January 8, 1992

Docket No. 52-002

Mr. E. H. Kennedy, Manager
Nuclear Systems Licensing
Combustion Engineering, Inc.
1000 Prospect Hill Road
Windsor, Connecticut 06095

Dear Mr. Kennedy:

SUBJECT: DRAFT NUREG-1449, "NRC STAFF EVALUATION OF SHUTDOWN AND LOW POWER OPERATION"

Enclosed is a copy of the draft NUREG-1449 entitled, "NRC Staff Evaluation of Shutdown and Low Power Operation," less predecisional information (i.e., Executive Summary, Potential Industry Actions, and Potential NRC Staff Actions are omitted). You are invited to review this information concerning the applicability of these issues to the CESSAR System 80+. The NRC staff will be available to discuss any concerns that you may have. Please be aware that the final version of NUREG-1449 is scheduled to be issued to the Commission by early February 1992.

Sincerely,
Original Signed By:
Thomas Wambach, Project Manager
Standardization Project Directorate
Division of Advanced Reactors
and Special Projects
Office of Nuclear Reactor Regulation

Enclosure:
Draft NUREG-1449

cc w/o enclosure:
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

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DRAFT NUREG-1449

w/o Chapters 7 & 8

NRC STAFF EVALUATION OF SHUTDOWN and LOW POWER OPERATION

U.S. Nuclear Regulatory Commission

ABSTRACT

The report contains the results of the NRC Staff's evaluation of shutdown and low-power operations at U.S commercial nuclear power plants. The report describes studies conducted by the staff in the following areas: operating experience related to shutdown and low-power operations, probabilistic risk assessment of shutdown and low-power conditions and utility programs for planning and conducting activities during periods the plant is shutdown. The report also documents evaluations of a number of technical issues regarding shutdown and low-power operations performed by the staff, including the principal findings and conclusions. Potential new regulatory requirements are discussed, as well as potential changes in NRC programs.

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ABBREVIATIONS

ac	alternating current
ACRS	Advisory Committee on Reactor Safeguards
AEOD	Office for Analysis and Evaluation of Operational Data
AFW	auxiliary feedwater
AIT	augmented inspection team
ALARA	as low as reasonably achievable
AOP	abnormal operating procedure
ASME	American Society of Mechanical Engineers
ASP	accident sequence precursor
ATWS	anticipated transient without scram
BNL	Brookhaven National Laboratory
B&W	Babcock and Wilcox
BWR	boiling-water reactor
CDF	core-damage frequency
CE	Combustion Engineering
CFR	Code of Federal Regulations
CNRA	Committee for Nuclear Regulatory Activities
CR	control room
CRGR	Committee to Review Generic Requirements
CRD	control rod drive
CRT	cathode-ray tube
CS	core spray
CST	condensate storage tank
EAL	emergency action level
ECC	emergency core cooling
ECCS	emergency core cooling system
EDG	emergency diesel generator
EOP	Emergency Operating Procedures
EPRI	Electric Power Research Institute
ESF	engineered safety features

FSAR	final safety analysis report
GDC	general design criteria
GE	General Electric
GL	generic letter
HPCS	high-pressure core spray
HRA	human reliability analysis
HVAC	heating, ventilation, and air conditioning
IIT	incident investigation team
INEL	Idaho National Engineering Laboratory
INPO	Institute of Nuclear Power Operations
JTA	job task analysis
K/A	knowledge and abilities
LCO	limiting condition for operation
LER	licensee event report
LOCA	loss-of-coolant accident
LOOP	loss of offsite power
LPCI	low-pressure coolant injection
LPS	low-power/shutdown
LPSI	low-pressure safety injection
MC	manual chapter
MPC	maximum permissible concentration
NEA	Nuclear Energy Agency
NPRDS	nuclear plant reliability data system
NRC	Nuclear Regulatory Commission
NRR	Office of Nuclear Reactor Regulation
NSSS	nuclear steam supply system

NUMARC	Nuclear Management and Resources Council
NSAC	Nuclear Safety Analysis Center
OECD	Organization for Economic Cooperation and Development
OGC	Office of the General Counsel
ORNL	Oak Ridge National Laboratory
PORV	power-operated relief valve
PRA	probabilistic risk assessment
PSW	plant service water
PWR	pressurized-water reactor
RCIC	reactor core isolation cooling
RCP	reactor coolant pump
RCS	reactor coolant system
RES	Office of Nuclear Regulatory Research
RHR	residual heat removal
RPV	reactor pressure vessel
RV	reactor vessel
RWCU	reactor water cleanup
RWSP	refueling water storage pool
RWST	refueling water storage tank
SAIC	Science Applications International Corporation
SBO	station blackout
SDC	shutdown cooling
SFP	spent fuel pool
SG	steam generator
SI	safety injection
SRO	senior reactor operator
SRP	Standard Review Plan (NUREG-0800)
SRV	safety-relief valve
STS	standard technical specifications

TAF	top of active fuel
T/C	thermocouple
TI	temporary instruction
TS	technical specification
VCT	volume control tank
<u>W</u>	Westinghouse

1 BACKGROUND AND INTRODUCTION

Over the past several years, the NRC staff has become more concerned about the safety of operations during shutdown. The Diablo Canyon event of April 10, 1987, highlighted the fact that the operation of a pressurized-water reactor (PWR) with a reduced inventory in the reactor coolant system was a particularly sensitive condition. From NRC's review of the event, the staff issued Generic Letter 88-17 on October 17, 1988. The letter requested that licensees address numerous generic deficiencies to improve safety during operation at reduced inventory. More recently, the incident investigation team's report of the loss of ac power at the Vogtle plant (NUREG-1410) emphasized the need for risk management of shutdown operations. Furthermore, discussions with foreign regulatory organizations (i.e., French and Swedish authorities) about their evaluations regarding shutdown risk have confirmed previous NRC staff findings that the core-damage frequency for shutdown operation can be a fairly substantial fraction of the total core-damage frequency. Because of these concerns regarding operational safety during shutdown, the staff began a careful, detailed evaluation of safety during shutdown and low-power operations.

On July 12, 1990, the staff briefed the Advisory Committee on Reactor Safeguards (ACRS) on its draft plan for a broad evaluation of risks during shutdown and low-power operation. On October 22, 1990, the staff issued the plan in the form of a memorandum from James M. Taylor, to the Commissioners, "Staff Plan for Evaluating Safety Risks During Shutdown and Low Power Operations." The staff briefed the ACRS on the status of the evaluation on June 5 and 6, 1991, and on June 19, 1991, the staff discussed the status of the evaluation in a public meeting with the Commission. On September 9, 1991, the staff issued a Commission paper (SECY-91-283) which reported progress to date on the evaluation and provided a detailed plan for addressing each of the technical issues identified.

1.1 Scope of the Staff Evaluation

In the staff's evaluation, "shutdown and low-power operation" encompasses operation when the reactor is in a subcritical state or is in transition between subcriticality and power operation up to 5 percent of rated power. The evaluation addresses only conditions for which there is fuel in the reactor vessel (RV). The evaluation addresses all aspects of the nuclear steam supply system (NSSS), the containment, and all systems that support operation of the NSSS and containment. However, the evaluation does not address events involving fuel handling outside of the containment, fuel storage in the fuel storage building, and events not involving the previously identified systems.

1.2 Organization

The Office of Nuclear Reactor Regulation (NRR) has the lead responsibility for conducting the evaluation. However, other Headquarters offices, such as the Office of Nuclear Regulatory Research (RES), the Office for Analysis and Evaluation of Operational Data (AEOD), and regional offices have contributed strong support. A group of senior managers representing these offices serve as the steering committee for the evaluation. This group met periodically to be briefed on the progress of the evaluation and to provide guidance. Members of the steering committee included William Russell, Associate Director for Inspection and Technical Assessment, NRR; Ashok Thadani, Director, Division of Systems Technology, NRR; Brian Sheron, Director, Division of Systems Research, RES (later replaced by Warren Minners, Director, Division of Safety Issue Resolution); Samuel Collins, Director, Division of Reactor Projects, Region IV; and Thomas Novak, Director, Division of Safety Programs, AEOD.

1.3 Summary of the Evaluation

In its original plan, the staff divided work necessary to complete the evaluation into the following six major elements containing a number of interrelated tasks to be completed over 18 months. The major program elements included:

I. Review and evaluate event experience and event studies.
II. Study shutdown operations and activities.
III. Conduct probabilistic risk assessment (PRA) activities and engineering studies.

IV. Integrate technical results to understand risk.

V. Evaluate guidance and requirements affecting risk management.

VI. Recommend new regulatory requirements as necessary.

Consistent with this program plan, the staff and its contractors have completed the following studies which, as indicated, are fully discussed later in this report:

° Systematically reviewed operating experience, including reviewing reports of events at foreign and domestic operating reactors, and documented the findings in the AEOD engineering evaluation (Chapter 2);

° With assistance from the Science Applications International Corporation (SAIC), analyzed a spectrum of events at operating reactors using the accident sequence precursor (ASP) methodology (Chapter 2);

° Visited 11 plant sites to broaden staff understanding of shutdown operations, including outage planning, outage management, and startup and shutdown activities (Chapter 3);

° Reviewed, evaluated, and documented the few existing domestic and foreign PRAs that address shutdown conditions (Chapter 4);

° Completed and documented a coarse Level 1 PRA of shutdown and low-power operating modes for a PWR and a BWR through RES contractors at Brookhaven National Laboratory and Sandia National Laboratory (Chapter 4);

° With technical assistance from the Idaho National Engineering Laboratory (INEL), completed and documented several thermal-hydraulic studies that address the consequences of an extended loss of residual heat removal (Chpater 6);

° With technical assistance from Science Applications International Corporation (SAIC), compiled existing regulatory requirements for shutdown operation and important safety-related equipment (Chapter 5);

- Coordinated a meeting with OECD/NEA (Organization for Economic Cooperation and Development/Nuclear Energy Agency) specialists to exchange information on current regulatory approaches to the shutdown issues in member countries, including drafting a discussion paper on the various approaches (Chapter 5); and

- Met periodically with the Nuclear Management and Resources Council (NUMARC) to keep the industry informed of NRC activities and to stay abreast of the industry's continuing initiatives.

To integrate its findings from these studies and to define important technical issues, the staff met for three days with contractors from several national laboratories who had been working on the shutdown and low-power evaluation or had special expertise in the issue. During this meeting, held April 30 through May 2, 1991, the staff identified five issues that are especially important for shutdow ,nd a number of additional topics that warrant further evaluation. These issues are

- outage planning and control
- stress on personnel and programs
- training and procedures
- technical specifications
- PWR safety during midloop operation

Topics identified for further evaluation included:

- loss of residual heat removal capability
- containment capability
- rapid boron dilution
- fire protection
- instrumentation
- ECCS recirculation capability
- effect of PWR upper internals
- onsite emergency planning
- fuel handling and heavy loads

- potential for draining the BWR reactor vessel
- reporting requirements for shutdown events
- need to strengthen inspection program

The staff proposed an evaluation plan for each of the issues and topics and documented the plans in a Commission paper issued September 9, 1991 (SECY-91-283). The evaluations have now been complete and the results form the basis for the staff's technical findings and conclusions provided in Chapter 6, and recommended actions as provided in Chapters 6, 7, and 8 of this report.

2 ASSESSMENT OF OPERATING EXPERIENCE

2.1 Retrospective Review of Events at Operating Reactors

The staff reviewed operating experience reports to ensure that its evaluation encompassed the range of events encountered during shutdown and low-power operation. The staff reviewed licensee event reports (LERs), previous studies performed by the Office for Analysis and Evaluation of Operational Data (AEOD), and various inspection reports to determine the types of events that take place during refueling, cold and hot shutdown, and low-power operation.

The staff reviewed events that occurred at foreign nuclear power plants using information found in the foreign events file maintained for AEOD at the Oak Ridge National Laboratory. The AEOD compilation included the types of events that applied to U.S. nuclear plants and those not found in a review of U.S. experience.

In performing this review, the staff found that the more significant events for pressurized-water reactors (PWRs) were the loss of residual heat removal, potential pressurization, and boron dilution events. The more important events for boiling-water reactors (BWRs) were (1) the loss of coolant, (2) the loss of cooling, and (3) potential pressurization. Generally, the majority of important events involved human error--administrative, other personnel, and procedural errors. In December 1990, the staff documented this review in the AEOD special report, "Review of Operating Events Occurring During Hot and Cold Shutdown and Refueling" which is summarized below. In addition, the staff selected 10 events from the AEOD review for further assessment as precursors to potential severe core-damage accidents. This assessment is discussed in Section 2.2.

The AEOD special report encompassed events that had occurred primarily between January 1, 1988, and July 1, 1990. An initial database was created which included 348 events gathered primarily from the Sequence Coding and Search System (SCSS) and significant events that occurred before or after the target period. Of the 348 events, approximately 30 percent were considered more significant and were explicitly discussed in the AEOD report.

The events were evaluated by plant type (i.e., PWR or BWR) and six major event categories: loss of shutdown cooling, loss of electrical power, containment integrity problems, loss of reactor coolant, flooding and spills, and overpressurization of the reactor coolant system; for PWRs, boron problems were also included. Less frequently occurring events, such as fires, were covered briefly.

The results of the AEOD study are discussed in Sections 2.1.1 through 2.1.7. Insights from the study are given in Section 2.1.8.

2.1.1 Loss of Shutdown Cooling

The loss of shutdown cooling is one of the more serious event types and can be initiated by the loss of flow in the residual heat removal (RHR) system or by loss of an intermediate or ultimate heat sink. Events involving loss of cooling that occur shortly after plant shutdown will quickly lead to bulk boiling and eventual fuel uncovery if cooling is not restored.

The evaluation included 13 PWR and 11 BWR events involving loss of shutdown cooling; these are listed in Tables 2.1 and 2.2.

More than 60 percent of the PWR events arose from human error--administrative, other personnel, or procedural. Equipment problems accounted for 16 percent of the events. The types of incidents that caused the events ranged from the RHR pump becoming air bound, to loss of power to the RHR pump, to the malfunction of the level indication in the control room.

These events resulted in temperature rises ranging from 15° to 190° (on the Fahrenheit scale).

For the BWR events, approximately 60 percent were caused by human error--administrative, other personnel, or procedural.

2.1.2 Loss of Reactor Coolant Inventory

The chance that reactor coolant will be lost from the reactor vessel can actually increase during shutdown modes because large, low-pressure systems, such as RHR, are connected to the reactor coolant system. The safety significance of such loss is that it could lead to voiding in the core and eventual exposure of the core.

The evaluation included 22 events involving loss of reactor coolant. The plants and dates of the events are listed in Tables 2.3 and 2.4:

The PWR events had various causes, such as opening of the RHR pump suction relief valve power-operated relief valve (PORV) and block valves opening simultaneously during PORV testing, and loss of pressure in the reactor cavity seal ring allowing drainage from the cavity. These events accounted for losses of reactor coolant inventory of up to 67,000 gallons.

Many of the BWR events included in the evaluation were caused by valve lineup errors and resulted in decreased levels of up to 72 inches.

Of the 10 PWR events reported in the AEOD evaluation, 6 were caused by human errors and 4 were caused by equipment problems. Of the 12 BWR events included in the evaluation, 10 were caused by human errors and only 2 were caused by equipment failure.

2.1.3 Breach of Containment Integrity

A breach of containment integrity in itself may not be of great safety significance, but this event, coupled with other postulated events, could substantially increase the severity of the initial event. Also, a breach of containment integrity in conjunction with fuel failure could cause the release of radioactive material. Eight events involving breach of containment were included in the AEOD evaluation. All were due to human error.

2.1.4 Loss of Electrical Power

The safety significance of the loss of electrical power depends on the part of the plant affected. The loss could range from complete loss of offsite power to the loss of a dc or an instrument bus; loss of electrical power generally leads to other events, such as loss of shutdown cooling.

The events included in th[illegible] OD evaluation are listed in Table 2.5.

Of the 11 PWR events [illegible] AEOD, 5 were caused by human errors, 5 were caused by maintenance, a[illegible]re. Of the original 45 events found in the AEOD study, appro[illegible] were caused by human error and approximately 20 percent wer[illegible]ment problems. The BWR statistics were just the opposite: only 20 percent [illegible]he events were caused by human errors and 50 percent were caused by equipment problems.

2.1.5 Overpressurization of Reactor Coolant System

Both PWR and BWR overpressurization events have occurred during shutdown conditions. Such events are precursors to exceeding the reactor vessel brittle fracture limits or the American Society of Mechanical Engineers Boiler and Pressure Vessel Code (ASME Code) limits. The reactor coolant system (RCS) generally overpressurizes in one of three ways: operation with the RCS completely full and experiencing pressure control problems, occurrences of inadvertent safety injection, or pressurization of systems attached to the RCS.

Too few of these events were included in the AEOD evaluation to indicate a trend regarding the cause of the events. However, the original database included 24 PWR pressurization events, and 66 percent of those events had been caused by human errors. Only three BWR events were in the original database.

