.1.

The event documented in Chapter 15 is initiated from low power, critical conditions. The Technical Specifications include requirements that provide reasonable assurance that the most adverse subcritical CEA withdrawal event scenario is the inadvertent withdrawal of a regulating CEA bank. For the evaluation presented here, the events that postulate initially subcritical conditions and an uncontrolled withdrawal of a regulating CEA bank were analyzed to demonstrate that the specified acceptable fuel design limit

(SAFDL) on DNBR is not violated and that no fuel melting is predicted to occur.

The essential reason that withdrawal of a regulating CEA bank from subcritical is less adverse than the same event at low power, core critical conditions is that one of two reactor trip functions terminates the subcritical event at low power levels (a CPC trip or a high logarithmic power level trip). The setpoints for these trip functions are such that the maximum power levels reached for this event are much lower than those for the low power event.

The CPCs may be bypassed at power levels below 0.0001 percent. This bypass is automatically removed upon exceeding this power level. Although a reactor trip does not occur when the CPCs are bypassed, the CPCs continue to perform their calculations and can be in a tripped condition. The CPCs would be in a tripped condition if fewer than all four RCPs are running or if the calculated radial peaking is out of range due to the presence of an undesired CEA bank in the core. As the core fission power increases during the CEA withdrawal event, the 0.0001 percent power bypass is removed, and if the CPCs are in a tripped condition, a reactor trip is generated.

The high logarithmic power level trip is required to be operable in all shutdown modes when the reactor trip breakers are closed and the CEA drive mechanism is capable of CEA withdrawal. Although the high logarithmic power level trip can be bypassed during Modes 1 and 2 critical, it cannot be bypassed during shutdown conditions.

For CEA withdrawal events that do not generate a CPC trip upon removal of the 0.0001 percent power bypass, the high logarithmic power level trip produces a reactor trip for which the setpoint is 0.05 percent power. (The corresponding setpoint used in the analyses is 0.2 percent.) If, however, the transient is the withdrawal of one of the early regulating banks, or if fewer than all RCPs are in operation, a reactor trip would be generated upon the removal of the CPC bypass at 0.0001 percent power. (The corresponding setpoint used in the analyses is 0.04 percent.)

CEA withdrawal events initiated from subcritical conditions were analyzed with assumptions that bound all shutdown conditions and determine the consequences when a reactor trip is due to each of the above functions. The cases that were analyzed are as follows:

(1) Regulating bank withdrawal, worst combination of burn-up dependent parameters, four pumps running, reactor trip on high logarithmic power level

(2) Regulating bank withdrawal, worst combination of burn-up dependent parameters, fewer than four pumps running, reactor trip upon removal of the CPC, 0.0001 percent power bypass due to CPC pump counting

The case with full forced reactor coolant flow experienced a peak heat flux of approximately 0.55 percent of full-power heat flux. This is well below the value of 21.0 percent calculated for the event documented in Chapter 15, Subsection 15.4.1. The DNBR for this case remained well above the DNBR SAFDL. Due to the earlier reactor trip, the case with fewer than all four RCPs running experienced a peak heat flux increase to less than 0.03 percent of full-power heat flux. This low value of heat flux would not cause DNB.

The deposited energy calculation assumed that all of the energy produced in the short power excursion is deposited in the fuel with no heat transfer from the fuel. The evaluation shows that the hot spot in the core does not experience melt for any withdrawal of a regulating CEA bank from subcritical.

In summary, the APR1400 DCD Technical Specifications provide reasonable assurance that the reactor does not become critical due to the inadvertent withdrawal of a shutdown CEA bank. The CPC and high logarithmic power level trip functions provide reasonable assurance that the consequences of an inadvertent withdrawal of a regulating CEA bank from subcritical conditions are substantially less adverse

than those documented in Chapter 15, Subsection 15.4.1. This event does not result in the violation of the DNBR SAFDL or in fuel melt.

3.4.2 Uncontrolled Control Element Assembly Withdrawal at Power

An uncontrolled CEA withdrawal at power event is not an issue for shutdown modes since the intent is to examine only high-power operations.

3.4.3 Control Element Assembly Misoperation

The types of anticipated operational occurrences that include one or more CEAs moving or displaced from normal or allowed control bank positions are as follows:

a. Dropped CEA or CEA subgroup

b. Statically misaligned CEA

c. Single CEA withdrawal

A postulated single CEA drop at power is analyzed for approach to the DNBR limit in Chapter 15, Subsection 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.

In the case of statically misaligned rod cluster control assemblies, the most limiting case and analysis are for Mode 1, which also bounds Mode 2 operation since statically misaligned rod cluster control assemblies have no effect on criticality and are not a concern below Mode 2.

The most limiting case for the withdrawal of a single CEA withdrawal is an occurrence while in Mode 1. In the subcritical modes, the shutdown margin specified in LCO 3.1.1 of the APR1400 Technical Specifications is determined to cover fully withdrawal of the most reactive CEAs from the core. Therefore, a single CEA withdrawal postulated to initiated from the subcritical modes does not insert enough reactivity to get criticality.

3.4.4 Startup of an Inactive Reactor Coolant Pump

Chapter 15, Subsection 15.4.4, presents an analysis of startup of an inactive RCP events that show that events postulated to be initiated in Modes 3 through 6 have acceptable consequences. Operation with fewer than 4 RCPs is not permitted in Modes 1 or 2.

3.4.5 Flow Controller Malfunction Causing an Increase in BWR Core Flow Rate

The event in which the flow controller malfunction causes an increase in BWR reactor core flow rate is categorized as a BWR event and is therefore not evaluated.

3.4.6 Inadvertent Decrease in Boron Concentration in the Reactor Coolant System

Analyses of inadvertent deboration events in shutdown modes are presented in Chapter 15. An additional evaluation that considers rapid deboration is presented in Subsection 2.6.

3.4.7 Inadvertent Loading and Operation of a Fuel Assembly in an Improper Position

The event in which there is inadvertent loading and operation of a fuel assembly in an improper position is evaluated in Chapter 15, Subsection 15.4.7, and is not mode dependent.

3.4.8 Spectrum of CEA Ejection Accidents

The CEA ejection analysis documented in Chapter 15, Subsection 15.4.8, is for an ejection from full-power initial conditions. This case was found to be more limiting than an ejection from Mode 2 critical initial conditions. The evaluation that is documented in Chapter 15, Subsection 15.4.8, credits the allowable CEA insertion limits given in Technical Specifications 3.1.6 and 3.1.7, which apply to Mode 1 and Mode 2 critical.

Technical Specifications 3.1.6 and 3.1.7 do not apply directly to Mode 2 subcritical or to Modes 3 through 5. However, the requirements of Technical Specification 3.1.1 and 3.1.2, which do apply in these shutdown modes and which use the criteria of Technical Specifications 3.1.6 and 3.1.7, provide reasonable assurance that the full-power CEA ejection case remains limiting.

For Mode 2 subcritical and for Modes 3 through 5, Technical Specification 3.1.1.b.1 requires that the calculated critical position be within the limits of Technical Specifications 3.1.6 and 3.1.7 when the reactor trip breakers are closed. This requirement provides reasonable assurance that a CEA ejection postulated to be initiated at these conditions would result in less net positive reactivity insertion than for a case initiated from a critical position. The case described in Chapter 15, Subsection 15.4.8, therefore remains limiting for conditions under which the reactor trip breakers are closed.

When the reactor trip breakers are open in Modes 3 through 5, Technical Specifications 3.1.1.b.2 requires that the value of k_{eff} must remain less than 0.99 when the highest worth CEA is excluded from the calculation. This requirement provides reasonable assurance that the reactor would not reach criticality for a CEA ejection event postulated to be initiated under these conditions.

The full-power event of Chapter 15, Subsection 15.4.8, is therefore the limiting CEA ejection event.

3.5 Increase in RCS Inventory

The purpose of this subsection is to present evaluations that demonstrate that all increases in RCS inventory events postulated to be initiated in a shutdown mode have acceptable consequences for the APR1400 design.

Chapter 15, Subsection 15.5, contains the results that show that all increases in RCS inventory events have acceptable consequences if they are postulated to occur in Mode 1 or in general in Mode 2 critical for the APR1400 design. If there is a question of whether an event has already been considered in Mode 2 critical, then, to provide reasonable assurance of completeness, this mode is also included in the evaluation.

The focus of the evaluations presented in this subsection is on providing reasonable assurance that the peak primary pressure is less than 110 percent of design pressure and that the P-T limits for brittle fracture of the RCS are not violated. Fuel performance, as measured by the DNBR, is used for verification of fuel integrity. Peak secondary pressure is also evaluated as necessary to provide reasonable assurance that it remains less than 110 percent of its design pressure.

3.5.1 Inadvertent Operation of the Emergency Core Cooling System

The evaluation presented in Chapter 15, Subsection 15.5.1, establishes that a postulated inadvertent operation of the emergency core cooling system (ECCS) would have acceptable consequences for any mode for the APR1400 design.

3.5.2 Chemical and Volume Control System 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 percent of design pressure. Further, the P-T 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 (RV) head is off.

In Modes 2 subcritical through Mode 4 before shutdown cooling and LTOP are placed in service, the peak RCS pressure is limited by the POSRV. Since only decay heat exists under these conditions, the peak pressures during the CVCS malfunction event are substantially less severe than the Mode1 case.