2.1.6 Flooding and Spills

The safety significance of flooding or spills depends on the equipment affected by the spills. The AEOD evaluation included 3 of the 29 PWR events in the

Table 2.1

Events Involving PWR Loss of Shutdown Cooling

Plant	Event Date
Millstone 2	12/09/81
Salem 1	03/16/82
Catawba 1	04/22/85
Zion 2	12/14/85
Crystal River 3	02/02/86
Oconee 3	12/16/87
Arkansas 1	10/26/88
McGuire 1	11/23/88
Arkansas 1	12/19/88
Braidwood 2	01/23/89
Salem 1	05/20/89
Arkansas 1	12/06/89
Votgle 1	03/20/90

Table 2.2

Events Involving BWR Loss of Shutdown Cooling

Plant	Event Date
Brunswick 1&2	04/17/81
Susquehanna 1	03/21/84
Fermi 2	03/18/88
FitzPatrick	10/21/88
Susquehanna 1	01/07/89
River Bend	06/13/89
Pilgrim	12/09/89
Duane Arnold	01/09/90
FitzPatrick	01/20/90
Susequehanna 1	02/03/90

Table 2.3

Events Involving PWR Loss of Reactor Coolant

Plant	Event Date
Haddam Neck	08/21/84
Farley 2	10/27/87
Surry 1	05/17/88
Sequoyah 1	05/23/88
San Onofre 2	06/22/88
Byron 1	09/19/88
Cook 2	02/16/89
Indian Point 2	03/25/89
Palisades	11/21/89
Braidwood 1	12/01/89

Table 2.4

Events Involving BWR Loss of Reactor Coolant

Plant	Event Date
Grand Gulf	04/03/83
LaSalle 1	09/14/83
LaSalle 2	03/08/84
Washington Nuc 2	08/23/84
Susquehanna 2	04/27/85
Hatch 2	05/10/85
Peach Bottom 2	09/24/85
Fermi 2	03/13/87
Washington Nuc 2	05/01/88
Pilgrim	12/03/88
Vermont Yankee	03/09/89
Limerick	04/07/89

Table 2.5

Events Involving Loss of Electrical Power

PWR	Event Date	Description of Event
Turkey Point 3	05/77/85	Loss of offsite power
Fort Calhoun	03/21/87	Loss of all ac offsite power
McGuire 1	09/16/87	Loss of offsite power
Harris	10/11/87	Loss of power to safety buses
Wolf Creek	10/15/87	Loss of 125-volt dc source
Crystal River 3	10/16/87	Loss of power to one of two vital buses
Indian Point 2	11/05/87	Loss of power to the 480-V ac bus
Braidwood 2	01/31/88	Instrument bus deenergized
Millstone 2	02/04/88	Loss of power to vital 4160-V ac train
Yankee Rowe	11/16/88	Loss of power to two emergency 480-V buses
Fort Calhoun	02/26/90	Loss of power to 4160 safety buses
BWR		
Pilgrim	11/12/87	Loss of offsite power
Nine Mile 2	12/26/88	Loss of offsite power
Millstone 1	04/29/89	Loss of normal power
Washington Nuc 2	05/14/89	Loss of offsite power
River Bend	03/25/89	Division II loss of power
Limerick	03/30/90	Loss of power to power supply uninterruptedly

original database. Of the original 29 PWR events, more than 50 percent were caused by human errors; 14 percent were caused by equipment problems. There were only 7 BWR flooding or spill events in the original database and the majority were caused by human errors.

2.1.7 Inadvertent Reactivity Addition

Both PWR and BWR plants had experienced inadvertent criticalities, some of which resulted in reactor scrams. The AEOD evaluation indicated that inadvertent reactivity addition in PWRs was caused primarily by dilution while the plant was shut down. Also boron dilution without the operator's knowledge was identified as a potentially severe event. In BWRs, inadvertent reactivity addition was most often caused by human error (the operator selected the wrong control) and feedwater transients.

The events included in the evaluation are listed in Table 2.6.

2.1.8 Insights From the Review of Events

The original database of shutdown events included events and a majority of the events had occurred since 1985. AEOD used experience and engineering judgment in selecting which of the 348 events were the more significant. Those significant events were then categorized to help AEOD determining the cause and idertify any trending.

Two major observations became apparent in the evaluation whether using the original database of 348 or the narrowed database of 30 more significant events. The first observation is that a greater percentage of the events were caused by human errors than by equipment problems. The second observation is that the events did not reveal new unanalyzed issues but instead appeared to represent an accumulation of errors or equipment failures or a combination of the two.

Table 2.6

Events Involving Inadvertent Reactivity Addition

PWR	Event Date	Description of Event
Surry 2	04/14-23/89	Boron concentration decreased by leak in RCP standpipe makeup valve
Turkey Point 3&4	05/28-6/03/87	Unable to borate Unit 3 VCT* because of nitrogen gas binding of all boric acid transfer pumps
Arkansas 2	05/04/88	Gas binding of the charging pumps from inadvertent emptying of the VCT*
Foreign reactor	1990	Boron dilution from a cut steam generator tube that had not been plugged
BWR		
Millstone 1	11/12/76	Withdrawal of the wrong control rod and a suspected high worth rod
Browns Ferry 2	02/22/84	Withdrawal of high worth rod
Hatch 2	11/7/85	Feedwater transient
Peach Bottom 3	03/18/86	Incorrect rod withdrawn
River Bend	07/14/86	Feedwater transient
Oyster Creek	12/24/86	Feedwater transient

* VCT = volume control tank

2.2 Accident Sequence Precursor Analysis

Using the accident sequence precursor (ASP) method, the staff and its contractors, Oak Ridge National Laboratory and Science Applications International Corporation (ORNL/SAIC), evaluated a sample of 10 shutdown events that could be significant. The staff reviewed this sample to determine the conditional probability of core damage, that is, the probability of core damage given that the initiating event has already occurred, from each type of event selected in order to help characterize the overall shutdown risk for U.S. nuclear power plants. As discussed in Section 2.2.1, the 10 selected events reasonably represented the reactor population of BWRs, PWRs, and the various vendors.

To date, the ASP program has been largely concerned with operational events that occurred at power or hot shutdown. Methods used in that program to identify operational events considered precursors, plus the models used to estimate risk significance, have been developed over a number of years. In particular, the ASP core-damage models have been improved over time to reflect insights from a variety of probabilistic risk assessment (PRA) studies. In applying ASP methods to evaluate events during cold shutdown and refueling, the same analytical approach was used. However, accident sequence models describing failure combinations leading to core damage had to be developed, with little earlier work as a basis.

This analysis was exploratory in nature. Its intent was to ensure that operating experience was assessed systematically, to develop insights into (1) the types of events that have occurred during shutdown and (2) which characteristics of these events are important to risk, and to develop methods that could be used in a continuing manner to analyze shutdown events. The staff did not intend to use this effort to make comparisons with analyses of at-power events in the ASP program.

The following section describes how the 10 events that were analyzed were selected. Section 2.2.2 summarizes the development of core-damage models and the estimation of conditional probabilities suitable for event ranking. Finally, Section 2.2.3 describes the results of the analyses and overall findings. The complete detailed analysis for each event is documented in Appendix A.

2.2.1 Selecting Events for Analysis

The staff selected 10 events that had occurred during cold shutdown and refueling for analysis. The staff chose these events after it had (1) reviewed the AEOD evaluation of non-power events discussed in Section 2.1 and (2) performed confirmatory searches using the Sequence Coding and Search System, a database of licensee event report (LER) information maintained at ORNL.

Events chosen were considered representative of the types of events that could impact shutdown risk and that could be analyzed using ASP methods. These events concerned loss of reactor inventory, loss of residual heat removal, and loss of electric power. One event involved a flood that had safety system impacts. The events chosen for analysis were considered more serious than the typical event observed at cold shutdown.

Events were also chosen so that all four reactor vendors were represented in the analysis. This allowed the staff to explore modeling issues unique to different plant designs and to develop models that could be applied at a later date to a broad set of cold-shutdown and refueling events.

The 10 events chosen for analysis are listed in Table 2.7. The 10 events are sorted by date and by vendor in Table 2.8. The 1990 loss of ac power and shutdown cooling (SDC) at Vogtle 1 is not included in the list because it had been evaluated previously with the ASP methodology.

2.2.2 Analysis Approach

The staff analyzed in detail each of the events listed in Table 2.7. This analysis included a review of available information concerning each event and plant to determine system lineups, equipment out of service, water levels and reactor pressure vessel (RPV) inventories, time to boil and to core uncovery,

Table 2.7

Cold-Shutdown and Refueling Events Analyzed Using ASP Methods, by Docket/LER No.

Docket/ LER No.	Description of Event (Date)	Conditional Core-Damage Probability*
271/89-013	10,000 gal of reactor vessel inventory was transferred to the torus at Vermont Yankee when maintenance stroked-tested the SDC valves in the out-of-service loop of RHR with the minimum flow valve already open. More than 45 min required to locate and isolate the leak. (3/9/89)	1.0×10^{-6}
285/90-006	Loss of offsite power (LOOP) with the emergency diesel generators (EDGs) not immediately available at Fort Calhoun. Breaker failer relay operated to strip loads, but EDG design feature prevented auto loading. (2/26/90)	3.6×10^{-6}
287/88-005	Loss of ac power and loss of RHR during midloop operation with vessel head on at Oconee 3. Testing errors caused a loss of power to feeder buses resulting in loss of SDC with no accompanying reactor temperature or level indication. (9/11/88)	1.7×10^{-6}
302/86-003	RHR pump shaft broke during midloop operation at Crystal River 3. Pump had been in continuous operation for about 30 days. A tripped circuit breaker delayed placing the second train on-line. (2/2/86)	1.4×10^{-6}

Table 2.7 (Continued)

Docket/ LER No.	Description of Event (date)	Conditional Core-Damage Probability*
323/87-005	Loss of RHR at Diablo Canyon 2 while at midloop operation. RCS inventory was lost through a leaking valve and air entrainment in both RHR pumps caused loss of SDC. Boiling in core region. (4/10/87)	5.2×10^{-5}
382/86-015	Loss of RHR during midloop operation at Waterford 3. Complications in restoring RHR due to steam binding and RHR pump suction line design. Boiling in core region. (7/14/86)	2.1×10^{-4}
387/90-005	Extended loss of RHR at Susquehanna 1. An electrical fault caused isolation of SDC suction supply to RHR system. Alternate residual heat removal was provided using the suppression pool. (2/3/90)	2.7×10^{-5}
397/88-011	Loss of reactor vessel inventory at WNP-2. The RHR suppression pool suction and SDC suction valves were open simultaneously, and approximately 10,000 gal of reactor water was transferred to the suppression pool. (5/1/88)	4.6×10^{-5}
456/89-016	RCS inventory loss at Braidwood 1. An RHR suction relief valve stuck open and drained approximately 64,000 gal of water from the RCS before being isolated. (12/1/89)	1.0×10^{-6}

Table 2.7 (Continued)

Docket/ LER No.	Description of event (date)	Conditional Core-Damage Probability*
458/89-020	15,000 gal of service water flooded the auxiliary building when a freeze seal failed at River Bend. One RHR train, normal spent fuel pool cooling, and auxiliary and reactor building lighting were lost. (4/19/89)	1.0×10^{-6}

*The estimated conditional probabilities are only considered usable for the relative ranking of the 10 events. The probability values cannot be compared with conditional probabilities calculated in the ASP program for events occurring at power.

Table 2.8

Cold Shutdown and Refueling Events Analyzed Using ASP Methods, by Vendor

Docket/ LER No.	Description of Event (Date)	Conditional Core-Damage Probability*
GENERAL ELECTRIC (BWR)		
271/89-013	10,000 gal of reactor vessel inventory was transferred to the torus at Vermont Yankee. (3/9/89)	1.0×10^{-6}
387/90-005	Extended loss of RHR at Susquehanna 1. (2/3/90)	2.7×10^{-5}
397/88-011	Loss of reactor vessel inventory at WNP-2. (5/1/88)	4.6×10^{-5}
458/89-020	15,000 gal of service water flooded the auxiliary building when a freeze seal failed at River Bend. (4/19/89)	1.0×10^{-6}
BABCOCK AND WILCOX (PWR)		
287/88-005	Loss of ac power and loss of RHR during midloop operation with vessel head on at Oconee 3. (9/11/88)	1.7×10^{-6}
302/86-003	RHR pump shaft broke during midloop operation at Crystal River 3. (2/2/86)	1.4×10^{-6}

Table 2.8 (Continued)

Docket/ LER No.	Description of Event (Date)	Conditional Core-Damage Probability*
COMBUSTION ENGINEERING (PWR)		
285/90-006	Loss of offsite power (LOOP) with the emergency diesel generators (EDGs) not immediately available at Fort Calhoun. (2/26/90)	$3.6x10^{-6}$
382/86-015	Loss of RHR during midloop operation at Waterford 3. (7/14/86)	$2.1x10^{-4}$
WESTINGHOUSE (PWR)		
323/87-005	Loss of RHR at Diablo Canyon 2 while in midloop operation. (4/10/87)	$5.2x10^{-5}$
456/89-016	RCS inventory loss at Braidwood 1. (12/1/89)	$1.0x10^{-6}$

*The estimated conditional probabilities are only considered usable for the relative ranking of the 10 events. The probability values cannot be compared with conditional probabilities calculated in the ASP program for events occurring at power.

vessel status, and so on. This involved review of final safety analysis reports (FSARs), augmented inspection team (AIT) reports, operating procedures, and supplemental material in order to understand the system interactions that occurred during the event, the recovery actions and alternate strategies that could be employed, and the procedures available to the operators.

Once the event had been characterized and its effect on the plant was understood, event significance was estimated based on methods used in the ASP program. Quantification of event significance involves determining a conditional probability of subsequent core damage given the failures that occurred. (See Sectirn 2.2.3 for the current limitations in this approach.) This was estimated by mapping failures observed during the event onto event trees that depict potential paths to severe core damage, and by calculating a conditional probability of core damage through the use of event tree branch probabilities modified to reflect the event. The effect of an event on event tree branches was assessed by reviewing the operational event specifics against system design information and translating the results of the review into a revised conditional probability of branch failure given the operational event.

In the quantification process, it was assumed that the failure probabilities for systems observed to have failed during an event were equal to the likelihood of not recovering from the failure or fault that actually occurred. Failure probabilities for systems observed to have degraded during an operational event were assumed equal to the conditional probability that the system would fail (given that it was observed degraded) and the probability that it would not be recovered within the required time period. The failure probabilities associated with observed successes and with systems unchallenged during the actual event were assumed equal to a failure probability estimated by the use of system success criteria and train and common-mode failure screening probabilities, with consideration of the potential for recovery. The conditional probability estimated for each event was useful in ranking because it provides an estimate of the measure of protection remaining against core damage once the observed failures have taken place.

Event tree models were developed to describe potential core-damage sequences associated with each event. For the purposes of this analysis, core damage was assumed to occur when RPV water level decreased to below the top of active fuel (TAF). Choice of this damage criterion allowed the use of simplified calculations to estimate the time to an unacceptable end state. Core damage was also assumed to occur if a combination of systems, as specified on the event tree, failed to perform at a minimum acceptable level and could not be recovered.

The event tree model used to analyze an event was developed on the basis of procedures that existed then. These procedures were considered the primary source of information available to the operators concerning the steps to be taken to recover from the event or to implement another strategy for cooling the core. Since procedures varied greatly among plants, the event trees developed to quantify an event were typically plant and event specific. Event trees applicable to each analysis are described in Appendix A.

In developing branch probability estimates for the cold-shutdown models, the probability of not recovering a faulted branch before boiling or core uncovery occured frequently had to be estimated. Applicable time periods were often 6 to 24 hours.

There are no operator response models (especially models out of the control room) and equipment repair models for these time periods. For the purposes of this analysis, the probability of crew failure as a function of time for non-proceduralized actions was developed by skewing applicable curves for knowledge-based action in the control room by 20 minutes to account for recovery time outside the control room. A minimum (truncated) failure probability of $1x10^{-4}$ was also specified. For long-term proceduralized actions, recovery was assumed to be dominated by equipment failure, and operator failure was not addressed. The probability of failing to repair a faulted system before boiling or core uncovery occurred was estimated using an exponential repair model with the observed repair time as the median.

Probability values estimated using these approaches are very uncertain.

Unfortunately, these same probabilities significantly influence the conditional core-damage probabilities estimated for the four more significant events and, therefore, those conditional probabilities are also uncertain.

The impact of long-term recovery assumptions is illustrated below. Changes in conditional probabilities resulting from a factor-of-three change in the non-recovery estimates are listed for the Susquehanna and Waterford events. As can be seen, within the range shown, the conditional probability for both events was very strongly related to assumptions concerning long-term recovery.

Operator response is probably the most important issue determining the significance of an event in shutdown, and until it is better understood, the relative importance of shutdown events compared to events at power cannot be reliably estimated.

2.2.3 Results and Findings

The conditional core-damage probabilities estimated for each event are listed in Table 2.7 and shown in Figure 2-1. The calculated probabilities are strongly influenced by estimates of the likelihood of failing to recover initially faulted systems over time periods of 6 to 24 hours. Very little information exists concerning such actions; hence, the conditional probability estimated for an event involved substantial uncertainty. Additionally, some conditional probabilities were strongly influenced by assumptions concerning (1) the plant staff's ability to implement non-proceduralized short-term actions, (2) the actual plant status at the time of the event, and (3) the potential for the event to have occurred under different plant conditions. Because of these factors, the estimated conditional probabilities are only considered usable for the relative ranking of the 10 events. The probability values cannot be compared with conditional probabilities calculated in the ASP program for events occurring at power.

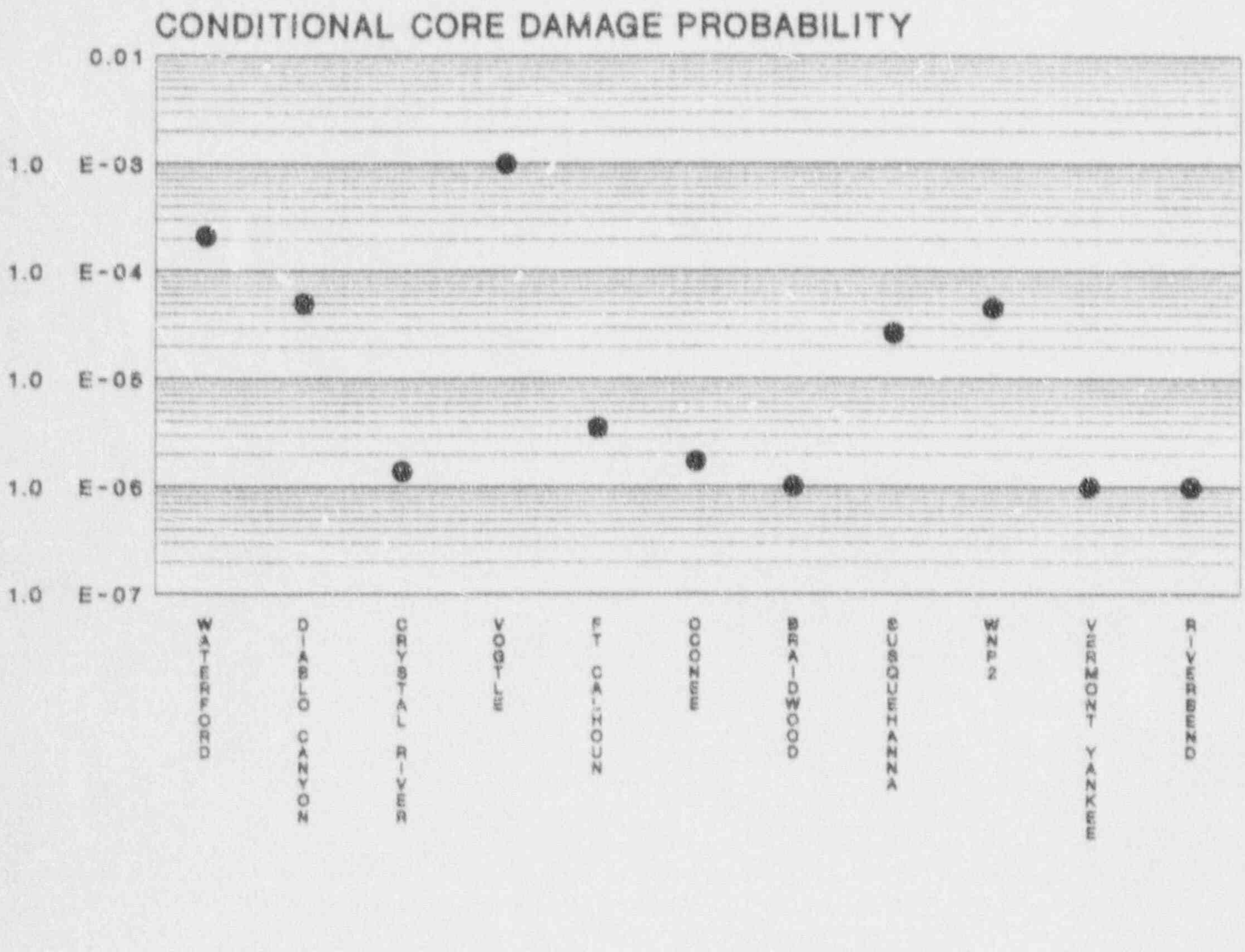
The distribution of events as a function of conditional probability is shown in Table 2.9. The result for the 1990 loss of ac power and SDC at Vogtle 1 is also included for completeness. The analysis performed for the Vogtle 1 event is also provided in Appendix A. Events with conditional probabilities below 1×10^{-5} are considered minor with respect to risk of core damage. Conditional probabilities above this value are indicative of a more serious event. Because of the uncertainties inherent in the analysis process, it is not possible to distinguish the relative significance of events with similar conditional probabilities (factor of three).

Table 2.9

Events Ranked by Order of Magnitude

Conditional probability range	Events ranked by conditional probability of subsequent core damage
10^{-3}	Loss of all AC power at Vogtle (See Appendix A)
10^{-4} to 10^{-3}	Loss of RCS inventory and SDC during mid-loop operation at Waterford (LER 382/86-015)
10^{-5} to 10^{-4}	Loss of RCS inventory and SDC during mid-loop operation at Diablo Canyon 2 (LER 323/87-005)
	RHR isolation of Susquehanna 1 (LER 387/90-005)
	Loss of RPV inventory at Washington Nuclear Power 2 (LER 397/88-011)
10^{-6} to 10^{-5}	3 events
10^{-6}	3 events

ACCIDENT SEQUENCE PRECURSOR STUDIES
FIGURE 2-1



The four events with conditional probabilities above 10^{-5} are

(1) Loss of Residual Heat Removal (RHR) During Mid-loop Operation at Waterford 3 on July 14, 1986. In this event, a non-procedu alized drain path was not isolated once the reactor coolant system (RCS) level was reduced to midloop. Draining continued and resulted in cavitation of the operating RHR pump. Restoration of shutdown cooling (SDC) took 3 hours, during which boiling occurred in the core region. Both RHR pump suction lines from the RCS were steam bound (most likely a result of the suction loop seal design at the plant). RCS inventory was restored using one of the low-pressure safety injection (LPSI) pumps (these are the same as the RHR pumps on this plant) taking suction from the refueling water storage pool (RWSP).

Shutdown cooling was eventually restored by using the pump warmup lines in conjunction with repeated pump jogging--a non-proceduralized action. The method specified in the procedure to restore RHR pump suction (use a vacuum priming system to evacuate the loop seal) would not have been effective since hot-leg temperature exceeded 212°F.

The dominant core-damage sequence for this event (which includes the observed failures plus additional postulated failures, beyond the operational event, required for core damage) involved failure to recover RHR and unavailability of the steam generators as an alternative means of removing decay heat.

(2) Loss of RHR During Midloop Operation at Diablo Canyon 2 on April 10, 1987. RCS inventory was lost through a leaking valve. Subsequent air entrainment in the RHR pump suction flow resulted in a loss of SDC. Boiling occurred in the core region. RCS inventory was recovered using gravity feed from the refueling water storage tank (RWST) and an RHR pump was restarted. Use of gravity feed from the RWST to recover RCS inventory was not proceduralized, but had been used earlier in the outage.

The dominant core-damage sequence involved failure to provide continued RCS makeup with the RCS open.

(3) Loss of RPV Inventory at Washington Nuclear Plant 2 (WNP-2) on May 1, 1988. The RHR suppression pool suction and shutdown cooling (SDC) suction valves were incorrectly cycled so that both valves were open at the same time. Approximately 10,000 gallons of water drained from the RPV to the suppression pool in less than a minute, and the SDC suction line isolated on low RPV level. The control rod drive (CRD) and condensate systems were used to recover RPV level. SDC was reestablished in 7 minutes.

The dominant core-damage sequence involved failure to restore RHR after the suction line was isolated and failure to implement substitute long-term SDC strategies (primarily suppression pool cooling).

(4) Loss of RHR at Susquehanna 1 on February 3, 1990. RPS bus B was lost during tests on the RPS bus breaker as a result of a short to ground in a dc distribution panel. The loss of RPS bus B prevented recovery of RHR, which had been previously isolated for the breaker testing, for more than 5 hours. During this time, another form of RHR was used, opening safety-relief valves (SRVs) and dumping steam to the suppression pool. CRD flow was used for RPV makeup.

The dominant core-damage sequence was similar to that for WNP-2--failure to recover RHR and failure to implement substitute long-term SDC strategics.

The factors that resulted in the higher conditional probability estimates for these events highlighted major issues impacting both risk and risk estimates for the shutdown events analyzed in this study: operator actions to recover failed equipment (Waterford), implementation of non-proceduralized recovery actions (Waterford and Diablo Canyon 2), and the potential for restoring failed systems in the 6- to 24-hour timeframe (Susquehanna and WNP-2).

These factors and other analysis findings are discussed below.

Design and Operational Issues Important to Risk During Shutdowns

Plant Procedures. Procedures in use at the time of the event had a significant effect on the analysis of the event, since what operators knew about alternative recovery strategies was assumed to derive primarily from the procedures. Ad hoc actions were postulated in some cases, but were considered much less reliable than proceduralized actions. Detailed guidance was limited in early procedures, and what did exist offered little information on how to recognize an event or implement a correct recovery course. Some procedures did direct operators to substitute systems if RHR could not be recovered, but information needed for determining when such systems would be effective (such as the minimum time after shutdown before the system could adequately remove decay heat) was not given.

Contemporary procedures offer much greater guidance and flexibility, both in the number of substitute systems that can provide residual heat removal and in information to help characterize an event. For example, Crystal River 3 now has a procedure specifically directing the operators to use five different systems for makeup water, whereas in 1986 (when the event analyzed in this study occurred), the procedures listed only two such systems. The current loss-of-RHR procedure for Braidwood lists seven other methods to reestablish core cooling, gives tabular guidance regarding which methods are effective for different operating states, and provides graphs as a function of time since shutdown for RCS heatup, required vent paths to prevent RCS pressurization, and required makeup flow for residual heat removal.

If events similar to those that were analyzed in this report occurred now, many would be considered less significant from the standpoint of risk of core damage because of the additional guidance and flexibility now included in the procedures.

Operator Recovery Actions. Differences between operator actions associated with recognizing that an event was in progress, deterting the cause of a problem, and implementing recovery actions are apparent in the descriptions of many of the 10 events. Several events were taking place for some time before someone either recognized there was a problem or was able to identify the exact nature of the problem. For example, during the Vermont Yankee event, operators took 15 minutes to recognize that the water level in the reactor vessel was decreasing and then they spent the next 30 minutes determining the source of the leak. Once it was found, the source of the leak was quickly isolated.

For the event at Braidwood, operators quickly concluded that an RHR suction relief valve had lifted. However, 2-½ hours were required to locate the valve that had lifted (it was on a non-operating train).

For both the Vermont Yankee and Braidwood events, SDC was not lost and a large amount of time was available to detect and correct the problem before core cooling would have been affected. This was important, because it gave the operators time to deliberately and systematically address each event. Availability of a long time period before the onset of boiling or core uncovery was reflected in lower probabilities for failure to recover a faulted system or implement actions away from the control room.

On the other hand, in the Waterford event (which happened when SDC was lost during midloop operation), boiling in the core region occurred approximately 45 minutes after SDC was lost. This is a short period of time to reliably implement recovery actions out of the control room. For the loss of SDC at Waterford, information concerning RHR pump restart (use of the vacuum priming system to evacuate the suction lines) was not correct for the RCS condition (temperature 212°F) that existed during the event. SDC was eventually restored by repeated pump jogging and the use of pump warmup lines to return some flow to the pump suctions.

<u>Design Features That Complicate Recovery of RHR</u>. The loss of SDC at Waterford illustrates a design feature that significantly affected recovery of SDC. At Waterford, loop seals exist in both the RHR suction and discharge lines. The loop seals are more elevated than the RCS loops and the top of the RWSP. During the 1986 event, SDC suction flow could not be quickly restored, apparently because steam was flashing in the loop seal. For that event, the procedure for responding to loss of SDC did not adequately address all RCS conditions that

could be expected following a loss of SDC, nor did it provide information on plant features that could complicate recovery. (Although not important in the recovery of the 1986 event, the loop seals would also prevent the use of gravity feed from the RWSP for RCS makeup.)

Diverse Shutdown Cooling Strategies. The availability of diverse SDC recovery strategies can play a significant role in reducing the significance of events that occur at reduced vessel inventory or high decay heat loads. Use of a diverse system to recover SDC would not require the recovery or repair of an initially faulted system, and presumably could be implemented more quickly in many cases.

Many of the new procedures identify diverse methods for residual heat removal. For example, the Braidwood procedure regarding loss of RHR identifies the following alternate core cooling methods:

- bleed and feed using excess letdown through loop drains and normal charging
- steaming intact/non-isolated steam generators (SGs)
- bleed and feed using pressurizer power-operated relief valves (PORVs)
- refuel cavity to fuel pool cooling
- safety injection (SI) pump hot-leg injection
- accumulator injection
- inventory addition via the RWST

All of these methods are not applicable at all times; however, they provide a flexibility significantly greater than a procedure in which just one alternative method is specified in addition to recovery of the faulted RHR system.

Factors That Strongly Influence the Significance of an Event. Analysis of the 10 events confirms the influence of a number of factors on significance. These factors are described below.

(1) High Decay Heat Load. A high decay heat load significantly reduces the time available for SDC recovery before boiling or core uncovery. This,

in turn, increases the probability of failing to recover SDC or implementing alternative cooling strategies, and may also increase the stress level associated with the event. The number of alternative systems that can effectively remove decay heat is also less than at low decay heat loads; that may further complicate recovery.

(2) RCS Inventory. Reduced RCS inventory also reduces the time available for SDC recovery with a similar impact on the reliability of operator actions.

(3) Status of Reactor Vessel Head. Events that occur when the head is removed are typically less significant than those that occur with the head on, since RPV makeup combined with core region boiling will provide residual heat removal.

(4) Availability of Diverse Systems for SDC. The availability of diverse systems that can provide core cooling reduces the risk associated with a loss of SDC, since availability of these systems does not depend on recovery of the RHR system, as discussed above.

(5) Adequate Procedures. The importance of procedures that give detailed information concerning response to a loss of RPV inventory or SDC, plus alternative strategies for recovery.

3 SITE VISITS TO OBSERVE SHUTDOWN OPERATIONS

Small teams of NRC personnel, each comprising from 2 to 4 technical people, observed low-power/shutdown (LPS) operations at 11 nuclear power plant sites during 1991. The teams' main objectives were to observe plant operations during shutdown and learn about the policies, practices, and procedures used to plan outage activities and conduct them safely. The teams' observations, supplemented by data obtained from recent NRC inspections at six other sites, are presented in this chapter. At the 17 sites, 29 units were operating--4 B&W, 5 CE, 6 GE, and 14 W. The report of the team efforts is documented in NUREG-1448.

On the average, a team spent about one week at a site during an outage. During that period, the team interviewed utility personnel at all levels and observed activities taking place in the areas of operations, management, and engineering, including daily meetings of the plant staff to assess progress and problems in the outage work in progress.

3.1 Outage Programs

Programs for conducting outages varied widely among the sites visited.

Susquehanna's program for conduct of outages was among the best. It included: (1) prudent, practical, and well-documented safety principles and practices; (2) an organization dedicated to updating and improving the p... as well as monitoring its use; (3) strong technical input to the program ...om the onsite nuclear safety review group and the corporate PRA group; (4) a controlled program manual concurred in by line management and well known to all personnel stationed at the site; and (5) training on the program and the program manual for all personnel. The high quality of the program at Susquehanna was reflected in outage activities observed by the team.

Another site that was visited had no comparable program and was poorly prepared and poorly organized, conditions reflected by failure to complete planned work in past outages, long outages, and by the team's other observations of work in progress. At several plants, licensees had neither documentation nor plans to provide any. Two plants made exceptional efforts to keep outages short. At one of these two plants, the team noted examples of less prudent operation than at other plan' it visited. The other plant had a greater number of recent shutdown-related events than any plant visited.

3.1.1 Safety Principles

Well-founded safety principles play a significant role in an outage program. Sites visited varied widely in this area. A high priority was seldom placed on such principles, and sometimes safety was based upon individual philosophies. Often, principles were "understood" in contrast to being clearly defined in a documented management directive.

Some licensees emphasized safety in outage planning and during outage meetings. They posted critical safety boundaries at key locations and identified and tracked critical safety equipment with as much emphasis as given to critical path. Some PWR licensees were particularly sensitive to midloop and reduced inventory operation. One site presented the following good safety principles in its program:

(1) Minimize time at reduced inventory.
(2) Maximize pathways for adding water to the RCS.
(3) Maximize availability of important support systems.
(4) Minimize activities requiring midloop operation.
(5) Maximize time with no fuel in the RV.

Some sites visited gave indepth consideration to such safety are s as criticality, containment, instrument air, electric power, gravity feed, SGs availability (in case of RCS boiling), use of firewater, and many other areas. Others relied upon an ad hoc approach should problems arise.

3.1.2 Safety Practices

A wide variety of safety practices was noted. Some utilities adhered to a "train outage" concept, removing an entire train, including electrical equipment, pumps, controls, and valves, from service. The other train was "protected, no work was allowed on it. Stated benefits were avoidance of train swaps, minimization of mistakes, and simplification of the operator's job. A "block" approach was also used in which a boundary was established and work was allowed within that boundary as long as no water was moved. Other utilities practiced different approaches that may allow more flexibility, but placed greater dependence on their personnel to avoid conflicts. Other safety practices observed by the team included:

(1) Provide sufficient equipment that no single failure of an active component will result in loss of residual heat removal.

(2) Add one injection system or train to that required by technical specifications (TS).

(3) Provide multiple power supplies, batteries, charging pumps, and such.

(4) Always have one ECCS available.

(5) Comply with TS; these are sufficient to ensure safety. (This practice is not endorsed by the staff.)

3.1.3 Contingency Planning

Some licensees provided in-depth preparation for backup cooling, whereas others placed more reliance on ad hoc approaches. Backup cooling includes such techniques as gravity feed, allowing RCS boiling in PWRs with condensation in SGs, and use of firewater. Again, there were many variations in both capability and planning. Some PWR licensees planned SG availability; others did not. Some who planned for the use of firewater and staged spool pieces had procedures; others did not. Most PWRs had some gravity feed capability during some aspects of shutdown operation; others did not. Those that did may or may not have had good coverage in procedures. No site visited had planned ECCS accumulator usage. All of these capabilities are potentially important and could effectively terminate many events.

3.1.4 Outage Planning

Planning ranged from initiating work a few months before an outage was scheduled, to having plans that covered the life of the plant, including anticipated license extensions. There was evidence that good planning, including experience, averted many outage difficulties. Conversely, poor planning appeared to be a cause of such outage difficulties as extended schedules and failure to complete work.

The following items provide additional perspective regarding planning adequacy and effectiveness:

(1) Well-planned and tightly controlled outage plans allowed for increase in the scope and number of unanticipated activities that seldom exceeded 10 to 20 percent. Conversely, growths of 40 percent and more than 100 percent correlated with outages that lasted longer than planned, that were not well managed, and that sometimes resulted in a return to power with significant work unaccomplished.

(2) Some licensees could enter an unscheduled outage and have a complete outage plan within hours. Others had no bases and worked only on the item causing the shutdown. In one case, a licensee entered a refueling outage a month early but accomplished little work before the originally scheduled start date. Another licensee entered a refueling outage a month early, moved the completion date up, and completed the outage in the original time allotted (a month early when compared to the original plan).