In Modes 4, 5, and 6 when LTOP is active or in Mode 6 with the RV head is off and LTOP is not active, the CVCS malfunction event is less limiting than the inadvertent SIS actuation described in Subsection 3.5.1 due to the much lower flow from one charging pump versus the four SI pumps.

3.6 Decrease in Reactor Coolant System Inventory

The purpose of this subsection is to present evaluations that demonstrate that all decreases in RCS inventory events postulated to be initiated in a shutdown mode have acceptable consequences for the APR1400 design.

Chapter 15, Subsection 15.6, provides the results that show that all decreases 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 APR1400 design. If there is a question of whether an event has already been considered in Mode 2 critical, then to provide reasonable assurance of completeness, this mode is also included in the evaluation.

Fuel performance, as measured by the DNBR, is used for verification of fuel integrity.

3.6.1 Inadvertent Opening of a Pressurizer Safety/Relief Valve

The inadvertent opening of a PZR safety/relief valve is identified as an inadvertent opening of the pilot operated safety relief valve (POSRV) for the APR1400 design. The evaluation of an inadvertent opening of a POSRV is described in Chapter 15, Subsection 15.6.5, which presents a small break loss-of-coolant accident (LOCA).

3.6.2 Double-Ended Break of a Letdown Line Outside Containment

The consequences of a letdown line break outside containment initiated from Modes 2, 3, 4, or 5 are less severe than the consequences of such events initiated from Mode 1. Chapter 15, Subsection 15.6.2, gives the evaluation for the break initiated from Mode 1. The three principal considerations in comparing the radiological consequences are as follows:

a. Time of an alarm to alert the operator

b. Total fluid release prior to operator action that isolates the break

c. Fraction of leak flow that flashes, releasing radioactivity to the atmosphere.

The three considerations are described below.

APR1400 DCD, Table 15.6.2-1, lists alarms that are actuated following a letdown line break during Mode 1. The same alarms are available during Modes 2, 3, and 4 so the operator receives an equivalent indication of the break initiation. Operator action is assumed after 30 minutes of the event initiation, as with the Mode 1 evaluation.

Parametric studies for the Mode 1 evaluation determined the worst set of initial conditions. They are maximum power, maximum temperature, maximum pressure, and high PZR level. The Mode 1 initial conditions bound the conditions that exist in Modes 2, 3, 4, and 5.

The early alarm and the bounding initial conditions satisfy the first two principal considerations that determine the total mass release until break isolation. The third principal consideration is the flashing fraction, which involves the cooling of the leak flow as it passes through the regenerative and letdown HXs on its way out of the break.

APR1400 Subsection 9.3.4.2.3 describes the operation of the chemical and volume control system (CVCS), including the regenerative and letdown HXs. The normal charging flow, assumed in the Mode 1 analysis, cools the letdown flow exiting the regenerative heat exchanger (HX) to less than 232.2 °C (450 °F). Additional cooldown of the letdown flow through the letdown HX results in further reduction of the letdown flow temperature to less than 48.9 °C (120 °F). Continuous charging flow is also normal in the shutdown modes. In conclusion, the total break flow, pressure, and temperature for a break initiated from Modes 2, 3, 4, or 5 yield no more flashed coolant than for the break initiated from Mode 1. Therefore, the radiological consequences are less severe for a letdown line break initiated from Modes 2, 3, 4 or 5 than the consequences reported in Chapter 15, Subsection 15.6.2.

3.6.3 Steam Generator Tube Rupture

Evaluation of the consequences of an SG tube rupture (SGTR) accident postulated to be initiated in a shutdown mode shows that the results are less adverse than those of the full-power event documented in Chapter 15, Subsection 15.6.3. Consequently, the radiological doses are well within the 10 CFR 50.34 guidelines, and fuel integrity is not challenged.

The worst-case initial conditions for an SGTR accident initiated in a shutdown mode are in Mode 3 at the highest possible RCS temperature and pressure and lowest possible RCS flow (fewer than four RCPs operating). This maximizes the initial leak rate and flashing fraction of the leaked fluid. In addition, the worst-case single failure is the assumption that the atmospheric dump valve (ADV) on the affected SG sticks open after the operator attempts to close this valve as described in Chapter 15, Subsection 15.6.3.

Since the turbine-generator is offline in Mode 3, there is no perturbation to the electrical grid that could cause a LOOP. A consequence of a LOOP is a further reduction in coolant flow, which would increase the core exit temperature (hence increasing the flashing fraction of the leaked fluid). However, even if the very unlikely event of a LOOP is assumed, fuel integrity is not challenged since only decay heat is being generated and natural circulation is sufficient for DHR.

For an SGTR accident postulated to be initiated in a shutdown mode, the following plant parameters and alarms are available in the control room to enable the operators to diagnose the event:

- Indications of event:
  - Decreasing PZR pressure
  - Decreasing PZR liquid level
- Indication of the event and the SG that is affected:
  - Increasing SG liquid level
  - Increasing SG liquid blowdown radiation level
  - Increasing main steam line radiation level

- Additional indications of the event if the steam bypass system is being used for cooldown:
  - Increasing steam jet air ejector radiation level
  - Increasing stack radiation level

The sequence of events for an SGTR accident postulated to occur in Mode 3 is as follows:

a. Prior to the SGTR accident, the reactor is shut down according to normal procedures, and consequently, the turbine-generator is offline. A postulated iodine spike caused by a reactor trip or primary system depressurization (event generated iodine spike [GIS]) is unlikely under these conditions. However, it is assumed that either a GIS has occurred or the RCS radioactivity concentration is at its Technical Specification limit (pre-accident iodine spike) since this resulted in the most adverse RCS radioactivity concentrations when the SGTR accident was assumed to be initiated in Mode 1 (as documented in Chapter 15, Subsection 15.6.3).

   In addition, it is assumed that the secondary activity is at the Technical Specification limit of 0.1 μCi/g I-131 dose equivalent since this assumption also maximizes the radiological doses. Although a LOOP is unlikely under these circumstances, a LOOP is conservatively assumed to occur concurrent with event initiation. It is assumed that the ADVs are being used to cool down the plant, as opposed to the usual method of using the steam bypass control valves, which results in the accumulation of radiological doses from event initiation. The maximum plant cooldown rate is achievable with the partial opening of one ADV on each SG. One ADV on each SG is therefore assumed to be open at tevent initiation.

b. The SG liquid blowdown radiation monitor immediately provides the operator with information that enables the operator to determine which SG is damaged. The SG liquid blowdown radiation alarm is activated within seconds of event initiation since it is set to alarm below the 3.7 E+03 Bq/g (0.1 μCi/g) Technical Specification limit. Based on the adverse RCS and SG radioactive concentrations at event initiation, the special purpose main steam line area radiation monitors may also alarm.

   For the double-ended rupture of an SG tube, the operator also has an immediate indication of event occurrence in the form of rapidly decreasing PZR pressure and level and increasing SG level. These parameter changes result in a high SG liquid level alarm, a low PZR pressure alarm, a low PZR liquid level alarm, and possibly the high SG blowdown radiation alarm.

c. Acting upon these indications, within 30 minutes after event initiation, it is assumed that the operator diagnoses the event and knows which SG has the damaged tube. At this point, the operator isolates feedwater to the affected SG, closes the main steam isolation valves to both SGs, and attempts to close the ADV on the affected SG.

d. The operator determines that the ADV did not close (continued control room indication of steam flow) and closes the block valve upstream of the stuck open ADV within 30 minutes of the initial attempt to isolate the affected SG (i.e., within 1 hour of event initiation).

e. The operator commences cooldown to shutdown cooling entry conditions by using the unaffected SG ADV. The operator also uses the PZR gas vent and SI flow to decrease RCS pressure and control subcooling. The operator steams the affected SG as necessary to prevent overfilling.

A comparison of the above sequence of events with the worst-case SGTR accident (see APR1400 DCD Tier 2, Chapter 15, Table 15.6.3-7) shows that an SGTR accident postulated to occur in Mode 3 results in less integrated steam released to the environment because there is no release by the main steam safety

valves since there is no initial transient imbalance between reactor power and turbine power. The integrated steam release through the stuck open ADV and the flashing fraction of the leaked primary fluid, while the affected ADV is discharging to the atmosphere, are essentially the same for the postulated Mode 3 and full-power events. Thus, the concentration of radioactive nuclides in steam released to atmosphere is no greater for the Mode 3 event than for the event documented in APR1400 Chapter 15. The total radiological release for this event is therefore less than that of the full-power event.

If the SGTR is postulated to occur at lower RCS temperature and pressure than assumed above, the radiological doses are even less than the above event because less integrated steam is released to the environment and the concentration of radioactive nuclides in this released steam is lower and because of the lower fraction of primary fluid flashed to steam in the both the affected and unaffected SGs.

The indications of event occurrence and the event consequences are somewhat different if a SGTR is postulated to be initiated when the SCS is in operation in Mode 4 or Mode 5 when the RCS loops are filled. An SGTR accident does not have any consequences in Mode 6 since primary and secondary pressures are both at atmospheric pressure. The plant parameters and alarms available in the control room that enable the operator to diagnose an event under these circumstances are as follows:

- Indication of event for all leak sizes and which SG is affected:
    - Increasing SG liquid level
- Indications of event for larger leak sizes (the cooldown rate may mask the leak rate for these indications when the SGTR leak rate is small):
    - Decreasing PZR pressure
    - Decreasing PZR liquid level
    - Imbalance between charging and letdown flow rates

The leakage rate could range from 0.0 L/min (0.0 gpm) to approximately 1,798.0 L/min (470 gpm), assuming a double-ended break of one SG tube, under these circumstances. At the maximum leak rate, the operator has sufficient time in which to diagnose the event and take appropriate mitigating actions. The mitigating actions could include using the SG blowdown system to prevent overfilling the SG and minimizing the pressure difference between the primary and secondary systems. At lower leak rates, the operator has even more time to take corrective action. The SCS relief valves provide reasonable assurance that the P-T limits for brittle fracture of the RCS are not violated although they are not actuated for this event. There is no approach to the criterion of 110 percent of SG design pressure. In addition, fuel integrity is not challenged since no approach to DNB occurs. The resulting radiological doses are be much less than for the SGTR accident initiated in Modes 3 and 4 and consequently are well within 10 CFR 50.34 limits.

3.6.4 Radiological Consequences of Main Steam Line Failure Outside Containment (Boiling Water Reactor)

The radiological consequences of main steam line failure outside containment for a BWR do not apply to the APR1400 design and are therefore not evaluated.

3.7 Radioactive Material Release from a Subsystem or Component

The accidents that could result in a radioactive material release from a subsystem or component, excluding fuel handling accidents, are likely to occur in all modes of operation. In the postulated fuel handling accident, the accident takes place in the containment or in the spent fuel pool (SFP) inside the

fuel handing area in the auxiliary building during Mode 6. Therefore, the consequences of the events presented in Chapter 15, Subsection 15.7, bound the consequences of events in shutdown modes.

4.0 Applicability of LOCA Analyses to Lower Modes of Operation

4.1 Issue

Most of the analyses of the loss-of-coolant-accident (LOCA) and all criteria associated with the accident focus on scenarios starting from 102 percent or 100 percent of rated thermal power as prescribed in 10 CFR 50, Appendix K, and Regulatory Guide (RG) 1.157 (Reference 22). In this subsection, scenarios initiated from all other modes of operation are addressed to demonstrate that the analyses performed in APR1400 DCD Tier 2, Chapter 6, are the bounding cases for all modes of operation. This subsection identifies the scenarios anticipated for non-Mode 1 operation and where required, the acceptance criteria for the different scenarios.

4.2 Acceptance Criteria

The acceptance criteria for lower operating mode LOCAs is established to be the same as those for higher operating mode LOCAs in 10 CFR 50.46 (Reference 23) and are as follows:

a. Peak clad temperature less than 1,204 °C (2,200 °F)

b. Less than 17 percent peak local clad oxidation

c. Less than 1 percent core-wide oxidation

d. Maintain coolable geometry

e. Maintain long-term cooling

4.3 Description

The postulation of types of LOCAs and corresponding break sizes in APR1400 DCD Tier 2 Chapter 6, the large break LOCA (e.g., double-ended guillotine pipe breaks), is not appropriate to be considered as LOCAs for lower operation modes. Only small break LOCAs and the following lower mode scenarios are considered: most adverse misoperation of valves, the likelihood of cross-train maintenance errors, and the possibility of steam generator (SG) nozzle dam blowout. The limiting size and location for a small break LOCA for Modes 3 and 4 are a direct vessel injection (DVI) line discharge leg (371.6 cm2 [0.4 ft2]). A DVI line break at this location minimizes injection flow into the reactor coolant system (RCS).

A description of the equipment that is available to mitigate the consequences of a LOCA in each mode of operation, the effectiveness of the 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 greater than 176.7 °C (350 °F), and power is greater than 5 percent. The analysis for LOCAs in this mode concludes that all criteria are met.

Mode 2 – In Mode 2, RCS temperature is greater than 176.7 °C (350 °F), and power is less than 5 percent. 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 in meeting LOCA criteria, are proportional to the lower power level in Mode 2.

Mode 3 – In Mode 3, RCS temperature is greater than 176.7 °C (350 °F), and keff is less than 0.99.

For the case in which RCS pressure is higher than 50.3 kg/cm^{2}A (715 psia), the equipment available in Mode 1 (SIT, four safety injection [SI] pumps) is available. Because the stored energy and decay heat are lower, the parameter space in Mode 3 is bounded by Mode 1 for LOCA. When the pressure is lower than 50.3 kg/cm^{2}A (715 psia), four 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 are not available when the pressure is lower than 50.3 kg/cm^{2}A (715 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 less than 176.7 °C (350 °F).

Three subspaces are considered:

a. Conditions:

- Pressure greater than 50.3 kg/cm^{2}A (715 psia)
- SITs and two SI pumps on automatic

Conclusions:

Subspace (1) is bounded by the Mode 4, subspace 3, analysis included in this report (which credits neither SITs nor SI pumps on automatic) because SITs are available to accommodate the higher pressure and two SI pumps are available to accommodate higher pressure.

b. Conditions:

- Pressure greater than 28.1 kg/cm2A (400 psia)
- Two SI pumps on automatic

Conclusions:

Space 2) is bounded by the Mode 4, subspace 3 analysis because two SI pumps are available to accommodate the higher pressure.

c. Conditions:

- Pressure less than 28.1 kg/cm^{2}A (400 psia)

Conclusions:

An analysis (subspace 3 described in Subsection 4.3.1) that credits neither SITs nor SI pumps on automatic was performed for this subsection.

Mode 5 – RCS temperature is less than 98.9 °C (210 °F). The equipment available in Mode 4 is available in Mode 5. Because the temperature is low in Mode 5, the Mode 5 analysis yields Mode 4 analysis.

Mode 6 – RCS temperature is less than 57.2 °C (135 °F). The equipment available in Mode 5 is available in Mode 6. Because the temperature is low in Mode 6, the Mode 6 analysis yields acceptable consequences with one SI pump operable within 20 minutes.

Subsection 4.3.1 describes the primary system boundary conditions and event scenario for the postulated LOCA used in this analysis. Subsection 4.3.1 also compares and contrasts this lower operating mode event scenario to the conservative design basis licensing LOCA event normally associated with emergency core cooling system (ECCS) evaluation analyses.

Subsections 4.3.2 and 4.3.3 describe the APR1400 plant parameters and conditions for the analysis and the computer codes and analysis methods used in the LOCA calculations. Subsection 4.3.4 details the objectives and bases of the LOCA analysis.

Subsection 4.3.5 describes the results of an analysis of a limiting LOCA during Mode 4. These calculations show that hot fuel rod conditions remain in conformance with ECCS acceptance criteria during an assumed [ ]TS delay for operator action without SI pump availability. Furthermore, the analysis for the postulated LOCA shows that availability of 1 SI pump at the [ ]TS mark maintains the hot rod cladding temperatures in conformance with the ECCS acceptance criteria.

4.3.1 Description of Loss-of-Coolant Accident Scenario

Following powered operation of the nuclear steam supply system (NSSS), cooldown proceeds at the Technical Specifications maximum rate of 55.6 °C/hr (100 °F/hr) (the maximum cooldown rate shown in Reference 24). Therefore, a primary coolant temperature of 176.7 °C (350 °F) could be reached as early as 2.05 hours after shutdown.

Safety injection tanks (SIT) and SI pumps are available to mitigate LOCAs from lower modes. SITs are available for pressures higher than 50.3 kg/cm^{2}A (715 psia). SI pumps are automatically actuated for pressure higher than 28.1 kg/cm^{2}A (400 psia).

If a postulated LOCA transient occurs at pressures and temperatures slightly below these conditions (i.e., pressure less than 28.1 kg/cm2A (400 psia), temperature less than 176.7 °C (350 °F)), and time greater than 2.05 hours), the LOCA is significantly less dynamic than a design-basis LOCA transient from full-power operating conditions (i.e., 158.2 kg/cm^{2}A [2,250 psia], approximately 316 °C [600 °F]) and full fission power and associated decay heat). Factors that 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 the 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 boil-off rate.

Based on these factors, a postulated lower operating mode LOCA followed by, if necessary, timely operator action to initiate SI flow is expected to be much less severe than a LOCA from full-power conditions. The most severe lower operating mode LOCA scenario occurs for (1) the largest potential pipe break, (2) after the most rapid possible cooldown from full power, (3) after reducing pressure slightly below which no SI 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 is a significant leak in one of the DVI lines of the RCS corresponding to the flow area of the DVI line. This DVI break size of 371.6 cm2 (0.4 ft2) was chosen even though leak-before-break technology would eliminate it from scope, which covers the size and limiting location of all traditional small break LOCAs. Decay heat levels based on a period of 2.05 hours after shutdown is assumed. No operator action for is also assumed. Forced circulation through the core during lower mode operation tends 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 SGs considerably benefits the RCS heat removal process for a postulated LOCA; therefore, for conservatism, the SG secondary sides are isolated for the event.

4.3.2 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. RCS pressure was selected at the maximum at which no SI pumps would automatically actuate due to the safety injection actuation system (SIAS) and no SITs would be available. RCS temperature was selected at the maximum in Mode 4. Table 4.3-1 is a list of initial conditions.