(3) In smaller, less-complicated plants, highly experienced licensee staffs could conduct apparently well-coordinated refueling outages with only a few months of planning. Key contributing factors appeared to be: having

few inexperienced people, having the experience of many refueling outages, having a good plan that was prepared quickly, and anticipating material needs well in advance of preparing the plan. Some other licensees, both experienced and relatively inexperienced, had what were judged as relatively poor plans, and their outages appeared to be in some disarray. Finally, some licensees with few refueling outages were able to conduct outages on schedule when they had good plans.

3.1.5 Outage Duration

Safety criteria and implementation effectiveness appeared to be more important to safety than outage duration. Refueling outage durations beyond roughly two months did not appear to increase safety. Conversely, a less-prudent safety approach may be instrumental in shortening outages. However, outage duration was also a function of plant type, the work to be done, planning, and implementation. A short outage was not necessarily an outage where safety has been reduced to shorten the outage, although shortness was an indicator that one should look closely to see how the short schedule was achieved.

The teams observed that several licensees felt pressured to reduce outage time further than they judged to be prudent. Reasons given included being rated by others on the basis of a short outage time, and being driven toward a fuel critical path to shorten outage time.

Numerous approaches to planning affected outage time, including:

(1) Do not reduce refueling outage time below a somewhat judgmental minimum because safety might be jeopardized (several licensees). Typically, these licensees applied safety criteria throughout the outage and these criteria sometimes determined critical path.

(2) Define one critical path, such as the refueling floor, and normally force everything else to fit.

(3) Allow critical paths to float depending upon the work schedule. Safety considerations may influence critical path. (Often, items 1 and 3 were followed simultaneously.)

(4) Describe the work and suggest schedules to "corporate headquarters." Receive or negotiate an allowable outage time.

3.1.6 Outage Experience

All licensees incorporated outage experience into planning and found feedback useful. Most provided for feedback during an outage. Some conducted team meetings immediately after completing significant tasks; others met following the outage. Most compiled outage reports and used these in planning the next outage. Typical results include:

(1) Place personnel with operations background into key positions and areas for planning and conducting outages.

(2) Locate the outage control locations ("war room") close to the control room (CR) to facilitate communication.

(3) Use an SRO who is adjacent to, but not actually in, the CR to handle the work orders.

3.2 Conduct of Outages

Typically, outages were conducted with a licensed shift or unit supervisor who controlled tagouts and approved each work package before initiation of day-to-day work. The daily (and other) outage meetings also provided an opportunity for identifying issues. Beyond this, various approaches were used, ranging from individuals who had their own criteria to various depths of written or unwritten guidance or criteria.

Some licensees were protective of critical equipment and made sure everyone was sensitive to such issues. For example, one licensee protected the operable train of safety equipment by roping off the areas and by identifying the operable train on every daily plan. Similar protected train approaches, including identification in the daily meetings, were found at several plants. Other techniques included providing critical plant parameters in the control room.

Licensees often changed their organizations for an outage, although some operated by incorporating shutdown features into the organization used for power operation and made few actual organization changes. There was a general trend to emphasize operations experience for outage positions at all levels. Licensees who had emphasized such experience considered it to be very beneficial in conducting a satisfactory outage.

Significant variations existed among sites visited in the ratio of utility manpower and total manpower, and in the percentage of personnel involved in the previous outage. Utilities that had a high percentage of people experienced in previous outages at that facility considered such experience to be a significant benefit. Among advantages cited were familiarity with the plant, less training, higher quality, shorter outages, and better motivated people.

Some licensees used task forces and "high impact teams" for critical-path and near-critical-path tasks. These groups were composed of experienced personnel who had performed the same function in past outages.

Contractors were used to various depths by different licensees. Their capabilities, licensee supervision, and influence on outages varied widely. Some licensees worked closely with their contractors and supervised them closely. These licensees appeared to get better results than those who neither carefully trained nor supervised their contractors. Previous contractor experience at the site was often stated to be an advantage and licensees often tried to use the same contractor from outage to outage.

Interestingly, a large plant staff did not translate into an effective outage, nor did a smaller staff at a "small" plant translate into an ineffective outage. Staff size also did not necessarily correlate with safe operating practices, although the team did encounter areas that were weak because they lacked manpower. Those plants judged to have the most effective safety programs were adequately staffed in areas directly related to safety, were well organized overall, and appeared to conduct effective outages.

All utilities conducted periodic reviews during outages. Typically, these involved overview and specialized meetings that were held once or twice a day. All levels of plant personnel and all disciplines were involved. All utilities provided computer-generated outage schedules in several formats and updated some of these every day (or more often). Schedules typically covered a day, 3 days, 7 days, and the complete outage, and further provided a breakdown ranging from an overview through complete scheduling of all activities. Critical-path scheduling was seen often. Some utilities provided prominent safety information on their schedules; others did not.

Most daily meetings appeared well focused and to the point. Achievement appeared to vary widely. Most expectations were routinely met at some plants, but at others the outage appeared to be in disarray.

A commonly applied test for a satisfactory outage was meeting or bettering the outage schedule. Corollary tests were meeting ALARA (as low as reasonably achievable) goals, avoiding personnel injuries, completing planned work, not having to repeat work during power operation because it was not done well during the outage, and not having reportable events.

3.2.1 Operator Training

Licensees often conducted extensive training immediately before a scheduled outage, a practice judged necessary by most licensees because of the specialized nature of, and the lack of everyday exposure to, LPS operation. This was not always done, however, and minimal training was evident at some sites.

Most licensees reported training for LPS operation, often with concentration immediately before an outage so that operators would be completely up to date.

Some operators and instructors said they thought LPS operation was important, but that the NRC had imp   otherwise by not emphasizing it more in qualification exams and simulator evaluations. Others felt that strong NRC interest in training was reflected in Generic Letter (GL) 88-17 inspections and independent resident inspector followup. Although GL 88-17 coverage was limited, the information has been applied to a wider range of PWR plant conditions by licensees.

LPS operations training was often specialized. Some licensees provided concentrated study in unique aspects of the outage to the operating shift expected to handle those aspects of the outage. Training often involved specific equipment, such as valves, RCP seals, and SG manways. Capabilities such as a control rod handling machine mockup for a BWR, SG plena mockups, valves, pumps, and an EDG model for maintenance training were encountered.

As in many other areas, the quality and scope of training was varied, and ranged from:

> Outage training is completed before the outage. Training for power operation with simulator upgrades is conducted before leaving the outage. Special tests are addressed as are evolutions, primary manway and nozzle dam work, level indication problems, procedures, and consequences of what can happen. Procedures changes, including background, are covered before crews take the watch.

to:

> Many plant operators have not had overall systems training for several years and have had no formal outage-specific training since the initial response to GL 88-17.

3.2.2 Stress on Personnel

Although the teams considered stress in general, it was investigated in depth at only one plant. This licensee emphasized short outages, and operators perceived their achievement as related to outage time. Four operators (of seven interviewed in depth) said the outages were too short. Much of the direct outage coordination was conducted from the CR, which was smaller than many multiple-unit CRs. In many instances, such activities appeared to affect plant operation. Further, all operators said the work load was high or very high. Operators also said they met the schedule with difficulty, that they sometimes took on more work than they could handle, that they had to cut corners to stay on schedule and then had to make repairs later, that they wrote procedures at the last minute in the CR, operated without some procedures, and had poor procedures for shutdown; all of the seven operators interviewed said they were poorly trained or that they had significant reservations regarding training. There were many other similar comments. All seven operators said stress was self-generated, and six also identified stress caused by pressure from non-operations personnel. Four operators said stress was severe enough to be a problem. These operators were working four 12-hour shifts followed by a break. No operator stated working hours were too long or that working hours contributed to a problem. This plant was judged to have significant operator stress problems that were reflected in numerous mistakes.

3.2.3 Technical Specifications

No TS were applicable during much of a refueling outage at one site as long as temperature measured at the RHR pump remained below 140°F or 200°F, depending upon the interpretation. (Note that this temperature is unlikely to increase if the RHR pump is not running.) Another site had no TS on emergency diesel generators (EDGs), batteries, and service water during shutdown operation. No plant visited had complete TS coverage.

Most of the industry stated that TS did not fully address LPS operations. The single exception reported that it planned outages on the basis of TS, and this was sufficient to ensure safety. Many personnel commented that existing TS were more appropriate to power operations than to LPS conditions.

Similarly, licensees were concerned with TS that caused extra work, resulted in extra dose, and sent an undesirable message to plant personnel. One example cited was the requirement for an operational pressurizer code safety valve while large openings existed in the RCS. The licensee estimated sever l hours of work and 500 mr of dose were involved to unnecessarily install and then remove the valve.

3.3 Plant and Hardware Configurations

The teams observed that configurations of plant systems and components used by licensees during outages varied widely among plants visited. During the visits, the teams examined configurations of equipment throughout the plants, including regions outside the protected area.

The teams' observations in selected areas are presented below.

3.3.1 Fuel Offload

The fuel at some units was regularly offloaded; some may or may not be offloaded. The fuel at other units would be offloaded only if there was no reasonable alternative.

An often-cited safety advantage for offloading was flexibility available because no fuel was in the RV, and the associated decrease of mistakes leading to a fuel cooling concern. Other considerations included: loss of fuel pool cooling, flexibility in providing fuel cooling if systems were lost, fuel storage volume heatup rate upon loss of cooling, criticality, reduced operator stress due to avoidance of such conditions as midloop operation, and the potential to damage fuel during handling. Fuel offload had a significant advantage in that an early midloop operation and sometimes all midloops, can be avoided, although not all licensees who offloaded also avoided an early midloop operation.

Several licensees performed an incore fuel shuffle and reported they encountered no problems with moving fuel within the core. They said that a complete core offload would lengthen their outages. Conversely, several licensees (both PWRs and BWRs) routinely performed a complete core offload, which they said was safer and provided more flexibility. Several licensees reported the offload path was faster than, or at least as fast as, an incore shuffle. Others offloaded or not on the basis of the planned outage work. Some decisions were based upon such considerations as the configuration (offload appeared to be difficult in Mark III BWRs), fuel distortion history, gains achievable with no fuel in the RV, and the reliability of the fuel handling machine.

3.3.2 Midloop Operation*

Midloop operation concerns appeared to have influenced outage planning at many sites, but not at others. The team observed licensees who:

(1) Do not enter midloop operation under any circumstances.

(2) Do not permit early midloop operation and defueling before installing nozzle dams.

(3) Apply special midloop criteria to refueling outages, but deviate for an unscheduled outage.

(4) Routinely enter midloop within a few days to a week of power operation.

Some licensees required an additional operator in the control room for midloop operation. Another, whose hardware was particularly sensitive, required three additional operators who had specific responsibilities in the conduct of reduced-inventory operations; that is, operation when the reactor vessel water level is lower than 3 feet below the reactor vessel flange.

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* A midloop condition exists whenever RCS water level is below the top of the flow area of the hot legs at the junction with the reactor vessel.

3.3.3 Venting in PWRs

Reactor coolant system vents were sometimes of insufficient size, being smaller than planned and smaller than required by licensee procedures. Licensee personnel who recognized the implications were often unaware of these conditions.

Some licensees provided an RCS vent by removing one or more safety valves from the pressurizer. Others removed a pressurizer manway. If boiling develops, significant backpressure can occur from friction in the surge pipe, water traps, and the elevation head of the water held up in the pressurizer.

Licensee personnel usually used covers or screens to keep foreign material from falling into pressurizer openings. These were often "make shift" affairs that could cause additional backpressure. The teams interviewed licensee personnel who were often unaware of the covers or screens.

3.3.4 Nozzle Dams* in PWRs

Some PWR plants use nozzle dams and some do not. The recent trend in B&W nuclear steam supply system (NSSSs) has been to use them, whereas a few years ago this was seldom so. One licensee reported outage savings of close to a week attributable to the use of nozzle dams, whereas another had them but did not use them and typically spent 3 to 14 days at midloop. Others indicated they might be at midloop for close to a month without them.

One licensee indicated there was no analysis to cover midloop operation with both nozzle dams and the RV head installed and such operation would not be

* Nozzle dams are temporary seals installed in RCS primary piping which isolate components such as steam generator from reactor vessel and reactor cavity water so that work can be done on the components.

permitted until the analysis was completed. The teams noted the general incompleteness of analyses of shutdown operation, and also noted this may in part involve SG cooling with nozzle dams isolating some, but not all, SGs.

3.3.5 Electrical Equipment

An outage typically represents times when equipment unavailability is high, unusual electrical lineups exist, and the likelihood of an electrical perturbation is increased by maintenance activities. The teams identified several events that could lead to electrical component damage or loss at some facilities, and concluded that almost all of those identified events could be easily eliminated. The team also found that protection and control of offsite electrical power systems varied.

Approaches to provide ac power included:

(1) Allow cooling via a system powered by a non-safety-related bus with no procedures for providing safety-related power to that bus.

(2) Provide one EDG and one source of offsite power.

(3) Provide one less source of power during shutdown to allow maintenance on one source at a time.

(4) Always have three sources of power, one of which is an EDG. (The site that advocated this did not have an EDG for about 2 weeks with fuel offloaded, but it had a temporary diesel available.)

(5) Have both EDGs operable when in midloop operation. (One licensee stated it did not consider it prudent to stay at midloop conditions with only one EDG and would leave midloop operation if the EDG could not be made operable quickly.)

(6) Allow both EDGs out of service when the fuel is offloaded.

(7) For midloop operation, normally have two EDGs and two offsite sources and allow no battery work, no reserve auxiliary transformer outage, no work that affects safeguards buses, or anything that affects the RCS. Otherwise require two off site and one on site always.

(8) Make at least three separate ac power sources available to the plant vital buses any time two RHR pumps are required to be operable. In practice, one of the sources has to be an EDG.

Additional variations include switchyard restrictions, restricting work on, or access to, vital areas such as near an operable EDG or operable electrical equipment, information requirements, administrative procedures, and whether variations are permitted and what level of management is necessary to approve such variations.

EDG maintenance and associated testing are usually performed during shutdown, although some licensees were performing this work at power. Also observed was removing an EDG from service via entering action statements immediately before shutdown.

Concerns also involved whether to have EDGs operating or operable. Potential decreases in EDG reliability due to grid disturbances and other perturbations, extensive testing, and running with a small electrical load were identified as potential problems with having EDGs operating.

Most plants had transformers and often breakers within the site's protected area. Switchyards were located nearby, but usually in whole or in part outside the protected area. These switchyards may contain a few transformers, but often contained only breakers and switches. They were usually fenced if outside the protected area, and usually had a locked gate. Often there was a control building within the switchyard, with attendant vehicle traffic. This building was seldom located adjacent to a switchyard entrance gate.

The teams did not observe any evidence of vehicle impacts within switchyards. However, they did find such evidence on both transformers and supports located within unfenced areas within site protected areas; they also found a number of damaged fences. In one case, the source of safety-related offsite power entered the turbine building roughly one foot from where heavy trucks and trailers were sometimes parked, and was protected only by an ordinary chain-link fence. Fire hydrants at all sites were protected by a profusion of concrete-filled pipes, but at many sites important transformers within a few feet of the hydrants were unprotected. Switchyards were typically full of towers and bus supports. Some of the weakest supports were located in the corners and typically supported ring buses--loss of which could cause a loss of offsite power. Yet these corner towers were often the towers most exposed to traffic within the switchyard, and were unprotected.

Some sites maintained CR control over switchyards outside the site's protected area. Other switchyards could be entered by anyone having a key to the padlock; often, a utility staff member not assigned to the nuclear facility had a key, and sometimes someone who was not even an employee of the same utility had a key. Sometimes control was provided if the plant was in a sensitive condition, such as a PWR in midloop operation, but at other sites switchyard work could proceed with little or no consideration of the nuclear plant status. At one plant, the team found the switchyard gate open and no one monitoring traffic at the gate. This switchyard was in an uncontrolled area.

3.3.6 Onsite Sources of ac Power

Onsite sources of electric power that were observed included diesel generators, hydro units, portable power supplies. The most common source of safety-related power was EDGs.

Many variations in EDGs and configurations were seen. Size ranged from a fraction of a megawatt to 8 MW. One two-unit plant had two EDGs and routinely performed maintenance on one EDG while one unit was at 100 percent power and the other was in a refueling outage. That site planned to add two more diesels. In contrast, the Susquehanna two-unit plant had five EDGs. The fifth could be used as a complete replacement for any of the other four with no difference in CR indication and plant operation. Susquehanna also provided a portable diesel for battery charging and other uses if an extended loss of all ac power should occur.

Roughly a third of the plants visited had the capability to resupply the EDG starting air tanks without ac power. The dominant method was a single-cylinder, diesel-powered compressor; but instrument air, a cross-connect with another EDG's air supply, and changing the drive belt from the electric motor to a one-cylinder engine were also observed.

3.3.7 Containment Status

Some PWR licensees closed the containments for conditions other than refueling; others did not, unless they entered a condition as described in GL 88-17. Some did not remove their equipment hatches during routine refueling outages; others did. Some provided containment closure capability that would withstand roughly the containment capability; others could lose containment integrity at roughly 1 psi. Some had proven containment integrity; others did not, and may not have attained an integral containment that meets GL 88-17 recommendations.

BWR secondary containments were judged unlikely to prevent an early release following initiation of boiling with an open RCS or during potential severe-core-damage scenarios. Among the BWRs, only the Mark III primary containment appeared potentially capable of preventing an early release without hardware modifications during such events. See Section 6.9 for a more complete assessment of containment capability. In general, no plans were found in BWRs for containment closure or for dealing with conditions under which the containment may be challenged.

3.3.8 Containment Equipment Hatches

A majority of the equipment hatches viewed at PWR sites visited can be replaced without electrical power. See Section 6.9.3 for a full discussion of equipment hatch design and operation. Many licensees appeared to be failing to check for adequate closure as addressed in GL 88-17.

The team learned that Arknasas Nuclear One had a requirement that an equipment hatch be capable of closure within approximately 15 minutes of a loss of RHR. Responsibilities were established for such actions as notification of loss of RHR, containment evacuation, closure operations, and verifications. Tools were kept in a closed box at the hatch and were clearly labeled "for emergency use only." Unannounced closure exercises had been conducted. Few other sites visited were as well prepared.