4.3.3 Analysis Computer Codes

An analysis of computer codes required the selection of realistic inputs such as decay heat to provide as much realism in the representation of the system transient response during a LOCA as possible. An adaptation of the realistic conditions for a small break LOCA was selected for this reason. For design basis LOCA calculations from plant initial primary pressures of 158.2 kg/cm^2A (2,250 psia), the largest break size analyzed using the CEFLASH-4AS code has historically been a 465 cm^2 (0.5 ft^2) break. For the 371.6 cm^2 (0.4 ft^2) DVI line lower mode LOCA analysis, the RELAP5/Mod 3.3 in NUREG/CR-5535 (Reference 25) was used. A realistic model for decay heat was used, which is based on the 1979 American Nuclear Society Standard 5.1 (Reference 26).

4.3.4 LOCA Analysis for Mode 4

The LOCA analysis examines the hot rod heatup response during a LOCA with respect to the ECCS acceptance criteria of peak cladding temperature and peak local cladding oxidation. The objective of this analysis is to determine whether the calculated response of the hot rod remains is in conformance with the ECCS acceptance criteria crediting operator action within [ ]TS

A bounding analysis, starting at 2.05 hours after shutdown, allows the LOCA to proceed without safety injection. Primary coolant inventory is assumed to be lost through an opening in the DVI line at the vessel penetration to the upper annulus primary piping. When core 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.

A limiting axial power shape is assumed for the LOCA. At 7,380 seconds (approximately 2.05 hours), the coolant inventory and consequently core cooling is reduced. Figure 4-1 shows the resulting hot rod clad temperature with [ ]TS after break.

4.3.5 Results of Loss-of-Coolant Accident Analysis for Mode 4

The LOCA event is analyzed for realistic or credible thermal-hydraulic blowdown conditions. The objective of this analysis was to determine requirements for Technical Specifications to provide reasonable assurance of adequate cooling flow in lower modes of operation. The realistic analysis examined the RCS primary coolant and hot fuel rod transient response during a LOCA.

For the analysis, a break in one of the DVI lines is postulated. For traditional small break scenarios, this is the largest and most limiting location for a break in the primary system because a break in the DVI line dictates that SI flow that is delivered through this line is lost to the containment. The flow area of the DVI line is 371.6 cm^2 (0.4 ft^2) and therefore the largest effective break area.

For this analysis, modified input for the large break LOCA of RELAP5/Mod 3.3 code is used to calculate the thermal-hydraulic system response. With boundary conditions in the RELAP5/Mod 3.3, a cooldown from full power is simulated to achieve a set of initial conditions at 2.05 hours. The calculation is then restarted at this time with the break size indicated above to study the system response during the LOCA.

The primary system at the start of the LOCA is at 28.1 kg/cm^2A (400 psia) and 176.7 °C (350 °F). The

reactor coolant pumps have been stopped, and the SGs (steam dump and bypass system) have been isolated. The operation of the SCS is also not simulated.

Because it is not safety grade, no charging flow is credited in the calculations. The first sets of calculations are made with no SI delivery. The results of these calculations are given in Subsection 4.3.5.1. A second calculation for the limiting break size is then made where SI delivery is initiated within [ ]TS. The results of this calculation are given in Subsection 4.3.5.2.

4.3.5.1 Results of Loss-of-Coolant Accident with No Safety Injection Delivery

This subsection presents the thermal-hydraulic results of a LOCA break size of 371.6 cm^2 (0.4 ft^2). The break is postulated to occur in a DVI line at the vessel penetration. LOCA transient results are shown up until the peak clad temperature (PCT) exceeds 605.8 °C (1,122.4°F) or approximately [20 minutes]TS from initiation of break. After that, the code was terminated. The purpose of the analysis is to define the RCS inventory makeup requirements. The 371.6 cm^2 (0.4 ft^2) break size is the cross-sectional area of the DVI line. For traditional small break scenarios (full power, Mode 1), this is the most severe break location and is also the limiting location for lower modes of operation as shown in the following subsection.

4.3.5.1.1 Primary Pressure

The primary system pressure transient for the 371.6 cm^2 (0.4 ft^2) break size at the DVI line location is shown in Figure 4-2. The subcooled decompression from 28.1 kg/cm^{2}A (400 psia) to 11.95 kg/cm^{2}A(170 psia) occurs within the first minute. The depressurization proceeds until the steam production in the core and its volume expansion from superheating due to the combination of secondary-to-primary SG heat transfer, core heat transfer from uncovered fuel rods, and from wall heat transfer exceeds the steam volumetric break flow. The primary system then begins a gradual depressurization.

4.3.5.1.2 Reactor Vessel Collapsed Liquid Level

Figure 4-3 shows the collapsed liquid level in the reactor vessel (RV) for the DVI line break. Inventory is lost above the level of the hot leg elbow within the first several minutes of the LOCA. The collapsed liquid level remains at this level near the hot leg centerline (7.89 m [25.87 ft]) until the SG U-tubes and hot legs become empty. During this draining period, steam produced by heat transfer from the fuel rods swells the collapsed level in the core and is released at the surface to make its way down the hot leg countercurrent to the draining liquid and around the U-tubes through the loop seal and pump to the break in the DVI line. Figure 4-3 shows that the collapsed liquid level drops from time of break due to boil-off.

4.3.5.1.3 Hot Spot Cladding Temperature

The cladding temperature result for the hot rod is shown in Figure 4-4. For the limiting DVI line break location, before the dry-out of the top of core, the cladding temperature follows the saturation temperature of the primary coolant. With decreasing core level, cladding heatup begins first at elevations near the top of the core and then progresses to lower elevations in the core, which have more local power. Figure 4-4 shows the clad surface temperature, which is located approximately 5 percent of the active core height below the top of the core. The time at which the hot spot cladding temperature exceeds 605.8 °C (1,122.4 °F), and the code is terminated.

4.3.5.2 Initiation of SI Delivery after Break

The results presented below show that for break sizes of 371.6 cm^2 (0.4 ft^2) or less, operator action within [ ]TS to initiate SI injection mitigates the postulated LOCA (i.e., prevents the cladding temperature from exceeding 1,204 °C (2,200 °F)). To determine the effectiveness of restoring cooling flow to the RCS, the 371.6 cm^2 (0.4 ft^2) break size was postulated to occur at DVI line. [ ]TS after the initiation of the

LOCA, a cooling flow of 63kg/sec (137 lbm/sec) is credited (equivalent to minimum runout flow of one SI pump).

For the DVI line break, Figure 4-5 shows that injected cooling flow restores a positive core flow and causes the collapsed liquid level in the core to recover the fuel rods. Figure 4-6 shows vapor void fraction at a peak clad temperature (PCT) node. The fuel rods are recovered within [23 minutes]TS from time of injection. Figure 4-7 shows clad surface temperature for the postulated DVI line break. The PCT with cooling flow restored in [ ]TS after break is compared with the case without cooling flow restoration. The restoration of cooling flow within [ ]TS prevents the cladding temperature from reaching the point of significant Zircaloy oxidation and a runaway cladding heatup. A lack of significant Zircaloy oxidation results in the maintenance of core geometry after LOCAs in lower modes of operation. The post-LOCA long-term cooling analysis for Mode 1 operation bounds long-term cooling analyses for LOCAs in lower modes because the decay heat levels are significantly lower. The results for this Mode 4 LOCA analysis show that restoration of a cooling flow of 63 kg/sec (132 lbm/sec) (equivalent to minimum runout flow of a SI pump) within [ ]TS of the start of a LOCA mitigates the consequences for break sizes of at least 371.6 cm^2 (0.4 ft^2).

4.4 Resolution

In this section the Mode 4 analysis result is presented. For the lower mode LOCAs not bounded by Mode 1 analyses, this Mode 4 analysis is bounding.

The analysis presented in Section 4 demonstrates that a LOCA event from T_{RCS} 176.7 °C (350 °F) and P_{RCS} 28.1 kg/cm^2A(400 psia) can be mitigated provided the equivalent of at least one SI pump is injecting to the RCS no later than [ ]TS after the start of the LOCA. For lower mode LOCAs not bounded by Mode 1 analyses, the break size and location analyzed are limiting for small breaks that occur in lower modes with neither SI pumps on automatic nor SITs available. For the case where SI pumps are not on automatic, the transient response for the postulated break in the DVI line nozzle of 371.6 cm^2 (0.4 ft^2) shows that in approximately [ ]TS from time of break, the collapsed liquid level drops below the top of the core and without cooling flow injection, the cladding temperature exceeds 605.8 °C (1,122.4 °F), and the code terminated. The time interval for exceeding ECCS acceptance criteria for PCT is considerably larger than the [ ]TS delay assumed for operator action. Due to the [ ]TS requirement for deliveries of injection flow, a Technical Specification to require operability of two SI trains in Mode 4, 5, and 6 is added. Provided that cooling flow equivalent to one SI pump can be provided within [ ]TS after a lower mode LOCA, it is concluded from the results of the analysis that the acceptance criteria for LOCAs detailed in Subsection 4.2 are met.