A common weakness was failure to check for adequate closure. GL 88-17 specified "no gaps," not the "four bolts" commonly observed. The four-bolt specification appeared to be insufficient at some plants with inside hatches (hatches that would be forced closed by containment pressurization).

Oconee provided a small standby generator in case ac power was lost. This could be immediately used to power the winches that normally raise and lower the hatch. This appeared to be an excellent approach to one of the problems of loss of ac power.

3.3.9 Containment Control

Some licensees carefully controlled containment penetrations during LPS operation. Others were concerned only with TS requirements regarding fuel movement and reduced inventory/midloop commitments in their response to GL 88-17. Provisions were found to bring services such as hoses and electrical wires into the containment via unused containment penetrations at several sites. Such provisions made it easier to close the equipment and personnel hatches. Some licensees simply removed a blind flange and passed wires or hoses through the opening. Others provided a manifold arrangement that may effectively eliminate most of the open penetrations. Occasionally, a permanent connection or an adaptation of a penetration such as was used for containment pressurization was found for introducing temporary utilities. U-pipes filled with water were observed in use as a containment penetration seal. These were judged to be of little use in protecting against an accident involving significant steam production or a core melt.

A number of licensees planned to initiate containment closure immediately upon loss of RHR. Others were less stringent, including such possibilities as initiating closure if temperature exceeds 200°F. That approach is likely to allow boiling before containment closure, and boiling may make it impossible to continue closure operations. In one case, the licensee assumed personnel could work inside the containment in a 160°F environment while accomplishing equipment hatch closure. More detail on this topic is included in Section 6.9.4 "Containment Environment Consideratons for Personnel Access."

Knowledge of what must be closed and providing the resources to actually close the openings and/or penetrations under realistic conditions were often overlooked. Tracking openings, providing procedures, and conducting walkthroughs that accounted for conditions reasonably expected to exist were seldom found.

3.3.10 Debris in Containment

Blocking a PWR containment sump with debris from outage work may prevent effective recirculation of reactor coolant following an accident during shutdown. For example, PWR emergency core cooling (ECC) sump screens were removed during refueling outages at some sites, and at others the screens were covered with heavy plastic sheeting. In one plant, one screen was removed and the other was 10-percent uncovered to allow a recirculation capability. In another, one sump was open and the other was closed. Similar conditions were seen in plants with ECC connections in the bottom of the containments without a sump. In one, both filters were removed to expose the pipe opening; in another, the filters were in place. Actual and potential debris existed at all of these sites, but was seldom considered with respect to recirculation capability during shutdown.

3.3.11 Temperature Instrumentation

Core temperature during shutdown in PWRs was obtained by measuring water temperature just above the core by thermocouples (T/Cs). Other temperature indications required an operating RHR system for accurate indication of meaningful RCS and core temperature over a wide span of RCS conditions.

Although this was addressed in GL 88-17, many operators were still unaware of the potential error associated with lack of flow. Numerous PWR heatup events have occurred where no temperature indication was available, although the frequency is decreasing with implementation of GL 88-17's recommendations. However, the team often observed poor application of the temperature coverage recommendation, principally involving not providing temperature indications for extended periods of time, restricting the indication to reduced inventory conditions, and failure to provide suitable alarms. Licensees who emphasized temperature indication generally provided temperature while the head was on the RV with the exception of within 30 minutes to 2 hours of head movement.

BWR coolant temperature was obtained by measuring the RV wall temperature and assuming natural circulation in the RV. The natural circulation assumption is not valid if water level is lower than the circulation paths in the steam separator. This was often unrecognized, and BWRs have encountered significant heatup with no indication of increasing temperature provided to the operators.

3.3.12 Water Level Instrumentation

BWRs were equipped with multiple water level indications that were on scale during both power and shutdown operation. PW' were often operated with all of the "permanent" level indicatiºns off scale or inoperative during shutdown. PWR licensees have added level instrumentation to cover shutdown operation in response to GL 88-17. The observed quality in the BWRs was generally superior to the PWRs. The team often found multiple damaged and/or incorrectly installed instrument tubing inside PWR containments. Only one short tube section with an incorrect slope was found in a BWR. Many personnel described problems with maintaining accurate level indication in PWRs. No one described this problem in BWRs.

BWR level systems typically used a condensing pot to ensure that connecting pipes remain full, yet no condensate is generat d during shutdown. No one indicat d this could lead to level indication error, nor did anyone identify this as a potential problem.

PWR level indications have significantly improved in the last 3 years. All PWRs now indicate level on the control board. In-containment installations often (but not always) showed evidence of professional installation that was lacking several years ago. Much less reliance was being placed on temporary tubing runs. Several licensees were still working to meet GL 88-17 recommendations.

Some PWRs were equipped with ultrasonic hot-leg and cold-leg level indications. A few have been in operation for years, and this indication has been used in foreign plants for some time. Most licensees appeared satisfied with indication accuracy and reliability, although problems were reported with equipment obtained from one vendor.

3.3.13 RCS Pressure Indication

RCS pressure indications were generally wide range and not appropriate for monitoring shutdown operation. A number of operations personnel identified that the computer provided monitoring and cathode-ray tube (CRT) indications that had a more-sensitive range.

3.3.14 RHR System Status Indication

GL 88-17 identified pump motor current, RHR pump noise, or RHR pump suction pressure for monitoring RHR operation in PWRs. Although many licensees have followed the recommendations in GL 88-17, some responses have been minimal. Weaknesses observed included failure to consider sampling rate when monitoring parameters, failure to provide trending information, too wide a pressure range to permit observation of behavior, and RHR systems operating with temperature off-scale low.

3.3.15 Dedicated Shutdown Annunciators

Numerous control room annunciators were typically lit during shutdown conditions. Arkansas Nuclear One had installed an annunciator board that addressed major shutdown parameters and was making it operational--the only such panel observed. Several operators indicated that even a grouping of existing parameters into an easily recognized pattern would be better than what they have. Others said they were familiar with the lit annunciators and had no difficulty recognizing an unusual pattern.

4 PROBABILISTIC RISK ASSESSMENTS

Risks associated with shutdown and refueling conditions have not been extensively studied and are not as well understood as those associated with power operation. There are few studies that address the full scope of understanding about shutdown risk in PWRs and none for BWRs. The Grand Gulf and Surry shutdown, probabilistic risk assessment (PRA) studies (currently at a preliminary level 1 stage) offer a better understanding of accidents and risks a plant can encounter during a refueling outage. However, to gain a deeper understanding of the subject requires more research and study of risks associated with outage conditions.

The following PRAs, including Grand Gulf, Surry, and foreign PRA studies are summarized here to identify significant issues and insights associated with nuclear power plant activities during shutdown and refueling outages.

4.1 NSAC-84

NSAC-84 was an extension of the Zion Probabilistic Safety Study completed in 1981. Procedural event trees were developed to account for changes in plant conditions during shutdown. Human errors and equipment failures unrelated to procedures were also considered. The initiating events included in the study consisted of: loss of RHR cooling, loss-of-coolant accidents (LOCAs), cold overpressurization (excess of charging, over-letdown, or an inadvertent safety injection). A shutdown database specific to Zion was developed from plant records and used in quantification.

Findings

The mean core-damage frequency (CDF) at shutdown was estimated to be 1.8E-5 per reactor-year.

Examination of the top 10 core-damage sequences revealed the following:

(1) Failures during reduced-inventory operation (including equipment unavailabilities and operator errors) appear in eight sequences, totaling 61 percent of the total CDF, while failure of the operator to respond during reduced-inventory operation appeared in five sequences, accounting for 44 percent of the total CDF.

(2) Malfunctions of RHR components require some type of operator intervention, all shutdown core-damage scenarios (due to overdraining of RCS, LOCAs, and RHR suction valve trips) are sensitive to the operator's failure to restore core cooling. The operator's failure to determine the proper actions to restore shutdown cooling appeared in six sequences, accounting for 56 percent of the total CDF.

(3) Loss of RHR cooling (primarily pump and suction valve trips) was the initiating event in eight sequences, totaling 56 percent of the CDF, while a LOCA was the initiating event in the other two sequences, totaling 6 percent of the CDF.

4.2 NUREG/CR-5015 (Loss of RHR in PWRs)

NUREG/CR-5015 was issued in response to Generic Issue 99 concerning the loss of RHR in PWRs during cold shutdown. This study used the NSAC-84 methodology (based on the Zion plant configuration) with several modifications which included the consideration of loss of offsite power (LOOP) events using a separate event tree and the use of generic event frequencies from PWR experience over a 10-year period from 1976 to 1986.

Findings

The mean CDF at shutdown was estimated to be 5.2E-5 per reactor-year, with the following breakdown by initiating event:

- loss of RHR 82%
- loss of offsite power 10%
- loss-of-coolant accident 8%

Examination of the findings reveals the following:

- Failure of the operator to diagnose that a loss of cooling has occurred and to successfully restore it accounted for 74 percent of the total CDF. The two dominant core damage sequences involved a loss of RHR pump suction due to overdraining of the RCS totaling 64 percent of the total CDF, while two other events involving the failure to restore RHR following a LOOP, accounted for 10 percent of the total CDF.

- Operator error during midloop operation accounted for 68 percent of the total CDF.

The findings of NUREG/CR-5015 appeared to correspond with those of NSAC-84. Operator errors dominated the risk, particularly during midloop operation. Loss of offsite power events resulted in a relatively small contribution, 10 percent of the total CDF.

4.3 Seabrook PRA for Shutdown Operation

The Seabrook PRA information was collected from a number of presentations the licensee made to the NRC. This study supplemented the level 3 Seabrook PRA by examining the likelihood of core damage for the plant in standard Modes 4 (hot shutdown), 5 (cold shutdown), and 6 (refueling). Radiological source terms and public health consequences were also considered. The approach used to model accident sequences was similar to that used in NSAC-84 with several enhancements which included: fire and flood initiating events unique to plant shutdown were quantified and considered, an uncertainty analysis of the results was performed, the PWR experience database from NSAC-52 was updated and examined with insights being incorporated into plant shutdown models, thermal-hydraulic calculations for determining of time to core boiling and uncovery wer performed for different plant configurations after shutdown.

Findings

The total shutdown CDF was 4.5E-5 per reactor-year while the total full-power CDF from Seabrook's individual plant examination (IPE) was 1.1E-4 per reactor-year.

Loss of RHR initiators contributed 82 percent of the CDF. About 71 percent of the total CDF occurred with the RCS vented and partially drained (i.e., midloop). The largest contributors to RHR failure were the hardware failure of an operating RHR pump due to its long mission time, and the loss of RHR suction due to either inadvertent closure of the RHR suction valves or low-level cavitation when the RCS was drained (operator error events)

Although LOCAs represented only 18 percent of the total CDF, they dominated early health risks. When the RCS was filled, the equipment hatch integrity was not required (the hatch integrity is required during reduced inventory conditions). Under these conditions, a postulated LOCA would leave the operator only a short time for restoring core cooling. The Seabrook study concluded that it was unlikely that the equipment hatch could be closed before the containment was uninhabitable. This scenario indicated the need for controls on containment integrity and emergency response procedures for LOCA events during shutdown. This insight might have been overlooked if the level 2 analysis was not performed. A major contribution to this frequency (accounts for 8 percent) was LOCAs from overpressure events resulting from stuck-open RHR relief valves or ruptured RHR pump seals.

4.4 Brunswick PRA for Loss of RHR Removal (NSAC-83)

This study performed a quantitative probabilistic evaluation of the reliability of RHR equipment given a variety of scenarios in which the plant's RHR function is challenged, including following transients that resulted in reactor scrams during a planned shutdown and during a cold shutdown scenario over time which could lead to a suppression pool temperature exceeding 200°F (assumed core damage). Other functions, such as inventory control, reactivity, and containment control, were not addressed. Brunswick-specific failure data were used, and generic probability values for operational errors were included as basic events in the fault trees.

Findings

The probability of a loss of RHR during cold shutdown was estimated to be 7.0E-6 per reactor-year. There were no dominant accident sequences listed. On the basis of an evaluation of the methodology, models, and findings presented in the report, the following is a list of major contributors to the loss of RHR during shutdown.

- loss of offsite power
- equipment unavailability due to maintenance
- loss of service water
- RHR and RHRSW pump failures
- pump suction valve failures
- common mode failure of RHR heat exchangers

4.5 Sequoyah LOCA in Cold Shutdown (SAIC)

This study addressed the probability of a core-melt accident in cold shutdown (Mode 5) which was initiated by a postulated loss-of-coolant accident (LOCA) at the Sequoyah Nuclear Plant. Two LOCA initiating events were considered: safe-shutdown earthquake and operator error (RHR-induced LOCAs were not considered). A total of 20 cases were analyzed with varying assumptions regarding time of LOCA initiation following a shutdown, LOCA size, availability of offsite power, and maintenance status.

Findings

The postulated core-melt frequency was estimated to be in the range from 7.53E-5 to 8.5E-7 per reactor-year. The major contributors to core-melt frequency included the following:

- operator-induced LOCAs
- availability of power to plant equipment
- maintenance
- operator errors during response (lack of procedures for securing equipment, inadequate RCS monitoring equipment)
- failure of an airbound RHR pump
- RHR suction failure

4.6 International Studies

The staff gained significant insights from studies performed in France. These studies focused on identifying the dominant contributors to risk from dilution events at shutdown and loss of RHR during midloop operation. The main PRA study excluded such external events as fires, floods, earthquakes, and source terms. The French categorized this study as a level 1 PRA.

4.7 NRC Shutdown PRA for Grand Gulf (Coarse Screening Study)

The study focused on the potential accidents that could occur at Grand Gulf during low-power operating conditions and identified those operating conditions or accident-initiating events that required further detailed study. This study was considered a preliminary level 1 PRA. The initiating events included in the study were: transients (loss of feedwater events, and loss-of-offsite-power events), LOCAs, loss of residual heat removal, special events (fire and flood), and hazard events.

Findings

Overall, the Grand Gulf study indicated the importance of anticipated operator recovery actions. It is important to note that 22 percent of the potentially high CDR sequences had 14 or more hours for the operator to recover. Many of the potentially high CDR sequences had at least 2 to 2.5 hours for recovery.

The findings of this coarse screening study indicated that 26 percent of the 1163 accident sequences were categorized as having potentially high CDF.

About 30 percent of the 1163 accident sequences were considered to have potentially medium CDF.

A postaccident human error probability of 1 was assumed and few human actions were given credit in recovery.

Two important initiating events were noted which can lead to core damage; they were loss of instrument air as a unique initiating event and loss of the RHR system.

Many potentially high and medium CDF occurred in plant operating stages from cold shutdown to refueling with water level raised to the steamlines, and refueling with water level raised to the steamlines.

In the potentially high CDF category, approximately 88 percent of the sequences occurred in an open containment situation, and about 38 percent of the sequences involved an open containment for the potentially medium CDF.

4.8 NRC Shutdown PRA for Surry (Coarse Screening Study)

This study assessed the potential core-melt accidents initiated by internal events as well as by fires and floods during low-power operation and shutdown. The potential CDF, accident sequences, and other qualitative and quantitative findings from the study will be used to make comparisons with those of accidents initiated during full power operation as assessed in NUREG-1150. This study is considered a coarse screening analysis for a level 1 PRA.

Findings

The coarse screening PRA analysis of Surry recognized that some plant configurations in an outage (low-power operation) were found to be more vulnerable than the others. These plant configurations were based on Surry operational practices which routinely involved entering LCO action statements during shutdown operations.

Surry entered midloop operation at a rate of approximately twice a year. The midloop condition can occur within a day after shutdown with decay heat as high as 12.4 MW. Core uncovery can occur in this condition as early as 1.5 hours after a loss of core cooling.

The use of temporary seals at the seal table as a temporary pressure boundary during shutdown operation can result in an immediate primary system leakage upon loss of core cooling capability and upon an RCS pressure increase. Further pressurization can lead to core uncovery quickly.

In a refueling outage when maintenance is conducted with the loops drained, reactor coolant loops can be isolated for extended periods of time, and one or more steam generators (SGs) will be isolated from the RCS, thus reducing the capability to dissipate heat through the SG secondary side. The station blackout (SBO) with the plant shutdown scenario at Surry represented a difficult situation for controlling the plant when the AFW lines to each SG are isolated with the MOVs located inside the containment and the SG relief valves fail closed on loss of air and can not be opened manually at the valves. This situation is unique at the Surry plant. The Surry emergency procedure regarding loss of all ac power instructs operators that it is essential to the mitigation of an SBO to manually dump steam through turbine bypass valves to the turbine main condenser. In this situation, the operating RHR system which is used to maintain core cooling will be pressurized beyond the system's design pressure for an increase in RCS pressure, and RCS relief valves are not capable of relieving a large volume of steam that would be generated in the vessel. The RHR overpressurization could occur as early as 0.7 hour after an SBO occurred.

The preliminary Surry analysis indicates that maintenance unavailabilities at shutdown were much higher than during power operation. Fewer technical specifications (TS) requirements were applied in the cold-shutdown condition. Inventory and makeup requirements to the RCS are not required in Surry's current TS. However, the operating procedure was written to require one high-head injection and one low-head injection be operable during a reduced-inventory condition as a result of Surry's response to Generic Letter 88-17.

Simultaneous maintenance on redundant trains may take place at Surry during a refueling outage; this was found to be a dominant cause for core damage in this study.