Table 4.3-1

APR1400 LOCA Analysis Mode 4 Initial Conditions

Parameter	APR1400 DC
Rated core power	3983 MWt
Time to enter mode 4 after shutdown	> 2.05 hour
Initial pressure	28.1 kg/cm^2A (400 psia)
Initial temperature	176.7 °C (350 °F)
SI pumps	Not on automatic
SITs	Not available
Steam generator	Isolated
Reactor coolant pumps	Tripped before start of event
Initial core mass flow rate	160.0 kg/sec (352.7 lb/sec)
Initial pressurizer liquid volume	0.0 m^3 (0.0 ft^3)
Normal onsite or offsite electrical power after turbine trip	Unavailable
DVI break area	0.037 m^2 (0.4 ft^2)
Moderator temperature coefficient, $	0

Table 4.3-2

Time Sequences of LOCA Analysis

TS

TS

Figure 4-1 Clad Surface Temperature Safety Injection, [ ]TS DVI Line Break

TS

Figure 4-2 Pressurizer Pressure No Safety Injection DVI Line Break

TS

Figure 4-3 Reactor Vessel Collapsed Liquid Level No Safety Injection DVI Line Break

TS

Figure 4-4 Clad Surface Temperature No Safety Injection DVI Line Break

TS

Figure 4-5 Reactor Vessel Collapsed Liquid Level Safety Injection,
TS
DVI Line Break

TS

Figure 4-6 Core Vapor Void Fraction at PCT Node Safety Injection, TS DVI Line Break

TS

Figure 4-7 Comparison of Clad Surface Temperature,With and without Safety Injection DVI Line Break

5.0 Applicability of the APR1400 Design Certification Chapter 6 – Containment Analyses to the Shutdown Evaluation

APR1400 DCD Tier 2, Chapter 6, Subsection 6.2.1.1 describes the containment functional design. A series of loss of coolant accidents (LOCAs) and main steam line breaks (MSLBs) were analyzed to determine the 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 NUREG-0800, “Standard Review Plan”, a series of LOCAs was analyzed at 102 percent power and a series of MSLBs were analyzed at powers from 102 percent to 0 percent. The 102 percent Double Ended Discharge Leg Slot (DEDLS) break with the maximum ECCS flow is the design basis event (DBE) for the APR1400. These cases are presented in APR1400 DCD Tier 2, Chapter 6.

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 issue that the DBE identified from the containment P/T analysis, which is presented in APR1400 DCD Tier 2 Chapter 6, was limiting. However, some safeguards equipment is removed from service at lower modes. Table 5.1-1 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 0 percent power (Mode 2) was analyzed to show the 102 percent power cases remain limiting for both containment pressure and equipment environmental qualification. This report describes the results of these evaluations. The results show that the 102 percent power cases remain limiting for both containment pressure and equipment environmental qualification. Table 5.1-2 presents a list of cases that were considered. These cases were analyzed for a design core power level of 3,983 MWh.

5.1 Loss of Coolant Accidents

APR1400 DCD Tier 2 Chapter 6 presents the results of the containment pressure and temperature analysis for a series of hot leg, suction leg, and discharge leg LOCAs. Consistent with NUREG-0800, Section 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 DEDLS break, which yields the maximum peak pressure of 51.09 psig (3.592 kgf/cm^2) and the maximum temperature of 134.6 °C (274.3 °F). This pressure was the highest of any LOCA or MSLB and this case is the containment DBE.

As power level decreases from 102 percent, the hot leg coolant and core temperatures decrease. At the same time, the total mass and pressure on the SG secondary sides increases. As a result, primary side stored energy decreases and secondary side energy increases. To show that the cases presented in APR1400 DCD Tier 2 Chapter 6 are limiting, LOCAs initiated at 0 percent power have been analyzed. Although the case in Chapter 6 that produced the highest LOCA containment peak pressure was the DEDLS break with maximum safety injection, the Double Ended Suction Leg Slot (DESLS) break with maximum safety injection were also chosen for this analysis. Table 5.2-1 lists the assumptions and initial conditions for this case. This initial condition is actually for the Mode 2 case. With the RCS cold leg coolant at 295 °C (563 °F), it represents the most NSSS energy of any Mode 2 case. Containment pressure and temperature calculation for this analysis was conducted using the GOTHIC computer code. The LOCA mass and energy release from the blowdown (CEFLASH-4A) and from the reflood and post-reflood (FLOOD3) were incorporated into GOTHIC input to produce the APR1400 containment pressure and temperature. For the containment initial conditions are consistent with APR1400 DCD Tier 2 Chapter 6. Tables 5.2-2 and 5.2-3 provide the chronology of events for the DESLS and the DEDLS break, respectively. The containment peak pressures in Mode 2 are 3.399 kgf/cm^2 (48.35 psig) for the DEDLS break and 3.459 kgf/cm^2 (49.20 psig) for the DESLS break compared to 3.592 kg/cm^2 (51.09 psig) of the

limiting case (DEDLS) at 102 percent power presented in APR1400 DCD Tier 2 Chapter 6. Results for the DEDLS and the DESLS break are shown in Figures 5.2-1 and 5.2-2.

For Modes 3 and 4, the NSSS stored energy is less than for the cases analyzed above. The 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 99 °C (210 °F), a LOCA in Modes 5 and 6 would not result in any significant containment pressurization.

5.2 Main Steam Line Breaks

In APR1400 DCD Tier 2 Chapter 6, MSLBs were analyzed at 102, 75, 50, 20, and 0 percent power, representing Modes 1 and 2. For the events in Chapter 6, the Technical Specifications require that all trains of the safeguards system must be operational. A single active failure is considered in the analysis. The cases were analyzed with either the failure of an Main Steam Isolation Valve (MSIV) or the loss of a containment spray train. The 75 percent power in Mode 1 with the loss of a containment spray train produced a containment peak pressure of 49.10 psig (3.452 kgf/cm^2).

The 0 percent power cases with loss of a CS train and MSIV single failure, which are the Mode 2, are included in the analyses for the Containment maximum P/T presented in APR1400 DCD Tier 2 Chapter 6 since those cases have the greatest coolant stored energy of SGs secondary side based on the pressure of 84.37 kgf/cm^{2}A (1,200 psia) and temperature of 295 °C (563 °F).

In Modes 3 and 4, the NSSS stored energy is less than Modes 1 and 2. As shown in Table 5.1-1, main steam isolation and containment spray are still available in these modes. Therefore, an MSLB during these modes would not be more limiting than an MSLB during Mode 1 or Mode 2.

In Modes 5 or 6, main steam isolation and containment spray may not be available. On the other hand, the RCS coolant temperature in Mode 5 is less than 99 °C (210 °F). If the SG coolant temperature was also at 99 °C (210 °F), no containment pressurization would occur following an MSLB. An analysis has been performed conservatively assuming that the temperature of the SG secondary coolant was still at 177 °C (350 °F) following the shutdown cooling of the RCS which reduced the RCS temperature to 99 °C (210 °F). Table 5.3-1 lists the assumptions and initial conditions. Table 5.3-2 lists the chronology of events. The containment peak pressure for this case is 13.49 psig (0.948 kgf/cm^2), well below the DBE result. Results for this Mode 5 MSLB are shown on Figure 5.3-1.

APR1400 DCD Tier 2 Chapter 6 provides containment analyses to support the establishment of the containment design pressure and temperature and an envelope for equipment environmental qualification. Various spectrums of primary and secondary line breaks were analyzed. With exception of the 0 percent power of MSLB cases (Mode 2), all of the cases analyzed were for Mode 1. As described 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 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 APR1400 DCD Tier 2 Chapter 6 even with containment spray and main steam isolation unavailable.

5.3 Inadvertent Operation of Containment Heat Removal Systems

The inadvertent containment spray actuation event presented in APR1400 DCD Tier 2 Chapter 6 is used to determine the maximum external containment design pressure. 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 results 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 spray water is provided by the CS pump from the IRWST. The lowest IRWST water temperature within the range of limiting conditions for operation (LCO), which is irrelevant to the operation mode, is assumed in the analysis.

5.4 Conclusion

APR1400 DCD Tier 2 Chapter 6 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 was analyzed. With the exception of the 0 percent power MSLB case (Mode 2), all of the cases analyzed were for Mode 1.

As described above, NSSS stored energy decreases in going from Mode 2 to Mode 6. Safeguards equipment important to containment integrity (containment spray and main steam isolation) are available in all modes with the possible exception of Modes 5 and 6. The analyses in this report 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 the limiting case presented in APR1400 DCD Tier 2 Chapter 6 even with containment spray and main steam isolation unavailable.

The inadvertent containment spray actuation event is used to determine the maximum external containment design pressure. As described above, computer runs were made to analyze the containment transient for shutdown modes. The most restrictive case in LOCA for Mode 2 is the DEDLS break with a maximum ECCS flow, which yields a peak pressure of 3.459 kgf/cm^2 (49.20 psig) and a temperature of 133.17 °C (271.71 °F). For the MSLB case for mode 5, the containment peak pressure is 0.948 kgf/cm^2 (13.49 psig). And the peak temperature is 108.57 °C (227.42 °F). After 30 minutes, the operator acts to terminate the increase in containment pressure. The results of the containment P-T analyses for shutdown modes are summarized in Table 5.5-1 and shown on Figures 5.2-1, 5.2-2 and 5.3-1.