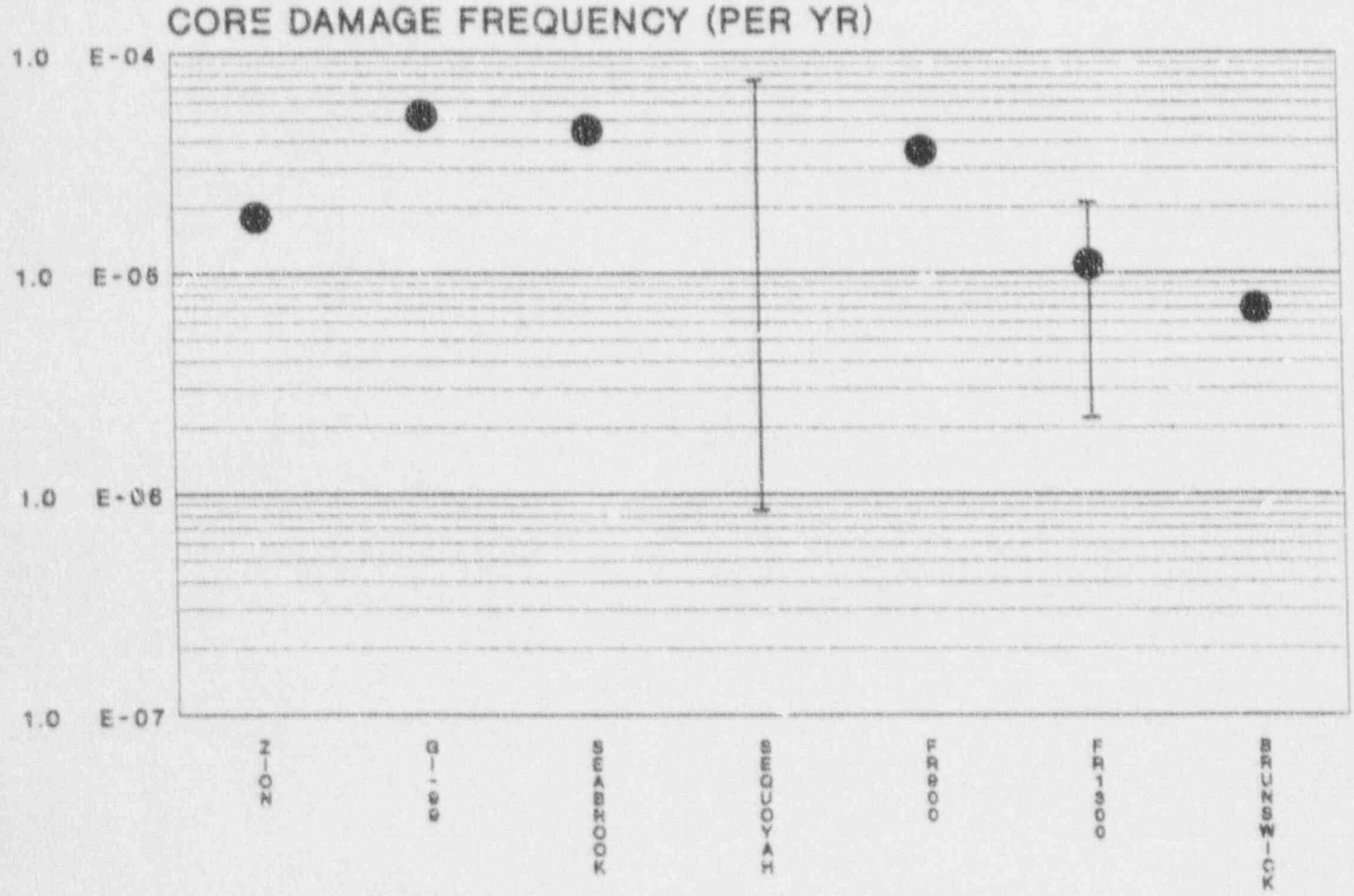
Fire or flood barriers that are used during power operations may be removed during shutdown.

4.9 Findings

Quantitative results of the PRA studies are shown in Figure 4-1. On the basis of the findings from each of the eight studies examined above, the most significant events, from a shutdown-risk perspective, can be summarized as follows:

(1) failures during midloop operation (PWRs)
(2) operator error, especially
  - failure to determine the proper actions to restore shutdown cooling (especially during midloop)
  - procedural deficiencies
(3) loss of RHR shutdown cooling, especially
  - operator error-induced
  - suction valve trips
  - cavitation due to overdraining of the RCS
(4) loss of offsite power
(5) LOCAs, especially
  - operator error-induced
  - stuck-open RHR relief valves
  - ruptured RHR pump seals
  - temporary seals ruptured
(7) loss of service water (BWRs)

SUMMARY OF PRA RESULTS
FIGURE 4-1



5 REGULATORY REQUIREMENTS FOR SHUTDOWN AND LOW-POWER OPERATIONS

The compilation of U. S. requirements and requirements in other countries was done as part of an OECD/CNRA (Organization for Economic Cooperation and Development/Committee on Nuclear Regulatory Activities) study led by the NRC. The results are presented in the October 1991 report NEA/NRA/DCOC(91)2, and are summarized below.

5.1 Facilities in the United States

5.1.1 Technical Specifications

Two types of regulatory requirements address shutdown and low-power operations: design requirements and operational requirements. The regulatory design requirements contained in the general design criteria (GDC) in Appendix A to 10 CFR Part 50 and the quality assurance requirements, in Appendix B to 10 CFR Part 50 do not generally depend on operational mode. The staff has interpreted the GDC requirements in the regulatory guides and the "Standard Review Plan," NUREG-0800.

The technical specifications for individual plants are the primary source of operational requirements to control shutdown and low-power operation. The current standard technical specifications (STS) address specific requirements during shutdown and low-power operation for reactivity control, inventory control, residual heat removal, and containment integrity. The STS requirements vary in degree of coverage and allowable limits when compared with those issued earlier in custom technical specifications.

5.1.1.1 Reactivity Control

The technical specifications requirements for PWRs during shutdown operation include a reduction in the shutdown margin from 1.5 percent to 1.0 percent delta-K/K during cold shutdown. Reactor protection system operability is not required once the reactor is shut down, except that flux monitors must be operable whenever controls can be moved. The restoration of an inactive loop is controlled by temperature and boron concentration limits during cold shutdown and refueling. Boron concentration limits are not applicable for the RWST during hot and cold shutdown and refueling operations, and the boron injection tank is not required to be operable during cold shutdown and refueling. However, sources of unborated water must be isolated from the primary system.

For BWRs, the most reactor protection system operability requirements are not in effect once the reactor is shut down. However, if control rods are being moved, flux monitors must be operable. The feedwater reactor trip may be disabled during the startup mode and the anticipated-transient-without-scram (ATWS) instrumentation is not required during startup. All control rod movement is restricted to one control blade at a time, unless the associated fuel cell contains no fuel. The shutdown margin must be at least 0.38 percent delta-K/K at all times.

5.1.1.2 Inventory Control

For both PWRs and BWRs, leakage limits and leakage detection system operability are not required during cold shutdown and refueling. The following additional requirements apply only to PWRs. Only one train of emergency coolant injection is required during hot shutdown and none is required in cold shutdown or refueling. The RWST is also not required to be operable during cold shutdown or refueling. Instrumentation requirements are controlled by the requirements of the systems supported by the instrumentation, that is, if the injection system is required to be operable, the system instrumentation is required to be operable. In addition, for PWRs, low-temperature overpressure protection is required in the hot- and cold-shutdown and refueling conditions. The requirements are that two PORVs or two RHR relief valves are operable and no more than one train of high-pressure injection can be operable.

For BWRs, two low-pressure injection trains are required during cold shutdown and refueling. This requirement is eliminated (i.e., no injection systems are required to be operable) if the refueling cavity is flooded. As with the PWR instrumentation requirements, the system instrumentation is required to be operable if the system is required to be operable. Cooling water systems associated with the injection systems are also generally required to be operable only when the injection systems are required to be operable, unless required to meet other technical specifications requirements.

5.1.1.3 Residual Heat Removal

In the low-power and shutdown modes, the PWR operability requirements for the residual heat removal function are mode dependent. During hot standby, two reactor coolant loops are required. In hot shutdown, any combination of two RHR loops and reactor coolant loops is acceptable. During cold shutdown, two RHR loops are required unless two steam generators are filled to at least 17 percent of the normal level for the steam generators; then two steam generators and one RHR loop are an acceptable combination. During refueling, two RHR loops or one with the refueling cavity filled are required. Generally, the secondary-side heat removal systems (main and auxiliary feedwater) are not required to be operable during hot and cold shutdown and refueling. However, if a steam generator is being used as a heat removal system during hot shutdown, the condensate storage tank, atmospheric dump valves, and one train of auxiliary feedwater (including instrumentation) must be available.

For BWRs, two divisions of RHR are required (with one operating) in the hot-shutdown, cold-shutdown, and refueling modes. With the refueling cavity flooded during refueling, only one RHR division is required.

One division of electric power is required to be operable in cold shutdown and during refueling, as opposed to two divisions during all other modes of operation. (A division is defined to include both an onsite and an offsite source of ac power.)

5.1.1.4 Containment Integrity

The containment integrity requirements for PWRs are not applicable during cold shutdown and refueling. This includes the operability of the containment spray system. In addition, the containment isolation instrumentation is not required to be operable during hot shutdown. During fuel movement operations, less-restrictive containment isolation requirements are in effect. One air-lock door must be maintained closed and a "four-bolt rule" is in effect for the equipment hatch.

In a BWR, the containment atmosphere can be de-inerted 24 hours before going to cold shutdown. Inerting containment can begin up to 24 hours after entering hot shutdown during restarts. Containment integrity, standby gas treatment system, and containment isolation instrumentation requirements are not applicable during cold shutdown and refueling. However, during fuel movement, the secondary containment must be operable.

The staff is reviewing the range of technical specifications requirements for shutdown and low-power modes, including those in the existing STS and those being developed within the Technical Specifications Improvement Program. In performing this review, the staff has determined that these requirements are generally less restrictive than the requirements in the full-power operations mode. For example, the TS allow fewer operators for PWRs and BWRs during cold-shutdown and refueling operations.

5.1.2 Other Regulatory Requirements or Policies

The staff also identified a number of important facts regarding regulatory requirements or policies pertaining to operator training, use of overtime, emergency planning, fuel handling, and heavy loads, fire protection and procedures.

5.1.2.1 Training (Coverage of Shutdown Conditions on Simulators)

The current Code of Federal Regulations (Title 10, Section 55.45(b)(2)(iv)) requires the simulation facility portion of the operating test will not be administered on other than a certified or approved simulation facility. NRC Regulatory Guide 1.149 endorsed the guidance of the American National Standard Institute's, "Nuclear Power Plant Simulators for Use in Operator Training," ANSI/ANS-3.5-1985. To date, nearly all of the industry's simulators have been certified to meet this guidance.

The ANSI/ANS Standard 3.5-1985 requires simulation of minimum normal activities from cold startup to full power to cold shutdown, excluding operations with the reactor vessel head removed.

5.1.2.2 Policy on Use of Overtime

Generic Letter 82-12 transmitted NRC's "Policy on Factors Causing Fatigue of Operating Personnel at Nuclear Power Plants." This policy provides specific guidance for the control of working hours during shutdown operations. This guidance allows the plant superintendent to approve associated deviations from the guidelines on working hours. The policy applies only to personnel who perform safety-related duties and the individuals who directly supervise them.

5.1.2.3 Fire Protection

The plant technical specifications (TS) allow various safety systems, including fire protection systems, to be taken out of service to facilitate system maintenance, inspection, and testing during shutdown and refueling.

The Appendix R fire protection criteria for the protection of safe shutdown capability does not include those systems important to ensuring an adequate level of residual heat removal during non-power modes of operation.

The current NRC fire protection philosophy (NUREG-0900, SRP Section 9.5.1) does not address shutdown and refueling conditions and the impact a fire may have on the plant's ability to remove decay heat and maintain reactor coolant temperature below saturation conditions.

5.1.2.4 Reporting Requirements

The current NRC regulations require that any operation or condition prohibited by the plant's technical specifications is reportable under 10 CFR 50.73. This includes both power operation and shutdown. However, as discussed earlier there are far fewer technical specifications applicable during shutdown.

5.1.2.5 Onsite Emergency Planning

The current guidance for classification of emergencies for nuclear plants during power operation (found in Appendix 1 to NUREG-0654, FEMA, Revision 1, titled "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants"), does not explicitly address the different modes of nuclear power plant operation.

5.1.2.6 Fuel Handling and Heavy Loads

Plant technical specifications require that fuel handling equipment be tested before use in order to prevent dropping fuel elements.

For both BWRs and PWRs, technical specifications require that a specified level of water be maintained above the reactor vessel head and spent fuel storage pools during refueling.

For PWRs, technical specifications require that penetrations in the containment building be closed or capable of being closed by an operable automatic valve on a high radiation signal in the containment, before initiating the refueling process.

For BWRs, technical specifications require that the integrity of the fuel handling building be assured before handling irradiated fuel.

Technical specifications for PWRs and BWRs require that the spent fuel cooling systems be operable and the water level and temperatures be maintained.

Risks associated with heavy loads are minimized by two alternative objectives as outlined in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."

The potential offsite doses due to heavy loads dropped on the spent fuel must be within 25 percent of the allowable levels in 10 CFR Part 100, while K_{eff} must not be greater than 0.95.

5.1.2.7 Plant Procedures

Appendix B to 10 CFR Part 50 requires that licensees provide control over activities affecting the quality of plant structures, systems, and components that prevent or mitigate the consequences of postulated accidents that could cause undue risk to the health and safety of the public. The control of these structures, systems and components is to be consistent with their importance to safety, and includes maintaining safety during shutdown as well as power operation. Activities affecting quality are to be performed in accordance with procedures or drawings of a type appropriate to the circumstances. Consequently, the regulatory basis now exist to require that licensees have procedures appropriate for the prevention and mitigation of risks associated with low- power and shutdown operations and to require that these procedures are commensurate with the risk to public health and safety.

5.1.3 Bulletins and Generic Letters

NRC use of generic communications, specifically bulletins and generic letters, provides insight into the events of interest and the evolution of requirements. These generic communications present a chronology of events and actions requested by the NRC (actions for plant licensees to take to preclude or mitigate events that could affect the nuclear power plant during low-power and shutdown operations) that have resulted in changes to regulatory requirements.

Two generic letters (87-12 and 88-17) are of interest to low-power and shutdown operations. They contain actions requested of licensees or identify actions taken by licensees that other licensees were asked to review for applicability to their facilities. They are the most comprehensive and most widely applicable of the generic letters. They most specifically address shutdown concerns and are the most current generic letters to contain recommendations regarding low-power and shutdown operations.

Table 5.1 lists eight generic letters related to shutdown and low-power operations and Table 5.2 lists the requirements and recommendations of Generic Letter 88-17.

5.2 International Facilities

In January of 1991 a questionnaire was sent to the regulatory agencies of several nations including the Committee on Nuclear Regulatory Activities (CNRA) member nations. This questionnaire, "Elements for a Survey on Low-Power and Shutdown Activities," was intended to gather information regarding approaches to the control of low-power and shutdown operations at nuclear power plants. The objective of the questionnaire was that the responses would address all low-power and shutdown requirements, both of the regulatory authority and of the facility operators. However, most responses addressed the regulatory requirements and simply acknowledged that operation during these modes was mainly controlled by procedures and requirements established by the facility operator.

In particular, the responses were to address requirements for reactivity control, inventory control, residual heat removal, containment integrity, and outage and maintenance management. Each country indicated that its regulatory body has established safety requirements that the operator was required to meet. However, the specific operating requirements were developed by the plant operator.

Technical specifications or their equivalent appeared to be the principal technique used to impose regulatory control of plant operation during shutdown and low-power operation.

These requirements were generally less restrictive in the shutdown mode than in the full-power operations mode. Low-power operation was often approached with the same requirements as full-power operation, although in specific instances the technical specifications requirements during low power were relaxed from the full-power requirements.

Table 5.1

Generic Communication--Generic Letters

Generic Letter	Title
80-42	Decay Heat Removal Capability
80-53	Transmittal of Revised Technical Specifications for Decay Heat Removal Systems at PWRs
81-21	Natural Circulation Cooldown
85-05	Inadvertent Boron Dilution Events
86-09	Technical Resolution of Generic Issue B-59, (n-1) Loop Operation in BWRs and PWRs
87-12	Loss of Residual Heat Removal (RHR) While the Reactor Coolant System (RCS) Is Partially Filled
88-17	Loss of Decay Heat Removal
90-06	Resolution of Generic Issue 70, 'Power-Operated Relief Valve and Block Valve Reliability,' and Generic Issues 94 'Additional Low-Temperature Overpressure Protection for Light Water Reactors' pursuant to 10 CFR 50.54 (f)

Table 5.2

Generic Letter 88-17* Requirements and Recommendations

Item	Requirement/Recommendation**
(1)	Discuss the Diablo Canyon event, related events, lessons learned, and implications with the appropriate plant personnel. Provide training shortly before entering a reduced inventory condition.
(2)	Implement procedures and administrative controls that reasonably ensure that containment closure will be achieved before the time at which a core uncovery could result from a loss of decay heat removal coupled with an inability to initiate alternate cooling or addition of water to the reactor coolant system.
(3)	Provide at least two independent, continuous temperature indications that are representative of the core exit conditions whenever the reactor is in midloop operation and the reactor vessel head is located on top of the vessel.

* This generic letter discussed the loss of decay heat removal capability that occurred on April 10, 1987, at Diablo Canyon Unit 2 while the plant was in the refueling mode of operation. Additional events at Waterford (on May 12, 1988), Sequoyah (on May 23, 1988), and San Onofre (On July 7, 1988) also contributed to this second generic letter addressing loss of decay heat removal capabilities at PWRs.

** Recommended for implementation before operating in a reduced inventory condition.

Table 5.2 (Continued)

Item	Requirement/Recommendation**

(4) Provide at least two independent, continuous reactor coolant system water level indications whenever the reactor coolant system is in a reduced inventory condition.

(5) Implement procedures and administrative controls that generally avoid operations that deliberately or knowingly lead to perturbations to the reactor coolant system or to systems that are necessary to maintain the reactor coolant system in a stable and controlled condition while the reactor coolant system is in a reduced inventory condition.

(6) Provide at least two available or operable means of adding inventory to the reactor coolant system in addition to the pumps that are a part of the normal decay heat removal systems.

(7) Implement procedures and administrative controls that reasonably ensure that all hot legs are not blocked simultaneously by nozzle dams unless a vent path is provided that is large enough to prevent pressurization of the upper plenum of the reactor vessel.

(8) Implement procedures and administrative controls that reasonably ensure that all hot legs are not blocked simultaneously by closed loop stop valves unless reactor vessel pressurization can be prevented or mitigated.

Table 5.2 (Continued)

Item	Program Enhancement+
(1)	Provide reliable indication of parameters that describe the state of the reactor coolant system and the performance of systems normally used to cool the reactor coolant system for both normal and accident conditions. The following should be provided in the control room: two independent indications of reactor vessel level and temperature, indications of decay heat removal system performance, and visible and audible indications of abnormal conditions.
(2)	Develop and implement procedures that cover reduced inventory operation and that provide an adequate basis for entry into a reduced inventory condition.
(3)	Ensure that adequate operating, operable, or available equipment is provided for cooling the reactor coolant system. Maintain existing equipment in an operable or available status, including at least one high-pressure system and one other system. Provide adequate equipment for personnel communications.
(4)	Conduct analysis to supplement existing information and develop a basis for procedures, instrumentation installation and response, and equipment/NSSS interactions and response.
(5)	Technical specifications that restrict or limit the safety benefit of these actions should be identified and appropriate changes should be submitted.
(6)	Item 5 (of the first 8 items of this table) should be reexamined and refined as needed.

+ The NRC requested that these program enhancements be implemented as soon as was practical.

Of the areas addressed in the questionnaire, the outage and maintenance management area appeared to be the most within control of the operators of the nuclear facility. General requirements to submit outage plans and refueling documentation were the most restrictive of the requirements imposed by any country, and most appeared to require some type of planning. In the other areas addressed by the questionnaire, some control over the plant configuration was exercised in the technical specifications (or their equivalent) in most countries.

Reactivity control requirements for PWRs tend to address two related items: boron concentration (including both boron injection system operability and the need to isolate the primary system from sources of non-borated water) and subcriticality margin. Additional requirements mentioned in many responses included requirements to maintain neutron flux monitoring instrumentation operable in all modes, unless the control rods cannot be moved.

Generally, fewer reactivity control requirements were imposed on the BWRs than PWRs. During refueling operations, restrictions were generally in place regarding the removal of control assemblies from the core. Either one rod at a time was allowed to be removed or the supercell around the control rod to be removed must be empty.

Several different approaches were taken to describe the inventory control requirements. Some countries described the instrumentation requirements for the shutdown and low-power operational modes. For these countries, additional instrumentation was required at various times during operation in these modes, particularly during PWR midloop operations.

The responses of several countries described injection capability requirements. Combinations of low- and high-pressure injection systems were required to be operable. Often, during the time that the refueling cavity was flooded, the injection system requirements were reduced. However, if maintenance was being performed on the primary system below the level of the core, this reduction in injection availability was not allowed.

In general, redundant heat removal capabilities were required at all times by most of the countries. In PWRs, this redundancy could often be supplied by any combination of operable steam generators and residual heat removal systems, shifting entirely to the residual heat removal systems once the steam generators cannot be used. For those countries that replied in detail, their responses indicated that the flooded refueling cavity can be considered a heat removal system, due to the large amount of water present. There are at least two countries that tied the operability of the residual heat removal system to the decay heat rate as a function of time after shutdown. For these countries, the requirements on system operability were reduced as the decay heat rate dropped.

In general, containment integrity requirements were waived under certain conditions in every country. Usually, during the refueling mode of operation when no fuel transfer was taking place, containment integrity was not required. Containment airlocks were not always required to remain operable during refueling. When they were allowed to be open during refueling, they must generally isolate on a high radiation signal. In BWRs with inerted containments, the containment generally may be de-inerted several hours before entering a cold-shutdown condition and did not have to be re-inerted until after entering hot-shutdown conditions.

Other than some staffing requirements, there were almost no regulatory requirements that specifically addressed outage and maintenance management. Many countries did require that outage and refueling plans be submitted to the regulatory bodies. These documents must outline the procedures and rule to be followed during an outage. However the procedures and rules were generally developed by the licensee.

It appears that significant variability exists among the programs in various countries and that the NRC's current requirements in the areas of shutdown and low-power operations were less stringent than those of most other regulatory agencies. However, the staff concludes that the NRC's continuing shutdown risk study appears to address all the significant issues.

6 TECHNICAL FINDINGS AND CONCLUSIONS

6.1 Overview

On the basis of the work it completed over the past 18 months, the staff concludes that risk varies widely during shutdown conditions at a given plant and among plants, and can be significant. The staff has observed an increasing recognition of the importance of shutdown issues among licensees and within the staff. The staff also observed a general improvement in safety practices during shutdown both as a result of regulatory actions and from the industries own individual and collective initiatives.

Variability of risk during an outage period results primarily from continuous changes in (1) plant configuration and activity level, which determine the likelihood of an upset and, to some degree, the severity; (2) the amount and quality of equipment available to recover from an upset; (3) the time available to diagnose and recover from an upset; and (4) the status of the containment. Variability among plants resulted from the many approaches used by utilities to address safety during a shutdown condition, differences in plant design features, and lack of a standard set of industry or regulatory controls for shutdown operations. Such variability, along with analytical limitations peculiar to shutdown (e.g., human reliability analysis), makes it difficult to quantify the risk during shutdown in U.S. reactors. The staff has focused its attention primarily on operating experience and the current capability in U.S. plants to avoid a core-melt accident and release of radioactivity. Insights from probabilistic assessments have also been valuable in understanding what is important to risk during shutdown.

As discussed in Chapter 1, about midway through the evaluation the staff identified a number of issues believed to be especially important, and a number of potential important issues. The staff has studied each of these issues and obtained specific findings which are discussed in this chapter.

6.2 Outage Planning and Control

In the absence of strict technical specification controls, licensees have considerable freedom in planning outage activities. Outage planning determines what equipment will be available and when. It determines what maintenance activities will be undertaken and when. It effectively establishes if and when a licensee will enter circumstances likely to challenge safety functions and it establishes the level of mitigation equipment available to deal with such a challenge.

Many shutdown events have occurred that represented challenges to safety during low-power/shutdown (LPS) operation. Some of these initiated when the power plant was in a sensitive condition as a result of inadequate planning and mistakes (examples: Diablo Canyon, 4/87, NUREG-1269; Vogtle, 3/90, NUREG-1410). Recognizing that the safety significance of such events is a strong function of outage planning and control; and that the NRC has not previously addressed the safety implications of outage planning, the staff initiated a study of such planning and its implications as part of the plant visits program described in Chapter 3, and has supplemented this with information from staff inspectors.

A wide variety of conditions and planning approaches was observed during the plant visits. These included:

(1) outages that were well planned and controlled

(2) outages that were poorly prepared and unorganized

(3) priority assigned to safety with the complete licensee organization striving for safety

(4) an ad hoc approach in which safety was dependent upon individual judgment

(5) the perception that short outages represent excellence

(6) personnel stress and events that appeared to be the result of overemphasis on achieving a short outage

(7) impact on plant operation from poor outage planning and implementation

(8) imprudent operation as a result of insufficient attention to safety

6.2.1 Industry Actions

The industry has addressed outage planning and control with programs that include workshops, Institute of Nuclear Power Operations (INPO) inspections, Electric Power Research Institute (EPRI) support, training, procedures, and other programs. One activity (a formal NUMARC initiative) has led to a set of guidelines for utility self-assessment of shutdown operations (NUMARC 91-06); these guidelines serve as the basis for an industrywide program that will be implemented at all plants by December 1992. This provides high-level guidance that addresses many outage weaknesses. Detailed guidance on developing an outage planning program is beyond the scope of the NUMARC effort.

NUMARC 91-06 states: "The underlying premise of this guidance is 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."

The staff met with NUMARC and the associated utility working group on several occasions to share technical insights and discuss program status. The initiative does appear to be a significant and constructive step and effects may have already been realized by a few utilities using draft guidance in a recent outage.

6.2.2 NRC Staff Findings

Based on the staff review of operating experience, PRAs, site visits and information from other regulatory agencies, the staff concludes that a well-planned, well-reviewed, and well-implemented outage is a major contributor to safety. It has further substantiated and/or determined that

(1) Consistent industrywide safety criteria for the conduct of LPS operation do not exist. (NUMARC 91-06 provides high-level guidance, but no criteria.)

(2) Many licensees have no written policy that provides safety criteria for LPS operation. Some are working on such a policy, whereas, at the time the staff visited the plant, others had no plan to prepare such a policy.

(3) Some licensees enter planned outages with incomplete outage plans.

(4) Some licensees cannot properly respond to an unscheduled outage because their planning is poor.

(5) Safety considerations are not always evident during outage planning.

(6) Changes to outage plans and ad hoc strategies for activities not addressed in the plan are often not addressed as carefully as the original plan.

(7) The need for training and procedures is not always well addressed in planning.

(8) Bases do not exist that fully establish an understanding of plant behavior and that substantiate the techniques depended upon to respond to events. Such bases would provide the information necessary for reasonable and practical technical specifications, procedures, training, LPS operation (outage) planning, and related topics.

(9) There is no regulation, regulatory basis, staff policy, or other guidance (such as technical specifications or staff studies) that currently requires or otherwise provides regulatory guidance for outage planning and plan implementation.

6.3 Stress on Personnel and Programs

A large amount of activity takes place during outages. The increased size of the work force at the site during outages, combined with the rapid changes in plant configurations that occur during these periods, creates a complex environment for planning, coordinating, and implementing tasks and emergency responses. As a result, outage activities can stress the capabilities of plant personnel and programs responsible for maintaining quality and operational safety. This stress can be reduced through outage planning that ensures (1) staffing levels are sufficient and jobs are defined so that workloads during normal or emergency outage operations do not exceed the capabilities of plant personnel or programs, (2) personnel are adequately trained to perform their duties including the implementation of contingency plans, and (3) contingency plans are developed for the mitigation of the consequences of events during shutdown.

The present NRC policy concerning working hours of nuclear plant staff, as written, provides objectives for controlling the working hours of plant personnel, and provides specific guidelines for periods when a plant is shut down. It permits plant personnel to work overtime hours in excess of the recommended hours, provided that appropriate plant management gives its approval. However, as noted in NRC Information Notice 91-36, in some instances a licensee's work-scheduling practices or policies were inconsistent with the intent of the NRC policy.

The staff reviewed the NUMARC document "Guidelines to Enhance Safety During Shutdown" and concludes that the guidelines establish a sound approach to addressing the issue of stress and its risks associated with low-power and shutdown operations. Effective implementation of these guidelines should reduce the potential for planned or unplanned outage activities to inappropriately stress the capabilities of plant personnel and programs by (1) improving control of outage activities, (2) reducing time that people performing higher risk activities, and (3) increasing preparedness to implement contingency actions, if needed. Consequently, stress on plant programs and personnel during outages is expected to be minimized.

6.4 Operator Training

Conditions and plant configurations during shutdown for refueling can place control room operators in an unfamiliar situation. Operators who are properly informed and who understand the problems that could arise during outages are essential in reducing risks associated with the outage activities. Through the comprehensive training programs, operators can gain such knowledge and understanding, thus increasing the level of safe operations at nuclear plants. The level of knowledges and abilities can be qualitatively measured by a comprehensive examination.

6.4.1 Examination on Reactor Operators

The knowledges and abilities (K/A) that an operator needs to properly mitigate the events and conditions described in Chapters 2 and 3 are addressed by NRC's K/A catalogs (NUREG-1122 and NUREG-1123). These catalogs, in conjunction with the facility licensee's job task analysis (JTA), provide the basis for the development of examinations that contain valid content. Present guidance for developing examinations is described in the Examiner Standards (NUREG-1021). This guidance allows for significant coverage of shutdown operations, but it does not specify any minimum coverage. NUREG-1021 provides a methodology for developing examinations that was derived, in part, from data collected from licensed senior reactor operators (SROs) and NRC examiners. The guidance also calls for examination content to include questions and actions based on operating events at the specific facility and other similar plants. Generally, a review of samples of initial written examinations indicates that low-power/shutdown operations are covered generally and the coverage is consistent with assuring adherence to the objectives of licensee training programs and the sampling methodology of NUREG-1021. However, if licensee training programs and procedures are revised, through an improved outage program, to place more emphasis on reducing shutdown risks, the staff expects that more extensive and broader examination coverage will result.

6.4.2 Training on Simulators

As of May 26, 1991, all facility licensees were required to have certified or approved simulation facilities unless specifically exempted. Nearly all of the industry's simulators have been certified to meet the guidance of the American National Standard, "Nuclear Power Plant Simulators for Use in Operator Training," ANSI/ANS-3.5-1985, as endorsed by Regulatory Guide 1.149 (RG 1.149). This standard calls for simulation of minimum normal activities from cold startup to full power to cold shutdown, excluding operations with the reactor vessel head removed. Therefore, these certified simulators are capable of performing many of the operations from a subcritical state to synchronization with the electrical grid.

ANSI/ANS-3.5-1985 is based on the concept that the scope of simulation should be commensurate with operator training needs. In accordance with ANSI/ANS-3.5-1985, the scope of simulation should be based on a systematic process for designing performance-based operator training, and modifications should be based on assessments of the training value this process offers. The scope of the necessary changes would be defined by operator tasks identified as requiring training or examination on a simulator. Presently, simulators are used in training and examinations in those areas where dynamic plant response provides the most appropriate means to meet the training objectives. Many events that are likely to occur during shutdown would result in the majority of operator actions taking place out in the plant rather than in the control room. As a result, such events might be more appropriately addressed through methods other than simulator training.

To the extent practicable, simulator training for shutdown conditions should continue to be conducted. The Examiner Standards document (NUREG-1021) already requires examiners to report observations of simulator performance in the examination reports. This feedback from the examiners is then used to determine if simulator inspections are necessary. Revising NUREG-1021 to place more emphasis on reducing shutdown risks, should result in more observations of simulator performance in this area being reported than are identified at present.

6.5 Technical Specifications

6.5.1 Residual Heat Removal Technical Specifications

Based primarily on the PRA studies discussed in chapter 4 and the thermal-hydraulic analysis in Section 6.6, the staff has concluded that current PWR standard technical specifications are not detailed enough to address the number and risk significance of reactor coolant system configurations used during cold shutdown and refueling operations. This is particularly true of PWR technical specifications. Safety margin during these modes of operation is significantly influenced by the time it takes to uncover the core following an extended loss of RHR. The conditions which affect this margin significantly include: decay heat level, initial reactor vessel water level, the status of the reactor vessel head, (i.e bolted on, on with bolts detensioned or removed), the number and size of openings in the cold legs, the existence of hot leg vents, whether or not there are temporary seals in the RCS which could leak if the system is pressurized and availability of a diverse, alternate method of residual heat removal in case of complete loss of RHR. The current technical specifications do not reflect these observations. The staff has also found that some older plants do not even have basic technical specifications covering the residual heat removal system.

In light of the above findings, the staff has identified a number of proposed improvements to Limiting Conditions for Operation in current standard technical specifications for the RHR systems, component cooling water systems, service water systems and emergency core cooling systems. These improvements are discussed in Chapter 7.

6.5.2 Electrical Power Systems Technical Specifications

Electric power and its distribution system is generally as vital for accident mitigation during shutdown conditions as it is for power operating conditions. There are, however, some shutdown conditions for which it is not as vital and during which, losses of power can be accommodated more easily (e.g fuel off-load and reactor cavity flooded). In PWRs, all normal residual heat removal systems and most components used in alternate methods are powered electrically. The same holds true for ECCS and instrumentation. BWRs are similar, but there are many more systems available to remove heat that are powered by steam; however, these systems can only be used when the reactor vessel head is on and the main steam system is pressurized. Electric power is also vital for securing containment integrity.

Current STS were written under the assumption that all shutdown conditions were of lessor risk than power operating conditions. This has resulted in most maintenance of electrical systems being done during shutdown. Consequently, requirements for operability of systems are relaxed during shutdown modes.

Operating experience and risk assessments discussed in chapters 2 and 3 indicate that there are some shutdown conditions (e.g. midloop operation) where such relaxation of operability requirements for electrical system is not justified. In addition, past STS in the electrical system area have been poorly integrated with technical specifications for other systems that the electrical systems must support. As a result, many plant specific technical specifications for shutdown conditions are also poorly integrated; and misunderstandings have occurred regarding how the electrical specifications should be applied to support other technical specifications for systems such as RHR systems. There are also some facilities that do not have any electrical system technical specifications for shutdown modes.

In light of the above findings and knowledge of shutdown operations gained from the site visits, the staff has concluded at this time that with proper planning, maintenance of electrical systems can be accommodated during shutdown conditions of lessor risk significance. Consequently, the staff has developed proposed improvements to technical specifications for electrical systems which: (1) ensure a minimum level of electrical system availability in all plants; (2) balance the need for higher availability of electrical systems during some shutdown conditions and the need to still do maintenance during shutdown operations; (3) bring logic and consistency to an area of nuclear plant operation that has been cumbersome for both plant operators and regulators. These improvements are discussed in Chapter 7.

6.5.3 PWR Containment Technical Specifications

As discussed in chapter 5 containment integrity for neither PWRs or BWRs is required by technical specification during cold shutdown or refueling conditions except during movement of fuel. The staff has concluded based on operating experience, thermal-hydraulic analyses and PRA assessments that ensuring PWR containment integrity prior to an interruption in core cooling under some shutdown conditions may be necessary (this is discussed more fully in Section 6.8.1). Changes to the technical specification on containment integrity would be the most direct and effective means of improving containment capability where needed. Consequently, the staff is considering the need for ne technical specifications to govern containment integrity for PWRs during some shutdown conditions, as discussed in chapter 7.