Table 5.1-1

ESFAS Instrumentation

Signal	Applicable Modes
Containment spray actuation signal	1, 2, 3, 4
Main steam isolation signal	1, 2, 3, 4

Table 5.1-2

Cases Analyzed

Mode	LOCA	MSLB
1	DESLSB, DEDLSB, DEHLSB [(1)] 102 percent power	102 percent, 75 percent, 50 percent, 25 percent power Loss of a CSS or MSIV single-fail.
	Containment max. P/T analysis (Calculation: 1-310-N380-003)	Containment max. P/T analysis (Calculation: 1-310-N380-003)
2	DESLSB, DEDLSB 0 percent power	0 percent power Loss of a CSS or MSIV single-fail.
	Shutdown risk analysis (Calculation:1-035-N312-007)	Containment max. P/T analysis (Calculation: 1-310-N380-003)
3, 4 [(2)]	Not required	Not required
	Less limiting than Mode 2	Less limiting than Mode 2
5	LOCA at full inventory (not required)	Analyzed under the conditions of RCS(210 °F) and SG(350 °F)
	RCS Temp. < 210 °F	Shutdown risk analysis (Calculation: 1-035-N312-007)
6	Inventory boil-off at refueling pool (Not Required)	Not required
	RCS Temp. < 135 °F	Less limiting than Mode 5

(1) DESLSB : Double-ended suction leg slot break
DEDLSB : Double-ended discharge leg slot break
DEHLSB : Double-ended hot leg slot break

(2) Less NSSS stored energy than Modes 1 and 2.
Same ESF available as in Mode 1 and 2.

Table 5.2-1

Initial Conditions for LOCA Initiated from Mode 2

Parameter	Value
RCS cold leg coolant temperature	295 °C (563°F)
Reactor thermal power	228.35 MWth
SG secondary side pressure	75.9 kg/cm^{2}A (1080 psia)
Containment atmosphere	1.13 kg/cm^{2}A, 49 °C (16.12 psia, 120 °F), 0 percent Relative humidity

Table 5.2-2

Accident Chronology for Mode 2 DESLSB with Maximum ECCS

Time (sec)	Chronology
0.0	Break occurs
6.55	Containment pressure Hi-Hi setpoint, 1.54 kg/cm^2 (22 psig)
18.61	First peak containment pressure (Blowdown), 2.8 kg/cm^2(39.94 psig)
	End of blowdown
116.55	Start containment spray injection
127.52	Peak containment temperature, 132.6 °C (270.69 °F)
128.12	Peak containment pressure, 3.4 kg/cm^2 (48.35 psig)
1000.0	End of analysis (End of post-reflood)

Table 5.2-3

Accident Chronology for Mode 2 DEDLSB with Maximum ECCS

Time (sec)	Chronology
0.0	Break occurs
5.54	Containment pressure Hi-Hi setpoint, 1.54 kg/cm^2 (22 psig)
18.91	First peak containment pressure (blowdown), 2.84 kg/cm^2 (40.39 psig)
	End of blowdown
115.54	Start containment spray injection
304.02	Peak containment temperature, 133.2 °C(271.71 °F)
304.22	Peak containment pressure, 3.46 kg/cm^2 (49.20 psig)
1000.0	End of analysis (End of post-reflood)

Table 5.3-1

Initial Conditions for MSLB Initiated from Mode 5

Parameter	Value
RCS average coolant temperature	99 °C (210 °F)
SG secondary side pressure	9.463 kg/cm^{2}A (134.6 psig)
SG secondary side temperature	177 °C (350 °F)
Containment atmosphere	1.13 kg/cm^{2}A, 49 °C (16.12 psia, 120 °F), 0 percent relative humidity

Table 5.3-2

Accident Chronology for Mode 5 MSLB

Time (sec)	Chronology
0.0	Break occurs
86.01	Peak containment temperature, 108.6 °C(227.42 °F)
129.01	Peak containment pressure, 0.95 kg/cm^2 (13.49 psig)
1,800.0	End of analysis (end of blowdown)

Table 5.5-1

Summary of Results of P/T analyses for Mode 2 and Mode 5

Run No.	Break Identification	Peak Pressure		Peak Temperature	
		kg/cm^2 (psia)	sec	°C (°F)	sec
1	DESLSB - max. ECCS (Mode 2)	4.433 (63.05)	128.12	132.60 (270.69)	127.52
2	DEDLSB - max. ECCS (Mode 2)	4.492 (63.90)	304.22	133.17 (271.71)	304.02
3	MSLB (Mode 5)	1.982 (28.19)	129.01	108.57 (227.42)	86.01

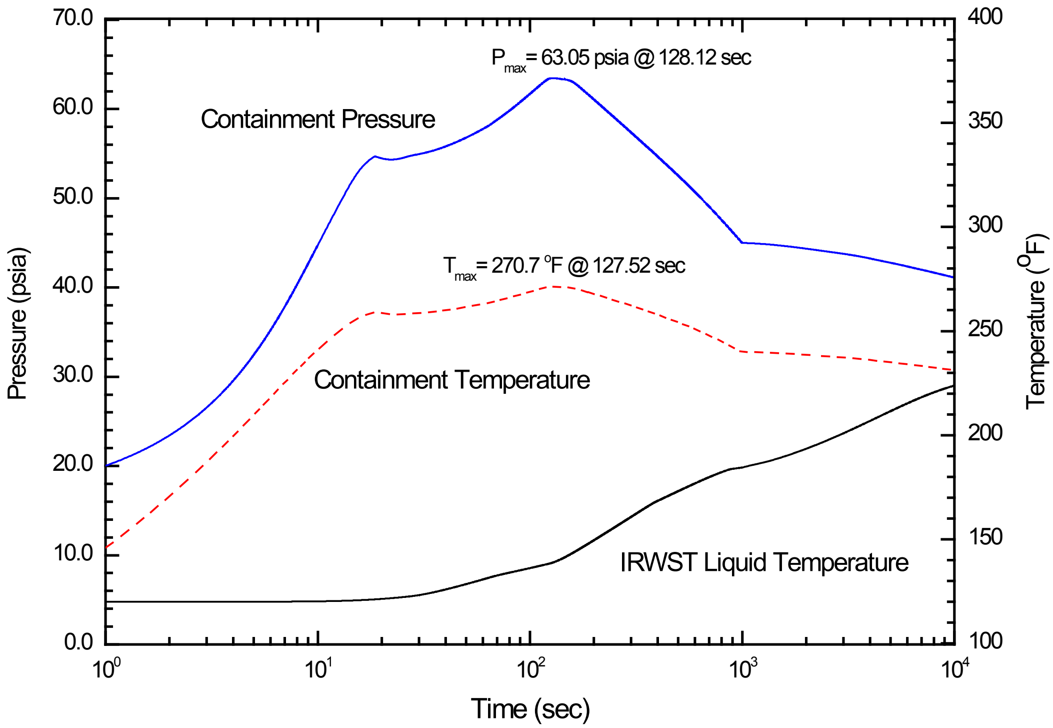

Figure 5.2-1 Containment Pressure and Temperature (DESLSB with Max. ECCS in Mode 2)

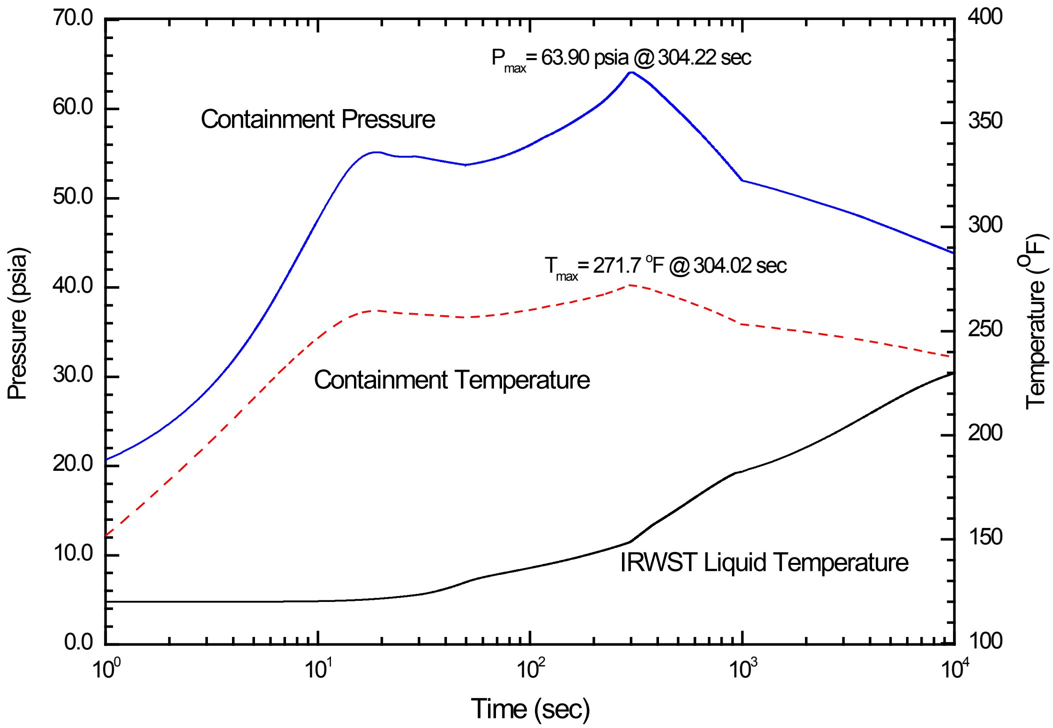

Figure 5.2-2 Containment Pressure and Temperature (DEDLSB with Max. ECCS in Mode 2)

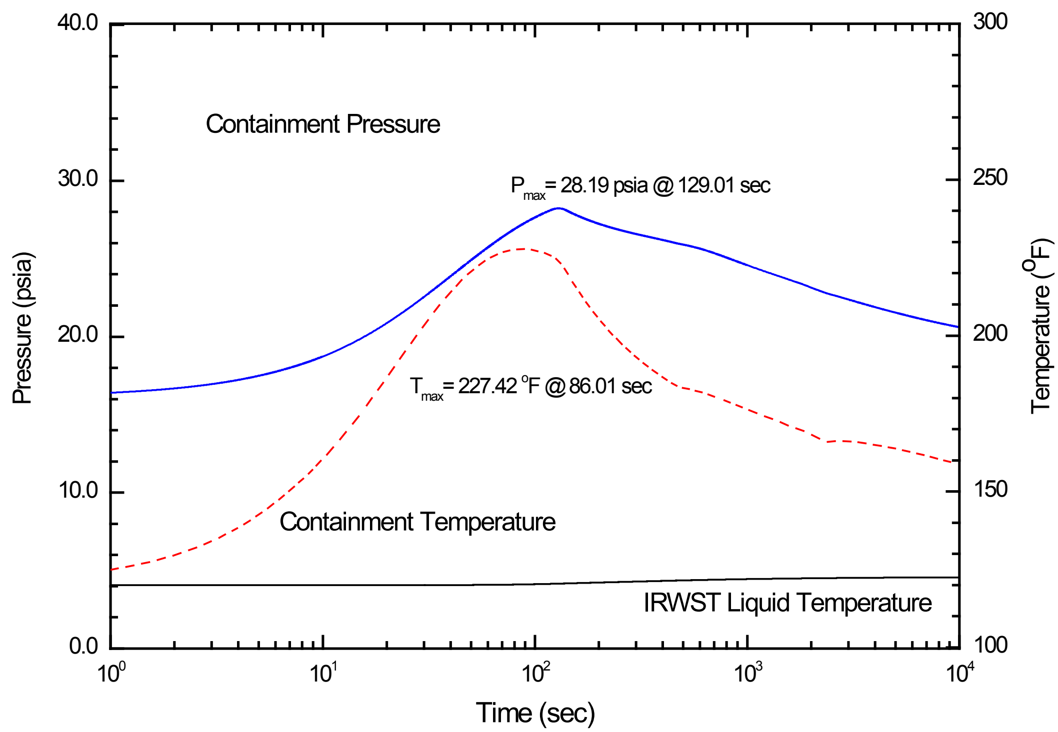

Figure 5.3-1 Containment Pressure and Temperature for Mode 5 MSLB

6.0 APR1400 Design Features for Simplicity of Shutdown Operations

6.1 Introduction

The APR1400 evolutionary advanced light water reactor (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 the APR1400 that improve shutdown operations. In the following subsection, the outage benefits accruing from these design features are presented.

6.2 Description

There are several features of the APR1400 design that aid in the management of an outage. Many of these features also relieve some of the stresses and pressures placed on licensed operators. These features are presented in Subsections 6.2.1 through 6.2.9.

6.2.1 Technical Specifications for Reduced Inventory

Technical Specifications for the APR1400 design specify restrictions on operation in reduced inventory. Reduced inventory is determined by water level in the reactor coolant system (RCS). This plant operation condition is defined as “RCS level lower than 91.44 cm (3 ft) below the reactor vessel flange.” These Technical Specifications provide guidance to the operations staff and the outage management team. This guidance provides reasonable assurance 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. The planning process reduces the chance for removal of components relied upon for detection and response to accidents. The specifications provide the senior licensed operator with a clear standard for determining minimal equipment availability, which alleviates some of the stress placed on the operator to make the final judgment as to whether equipment is required to be operable for reduced inventory operations.

6.2.2 Shutdown Cooling System

The shutdown cooling system (SCS) design provides two divisions for decay heat removal (DHR) capability. The divisions are completely redundant (i.e., they share no components or equipment). This redundancy provides the operator with a standby DHR system if any component fails to perform its function. A reasonable assurance of 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 (ECCS). Therefore, the SCS is not required to be operable in Modes 1 through 3. This feature of APR1400 design increases the availability of SCS during Modes 4 and 6. 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 also eliminates the necessity of finding a window during the outage to allow work on DHR equipment.

Besides the features described 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, which 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 DHR is required.

6.2.3 Containment Spray System

The containment spray system (CSS) design provides two divisions of equipment that can be used as an alternate DHR flow path. The CSS pumps are interchangeable with the SCS pumps. This feature provides the operations staff with increased flexibility in 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 APR1400 DHR capability. With these alternate pumps, the operators have reasonable assurance that redundant equipment is available.

6.2.4 Component Cooling Water System

The component cooling water system (CCWS) design has two redundant divisions. Each division 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, 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.

6.2.5 Essential Service Water System

The essential service water system (ESWS) design has two redundant divisions. Each division contains two pumps. This interdivisional redundancy of the pumps provides 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.

6.2.6 Electrical Distribution System

The electrical distribution system (EDS) design has two redundant safety divisions (see Figure 2.4-3.) Each division is capable of being powered from four separate and diverse sources during shutdown modes and reduced inventory operations. These sources include:

a. Preferred Offsite Power I

b. Preferred Offsite Power II

c. Diesel Generators

The EDS provides the outage planner with the flexibility to remove a source of power for maintenance and 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 features also provide the licensed operator with alternate sources to which safety buses can be aligned. The operator is aware of the approved alternate alignments through procedures and training. Therefore, the stress on 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 four safety buses (two per division). The 1E loads are evenly distributed on the buses to provide reasonable assurance of redundancy of system components. For example, each bus powers one of four component cooling water (CCW) 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.

6.2.7 Human System Interface (HSI) System

The human system Interface system (HSIS) is an integral part of the APR1400 design. The design goals

of HSIS 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 the HSIS is consistent with operator information requirements when performing operational tasks during plant evolutions or responding to unexpected conditions. The operator can obtain information from the following sources in the HSIS:

- Large plant overview status board (large display panel [LDP])
- Alarm tiles and associated message windows
- Discrete indicators that provide frequently useful and important information.
- Cathode ray tube (CRT) displays containing essentially all power plant information.

APR1400 DCD Tier 2 Chapter 18 provides further information on the HSIS.

The HSIS uses the same parameter conventions for the indicators and alarms for shutdown operations as required for power operations. These features simplify operator training. Aid in outage planning and reduce operator stress are as follows:

- LDP – The LDP provides the operators, especially senior reactor operators, with an overview of plant status during shutdown conditions. The overview allows the operators to view system status during outage activities. Having an overview of the plant reduces uncertainty of the availability of required safety systems. Knowledge of the availability reduces the stress placed on the operators by uncertainties.
- Alarms – Mode and equipment dependent alarms are a special feature of HSIS. This feature eliminates the alarming of alarms not applicable to the current mode, operating conditions, or equipment status. A large number 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 use the same indicators on a simulator. It also eliminates the use of indicators solely for shutdown. Discrete indicators receive multiple channel input signals to be displayed on one indicator. These signals are validated and provide 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 HSIS are arranged in a structured information hierarchy. The structure provides the operator with information that is consistent with operational needs. Levels of display information start with the LDP and continue through detailed plant information. A feature of the CRT display is graphic representation of systems, which 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 flow path status. This feature also provides status of maintenance in progress on important valves in the plant.

6.2.8 Reduced Inventory Instrumentation

The APR1400 design provides the instrumentation to provide reasonable assurance that the control room operator (CRO) is informed of the decay heat removal (DHR) system performance and the reactor coolant system level and temperature. The instrumentation includes:

a. Reactor coolant system level

   (1) Heated junction thermocouple

   (2) dP

b. Reactor coolant temperature

   (1) Core exit thermocouple

   (2) Heated junction thermocouple

c. Shutdown cooling system flow

d. Shutdown cooling system pressure

e. Shutdown cooling system temperature

f. Shutdown cooling pump motor current

See Subsection 2.8 of this report for a description of this instrumentation. The instrumentation is coupled with the APR1400 control room to provide the CRO with indications and alarms to monitor reduced inventory operations. (See Subsection 6.2.7 for a description of the APR1400 features).

The CRO has the responsibility to keep the reactor core cooled and covered during reduced inventory operations. Although the burden of responsibility placed on the CRO cannot be relieved, the CRO's tasks are made simpler to perform. With the APR1400 instrumentation, the operator is not required to rely on local gauges and/or temporary instruments installed only for shutdown operation. This design increases the operator's confidence level of parameter indications and therefore decreases the stress operators are placed under when in reduced inventory operations.

6.2.9 Containment

The containment for the APR1400 is a cylindrical concrete containment with a diameter of 45.7 m (150 ft), carbon steel liner plate, and a free volume of 88,575 m^3 (3.128×10^6 ft^3). The APR1400 design incorporates a large operating deck inside the containment. The design of the containment operating deck includes open floor space allocated for storage and maintenance during outages. Portions of the operating deck are suitable for prestaging and laydown of equipment in support of maintenance activities. Examples of items that may be prestaged are:

- Valve maintenance tools and test equipment

- Reactor vessel and head equipment

- Steam generator inspection and repair equipment

Prestaging equipment required for containment work activities eliminates having to open the equipment hatch unexpectedly due to work schedule and scope changes.

The layout of the operating deck allows for multiple maintenance activities during outages and reduces the number of times the equipment hatch must be opened. Valve maintenance, reactor vessel stud cleaning, and reactor coolant pump work are performed in shielded areas on the perimeter of the operating deck. The space provided for these maintenance activities combined with crane support and prestaged tools eliminates the need to transfer components through the equipment hatch to work spaces outside containment.

The elimination of equipment hatch openings during critical plant evolutions such as RCS reduced inventory operations helps to provide reasonable assurance containment integrity is maintained. The design simplifies outage planning by providing space for prestaging and work activities to be accomplished in parallel without reliance on equipment hatch opening.

6.3 Conclusion

The APR1400 design incorporates features that are consistent with the objective provided in Subsection 6.1. The features provide the outage management team with redundancy, diversity, and guidance to simplify outage planning. In addition, these features provide the operator with reliable and redundant equipment and indications. The features also provide the operator with some relief from stresses when no redundancy or parameter indication is available. Operator training is simplified due to equipment redundancy and diversity available in shutdown operations.

7.0 Conclusions

This report provides responses to issues on shutdown risk. The report shows that the APR1400 design features, along with Technical Specifications and operational guidance, result in acceptable consequences from accidents during shutdown.

APR1400 is engineered with features that enhance shutdown safety as follows: (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 procedural guidance and Technical Specifications that address important shutdown evolutions. This report shows that these features successfully prevent the occurrence of shutdown risk events or mitigate the consequences of the events.

The probabilistic safety assessment shows that the core damage frequency for internal events while the plant is in shutdown modes is about the same as the frequency while the plant is at power operation. Evaluations of APR1400 DCD Tier 2 Chapters 6 and 15 events while in shutdown modes show that all acceptance criteria are met. In addition to the Chapters 6 and 15 analyses, the APR1400 design was reviewed with regard to other issues such as flooding and spills, fires, dropping heavy loads, and containment closure times. Again, all acceptance criteria are met.

As a result of the shutdown risk study, several changes to the Technical Specifications have been made. In addition, two design changes provide improved RCS water level indication during mid-loop operation. One of these changes is the addition of two narrow range delta P cells between the DVI nozzle and the shutdown cooling nozzle at the hot leg. The other change is the addition of two heated junction thermocouple strings to cover the hot leg region. These changes provide a more accurate measurement of the water level in the vicinity of the hot leg elevation.

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APPENDIX A
PROCEDURAL GUIDANCE TO SUPPORT REDUCED RCS INVENTORY OPERATIONS

TABLE OF CONTENTS

Appendix A : Procedural Guidance to Support Reduced Reactor Coolant System Inventory Operations

A1.0 OBJECTIVE

Appendix A provides guidance to develop reduced inventory operating guidelines and procedures. It contains information provided by the plant designer based on an analysis and review of reduced inventory operations.

A2.0 INITIAL CONDITIONS

A.1.1 The earliest time to enter reduced inventory operation is 4 days for shutdown from full power.

A2.1 The reactor is subcritical, [K_{eff} less than 0.99] for greater than [96 hours].

A2.2 Reactor coolant system (RCS) cold leg temperature [less than 98.9 °C (210 °F)].

A2.3 RCS level greater than El. [119 ft 1 in].

A2.4 Technical Specification surveillance requirements for reduced inventory are met.

A2.5 Maintenance activities are not being performed on the shutdown cooling system (SCS) or the operable containment spray pump.

A3.0 PRECAUTIONS

A3.1 Reduced inventory operations duration should be minimized to reduce risk of core uncovery due to the loss of decay heat removal (DHR).

A3.2 Perturbations affecting reactor coolant system (RCS) level should be minimized during reduced inventory operations to minimize the possibility of loss of DHR capabilities.

A3.3 Isolation (closure of a containment isolation valve) in the non-operating SCS loop can reduce the possibility of an inadvertent draindown to the RCS.

A3.4 Operations directly affecting the reactor vessel (RV) pressure boundary (i.e., in-core instrumentation seal table evolutions) are prohibited during mid-loop operations.

A3.5 Disassembly and removal of the RV head should be avoided when mid-loop operation is planned and these evolutions affect the operability of the high resolution level instrumentation used for determining RCS water level in the hot leg region.

A4.0 OPERATIONAL GUIDANCE

A4.1 Verify RCS vent path established.

A4.2 Verify that the shutdown cooling/containment spray cross connection isolation valves are administratively closed.

A4.3 Perform the RCS drain procedure to lower the RCS level to the desired reduced inventory elevation identified below:

Scheduled Maintenance Activity RCS	Elevation
S/G cold leg nozzle dams	[ ]
S/G hot leg nozzle dams	[ ]
RCP seal housing removal	[ ]
DVI nozzle 2A or 2B valve maintenance	[ ]

A4.4 Monitor the following RCS/SCS system parameters during reduced inventory operations.

RCS core exit temperature	[List instruments]
SCS system flow rate	[ ]
RCS boron concentration	[ ]
SCS system temperature	[ ]
RCS pressure	[ ]
RCS level	[ ]

NOTE

Decay heat production decreases steadily with time after shutdown. SCS flow rate should be throttled to match heat removal requirements to reduce the possibility of vortexing.

A4.5 Adjust SCS flow rate to match DHR requirements. Minimum flow must be maintained at greater than (11.4 m^3/min (3,000 gpm)).

A4.6 Perform the scheduled maintenance activities while in the reduced inventory mode.

If reduced inventory maintenance requires the installation of SG nozzle dams, the cold leg dams shall be installed first, prior to the hot leg dams and removed last, after hot leg nozzle dam removal.

A4.7 After the completion of the desired maintenance activities, restore RCS level to greater than elevation [119 ft 1in] per the applicable RCS makeup procedure.

A5.0 ABNORMAL OPERATING CONDITIONS

A5.1 Loss of shutdown cooling flow.

NOTE

There are a number of potential initiators that lead to the loss of shutdown cooling flow. The more probable initiators and the immediate actions to restore DHR are as follows

A. Pump failure (e.g., bearing failure, motor failure, shaft breakage)

Actions

1. Verify RCS level greater than minimum RCS level [El. 39.7m (130 ft)]

2. Align the alternate SCS division, if required, for DHR.

3. Start alternate division SCS system pump and verify DHR capability.

4. Align the containment spray pump in the failed division for operation; hold system in standby.

5. Determine cause of SCS pump failure and determine most reliable means (division) of heat decay removal. Realign plant systems, if required, to support DHR operation. If technical specification surveillance requirements / limiting conditions for operation (LCOs) cannot be met, actions should be taken to raise RCS level to greater than El. 119 ft 1 in as soon as possible.

B. SCS flow degradation due to vortexing

1. Secure the operating SCS pump.

2. Restore RCS level using one or more of the systems identified below. The methods of level restoration are specified in the order of preference:

   a. Operable safety injection system

   b. Alternate SCS via in-containment refueling water storage tank (IRWST) (requires manual valve realignment)

   c. Operable containment spray pump

   d. Charging pump alignment to the boric acid storage tank (BAST) or designated alternate borated water source (verify boron concentration and level before use)

   e. Safety injection tanks (verify level before use)

3. Start alternate division SCS system pump and verify DHR capability.

4. Vent (if necessary) and verify containment spray pump operability as backup to SCS pump.

5. Vent failed loop SCS system pump.

6. Determine most reliable means (division) of DHR. Realign plant systems, if required, to support DHR operation. If Technical Specification surveillance requirements/LCOs cannot be met, actions should be taken to raise RCS level to El. greater than [119 ft 1 in] as soon as possible.

C. Inadvertent SCS pump suction isolation valve closure

1. Align, if necessary, and start the alternate SCS division pump to restore DHR.

2. Realign the failed division flow path. If the SCS pump cannot be aligned, align the failed division containment spray pump.

3. Determine most reliable means (division) of DHR. Realign plant systems, if required, to support DHR operation. If Technical Specification surveillance requirements/LCOs cannot be met, actions should be taken to raise RCS level to El. greater than [119 ft 1 in] as soon as possible.

D. Loss of offsite power/station blackout

1. Align, if necessary, and start the alternate division SCS pump if power is available to the alternate pump.

2. If no power is available, restore power immediately.

3. Verify diesel generator operation and align/start the applicable division SCS pump to restore DHR.

4. Start and align the combustion turbine, if available, to the Class 1E bus only if an emergency diesel generator is not available.

5. Determine the most reliable means (division) of DHR. Realign plant systems, if required, to support DHR operation. If Technical Specification surveillance requirements/LCOs cannot be met, actions should be taken to raise RCS level to El. greater than [119 ft 1 in] as soon as possible.

A5.2 Loss of coolant inventory

A. Stop/isolate leak.

B. Secure the operating SCS pump if vortexing is indicated.

NOTE

In the event of DHR interruption due to the loss of forced shutdown cooling flow, alternate methods of DHR (i.e., SGs should be considered if available).

C. Restore RCS inventory as described in Section A5.1.B.2.a through e.

D. Restore SCS flow after inventory recovery and associated venting operations are completed.