6.6 Residual Heat Removal Capability

6.6.1 Pressurized-Water Reactors

Decay heat is removed in PWRs during startup and shutdown by dumping steam to the main condenser or to the atmosphere and restoring inventory in the steam generators with the auxiliary feedwater (AFW) system. During cold shutdown and refueling, the residual heat removal system is used to remove decay heat. Because of the relatively high reliability of the AFW system and the short time spent in the startup and shutdown transition modes, losses of decay heat removal during these modes have been infrequent. However, loss of decay heat removal during shutdown and refueling has been a continuing problem. In 1980, a loss of residual heat removal event occurred at the Davis-Besse plant when one RHR pump failed and the second pump was out of service. Following a review of the event and the requirements that existed at the time, the NRC issued Bulletin 80-42 and later a Generic Letter 80-43 calling for new technical specifications to assure that one RHR system is operating and a second is available (i.e., operable) for most shutdown conditions. The Diablo Canyon event of April 10, 1987, highlighted the fact that midloop operation was a particularly sensitive condition. Following its review of the event, the staff issued Generic Letter 88-17, requesting that licensees address numerous generic deficiencies to improve the reliability of decay heat removal capability. More recently, the incident investigation team's report of the loss of ac power at the Vogtle plant (NUREG-1410) raised the issue of coping with a loss of RHR during an extended period without any ac power. In light of the continued occurrence of events involving loss of RHR and the issues raised in NUREG-1410, the staff has performed assessments of the effectiveness of GL 88-17 actions and alternate methods of decay heat removal. These assessments are discussed next.

6.6.1.1 Effectiveness of GL 88-17 Actions

Actions requested in GL 88-17 are listed in Table 5.2. The staff assessed the response to GL 88-17 through NRC inspections conducted to date and the site visits discussed in Chapter 2. The more important subject areas have been evaluated in terms of overall performance since issuance of GL 88-17, as discussed below.

Operations

Operations with the RCS water level at midloop have diminished generally. Some utilities now perform activities requiring reduced inventory with the reactor defueled. Others have taken steps to minimize time spent in reduced inventory or plan sensitive activities later in the outage when the decay heat level is lower. However, midloop operation is still used widely; in fact, one utility stayed at midloop for 37 days in its most recent outage.

Events

Loss-of-RHR events have continued to occur even 2 years after the issuance of GL 88-17. Three events discussed in Chapter 2 occurred in 1991. All three occurred at sites that had also experienced such events before GL 88-17 was issued.

Procedures

As discussed in Chapter 2, procedures for responding to loss-of-RHR events have generally improved in terms of the level of information provided to operators and the specification of alternate systems and methods that can be used for recovery. In addition, inspection teams have found that procedures written in response to GL 88-17 have been applied effectively outside the intended envelope for lack of other procedures, for example, loss of inventory.

However, some concerns still exist. Although procedures often specify use of the steam generators or the ECCS as alternate methods for removing decay heat, it has been observed, as discussed in Chapter 3, that neither steam generator availability nor a clear flow path via the containment sump has been planned for and maintained. In addition, it has also been observed that complete thermal-hydraulic analyses and bases have not been developed which would ensure that operators have been given the necessary information to respond to a complicated event involving steam generation in the RCS, including one following a station blackout. A number of important considerations relating to alternate decay heat removal have not been observed in training literature nor plant procedures. These are discussed in Section 6.6.1.2.

Instrumentation

Licensees have responded appropriately to GL 88-17 by providing two independent RCS level indications, two independent measurements of core exit temperature, the capability to continuously monitoring RHR system performance, and visible and audible alarms. However, wide variability exists among sites in the quality of installations and controls for using them, as discussed below:

(1) Many operators were unaware [illegible] core temperature cannot be inferred from measurements in the RHR sys[illegible]n the RHR pumps are not running, and sometimes core exit thermoc[illegible] have not been kept operable even though the vessel head was installed.

(2) Potential problems associated with water level indications have been observed, including damaged and/or incorrectly installed instrument tubing, lack of independence, and poor maintenance.

6.6.1.2 Alternate Residual Heat Removal Methods

In response to the incident investigation team's report of the loss of ac power at the Vogtle plant (NUREG-1410), the staff, with the assistance of the Idaho National Engineering Laboratory, has conducted in-depth studies of passive, alternate methods of residual heat removal that could potentially be used when the RHR system is unavailable. The initial study (NUREG/CR-5820) identified fundamental passive cooling mechanisms that could be viable for responding to an extended loss of RHR and evaluated plant conditions and procedural actions that could be used to exploit those mechanisms, as well as problems in such exploitation. The important cooling processes include gravity drain of water from the RWST into the RCS, core water boiloff, and reflux cooling. A second study (Appendix B) examined the transient response of a PWR with U-tube steam generators following a loss-of-RHR event using the RELAP5/MOD3 reactor analysis code with a model modified for reduced inventory conditions. The significant findings from these studies are discussed below.

<u>Gravity Drain From the Refueling Water Storage Tank</u>

Most, but not all, PWRs are theoretically capable of establishing a drain path between the refueling water storage tank (RWST) and the RCS. However, the relative elevation difference between the RWST and the RCS, which determines how much water is available, can vary significantly from plant to plant. Under ideal conditions for a spectrum of plants studied, RWST feed-and-bleed of the RCS could maintain flow to the vessel and remove decay heat for as little as 0.4 hour for one plant to as much as 18 hours for another, assuming the loss of RHR occurred 2 days after shutdown; for unthrottled flow, the times are 0.2 hour and 5.2 hours.

Gravity Feed From Accumulators or Core Flood Tanks

The limited liquid contents and difficulty in throttling flow from accumulators or core flood tanks makes their use of marginal value in terms of long-term core cooling.

Reflux Cooling

Initiation of reflux condensation cooling depends on the ability of steam, produced by core boiling, to reach condensing surfaces in the steam generator U-tubes. During a plant shutdown condition, the reactor coolant level may be at reduced inventory with air or nitrogen occupying the upper volumes of the primary system. This air inhibits steam flow from the reactor vessel to the steam generator U-tubes. Important aspects of reflux initiation are (1) the initial reactor coolant water level, (2) the need to establish and preserve horizontal stratification of the liquid in the hot legs, (3) the primary system pressure needed to establish a sufficient condensing surface, and (4) the possible need for draining or venting the primary system in order to obtain a stable reflux cooling mode at an acceptable pressure.

The ability to remove decay heat through one steam generator by reflux condensation following a loss-of-RHR event during reduced inventory operation represents an alternative way to remove decay heat, one that does not require adding water to keep the core covered with a two-phase mixture. In many instances, nozzle dams are installed in the hot- and cold-leg penetrations to one or more steam generators, and the reactor vessel head is installed with air in the unfilled portion of the RCS above the water level. Should the RHR system fail, the peak pressure and temperature reached in the reactor coolant system are important since the nozzle dams must be able to withstand these conditions to prevent a loss-of-coolant accident. Failure of a hot-leg nozzle dam would create a direct path to the containment through an open steam generator manway. Such an event could also result in peak RCS pressures sufficient to cause failure of the temporary thimble seals used to isolate the instrument tubes. These thimble seals are used during plant outages while instrument maintenance is performed.

Analyses were performed in the Appendix B study to identify the time to core uncovery due to the blowout failure of the hot-leg nozzle dam with the manway removed from the steam generator inlet plenum. Nozzle dam blowout was assumed to occur at 25 psi. The actual failure pressure is not well known and likely varies among different designs. An analysis was also performed to determine the time to core uncovery if all of the temporary thimble seals failed.

The results of the analyses are as follows:

° Analyses of the loss of the RHR system from midloop operation at 1 day and 7 days following shutdown reveal that the RCS can reach peak pressures in the range 25 psi when a single U-tube steam generator is used for residual removal. Moreover, RCS peak pressure is insensitive to decay heat level or the time of loss of RHR system following shutdown.

° Additional analyses of the use of U-tube steam generators for residual removal show that RCS peak pressures approach 80 psi with initial RCS water levels above the top elevation of the hot leg. At these higher water levels, the fluid expansion fills the steam generator tubes with sufficient liquid to prevent residual removal until pressures reach 80 psi or until sufficient primary to secondary temperature difference is established. Peak RCS pressure is, therefore, sensitive to the initial liquid level at the time the RHR system is lost.

° Since RCS pressures near the design conditions for nozzle dams and temporary thimble seals can be attained, the successful use of the steam generators as an alternative residual removal mechanism is not assured. The loss of the RHR system with initial RCS water levels above the top of the hot leg suggests use of the steam generators as an alternate means of decay heat removal will result in sufficient pressure to challenge the integrity of all of the RCS temporary boundaries.

° Analyses of the failure of the RCS temporary boundaries (i.e., nozzle dams, thimble seals, etc.) or openings such as the safety injection line demonstrate that if the RHR system fails within the first 7 days following shutdown, there is very little time (i.e., about 30 to 90 minutes) to prevent core uncovery and isolate the containment. Failure of many thimble tube seals would produce significant leakage, but much less than a nozzle dam failure.

6.6.2 Boiling-Water Reactors

During a normal shutdown, initial cooling is accomplished by using the main turbine bypass system to direct steam to the main condenser, and by using the condensate and feedwater systems to return the coolant to the reactor vessel. The circulating water system completes the heat transfer path to the ultimate heat sink. This essentially is the same heat transport path as is used during power operation except that the main turbine is tripped and bypassed and the steam, condensate, and feedwater systems are operating at a greatly reduced flow rate. When the steam and power conversion system is not available, high-pressure shutdown cooling is provided by isolation condensers (early BWRs) or by the reactor core isolation cooling (RCIC) system (later BWRs). No BWRs have both isolation condensers and an RCIC system.

The RHR system provides for post-shutdown core cooling of the RCS after an initial cooldown and depressurization to about 125 psig by the steam and power conversion system, the isolation condensers, or the RCIC system. Early BWRs have dedicated RHR systems that are separate from the low-pressure ECCS subsystems. Later BWRs have multi-mode RHR systems that perform the shutdown cooling function as well as a variety of ECCS and containment cooling functions. The RHR shutdown cooling suction line is opened to align the suction of the RHR pumps to a reactor recirculation loop on the suction side of an idle recirculation pump. Flow is established through the RHR heat exchangers and the primary coolant is then returned to the reactor vessel via a recirculation line (on the discharge of an idle recirculation pump) or a main feedwater line (later model BWRs only). The RHR heat exchangers transfer heat to the RHR service water system. The RHR service water system is a single phase, moderate-pressure system that is dedicated to providing cooling water for the RHR heat exchangers. In later BWRs (BWR/5s and BWR/6s), RHR cooling is supplied by an essential service water system that also provides cooling for other safety-related components. In either case, the service water systems may operate on an open, closed, or combined cycle. The service water and the circulating w... systems may operate on different cooling cycles (i.e., a closed-cycle service water system and an open-cycle circulating water system).

Due to the relatively high discharge pressure of the RHR service water pumps (about 300 psid), the service water system can be used in an emergency to flood the BWR core or the containment. This capability is implemented by opening the cross-tie between the service water system and the RHR return line to the RCS. In a multi-mode RHR system, this return line branches to the reactor vessel, the suppression pool, and the drywell.

Loss of Residual Heat Removal Capability

As indicated in Chapter 2, the frequency and significance of precursor events involving reduction in reactor vessel (RV) water level and/or loss of RHR in BWRs have been less than for PWRs. One reason for this is that BWRs do not enter a reduced inventory or midloop operating condition as do PWRs. Another reason is that a reduction in RV water level will normally be terminated by the reactor protection system before the level falls below the suction of the RHR pumps.

Should RHR be lost, operators can usually signifi...tly extend the time available for recovery of the system by adding water to the core from several sources, including condensate system, low-pressure coolant injection (LPCI) system, core spray (CS) system, and control rod drive (CRD) system. Adding inventory raises water to a level that can support natural circulation. In the event that RHR cannot be recovered in the short term, alternate residual heat removal methods, covered by proc... res, are normally available. If the RV head is tensioned, the reactor pressure vessel (RPV) is first allowed to pressurize and then steam is dumped to the suppression pool via a safety-relief valve (SRV) and makeup is provided by one of the water sources listed above. If the condenser and condensate system are available, then decay heat can be removed by dumping steam to the condenser and adding makeup water from the condensate and feedwater system. If the vessel head is detensioned then decay heat must be removed without the RPV pressurized. This requires opening multiple SRVs to dump steam to the suppression pool and cooling the suppression pool by recirculating water using the CS or LPCI pumps. For all cooling methods involving the suppression pool, suppression pool cooling must be initiated in sufficient time to prevent suppression pool temperature from becoming so high that the pumps lose net positive suction head. If the RPV head is removed and the main steamline plugs are put in place, the preferred method of residual heat removal is to flood the reactor cavity and place the fuel pool cooling system in operation. A second undesirable, but nevertheless acceptable alternative is to boil off steam to the secondary containment and add makeup water from any source capable of injecting water at a rate of a few hundred gallons per minute. As discussed in Section 6.9.1, this method of residual heat removal can lead to failure of the secondary containment.

The results of the accident sequence precursor analysis discussed in Chapter 2 indicate that the frequency and severity of loss of RHR incidents in BWRs are less than for PWRs. In addition, the review of BWR alternate residual heat removal methods indicates significant depth and diversity. For these reasons, the staff concludes that loss of RHR in BWRs during shutdown is not a significant safety issue as long as the equipment (pumps, valves and instrumentation) needed for these methods is operable and clear procedures exist for applying the methods.

6.7 Temporary Reactor Coolant System Boundaries

In the course of the evaluation, the staff has identified and examined plant configurations used during shutdown operations involving temporary seals in the reactor coolant system. This includes freeze seals that are used in a variety of ways to isolate fluid systems temporarily, temporary plugs for nuclear instrument housings, and nozzle dams in PWRs. The staff has identified instances where failure of these seals, either due to poor installation or an overpressure condition, can lead to a rapid non-isolable loss of reactor coolant. This concern is of special importance in PWRs because the emergency core cooling system (ECCS) is not designed to automatically mitigate accidents initiated at pressures below a few hundred psig and is not normally fully available for manual use during these conditions. In BWRs, the ECCS is normally required to be operable when there is fuel in the reactor vessel and activities are taking place that have the potential to drain the reactor vessel. In addition, the ECCS is actuated automatically when water level is low in the reactor vessel.

6.7.1 Freeze Seals

Freeze seals are used for repairing and replacing such components as valves, pipe fittings, pipe stops, and pipe connections when it is impossible to isolate the area of repair any other way. Freeze seals have successfully been used in pipes as large as 28 inches in diameter. However, as a result of inadequate use and control, some freeze seals have failed in the nuclear power plants, and some of the failures have resulted in significant events. This has raised a question regarding the adequacy of 10 CFR 50.59 safety evaluations of freeze seal applications.

To assess problems associated with freeze seals, the staff reviewed the operational experience on freeze seal failures, safety-significant findings on freeze seal failures, industry reports on freeze seal use and installation, and the applicability of industry guidance (NSAC-125) for performing safety evaluations on freeze seal applications.

6.7.1.1 Operational Experience on Freeze Seal Failures

° <u>River Bend, 1989</u>

Failure occurred in a freeze plug used in a 6-inch service water line to allow inspection and repair work on manual isolation valves to a safety-related auxiliary building cooler. The failure resulted in a spill of approximately 15,000 gallons of service water into the auxiliary building and caused the loss of non-safety-related electrical cabinets (i.e., shorting and an electrical fireball damaged cabinets and components). Draining water also tripped open a 13.8 kV supply breaker, resulting in loss of the residual heat removal system, spent fuel pool cooling system, and normal lighting in the auxiliary and reactor buildings. The leak was isolated in 15 minutes and the RHR system restarted in 17 minutes.

° Oconee 1, 1987

Approximately 30,000 gallons of slightly radioactive water leaked into various areas of the auxiliary building and a portion drained beyond the site boundary due to failure of a freeze plug which was used to facilitate replacement of a 3-inch-diameter section of low-pressure injection piping.

° Brunswick 1, 1986

Failure of a freeze seal used in the discharge piping of the control rod drive system pump 1A caused hydraulic perturbation to a high-level/turbine trip instrument, resulting in a feed pump trip and subsequent automatic scram at 100-percent power.

The freeze seal failure at River Bend prompted a visit by an NRC augmented inspection team (AIT) to perform an onsite inspection shortly after the event. The AIT found:

(1) inadequate control of freeze seal work

(2) lack of training for personnel performing the work

(3) lack of awareness by plant personnel of the potential for freeze seal failure

(4) flooding that exceeded the design capacity of the floor drain system

(5) no damage to safety-related equipment

A 10 CFR 50.59 safety evaluation of the freeze seal operation was not performed. The plant operating procedure was subsequently revised to include corrective measures for freeze seal installation and control. However, the licensee included no statement to assure or require that a 10 CFR 50.59 safety evaluation be performed before allowing use of a freeze seal.

In regards to the incident that occurred at Oconee Station, Unit 1, in 1987, the utility was cited by the NRC for inadequate freeze seal procedures. A review of the licensee's freeze seal "safety evaluation checklist" found that the checklist questions were similar to 10 CFR 50.59 questions. However, the checklist was not processed through the licensee's safety committee, as would have been done for a formal 10 CFR 50.59 safety evaluation.

Information Notice 91-41, "Potential Problems With the Use of Freeze Seals," identified potential problems related to the freeze seal in PWRs and BWRs, specifically including both the River Bend and Oconee 1 incidents. The information notice indicated that freeze seal failure in a PWR reactor boundary system could result in immediate loss of primary coolant. In BWRs, failure of a freeze seal in a system connected to the vessel's lower plenum region, such as the reactor water cleanup (RWCU) system, could result in the water level in the reactor vessel falling below the top of the active fuel. The estimated time for this to occur is less than 1 hour if the seal failed completely and makeup water was not added to the reactor. The information notice indicated concerns that freeze seal failures in secondary systems can also be significant because of the potential for consequential failures, such as the loss of residual heat removal in the River Bend event. The information notice identified procedural inadequacies that resulted in a failure to install and monitor a temperature detection device, and a lack of personnel training in the use of freeze seals. Other important considerations identified in the notice included: "examining training, procedures, and contingency plans associated with the use of freeze seals, and evaluating the need for and availability of additional water makeup systems and their associated support systems." No specific statement was included regarding the applicability of a 10 CFR 50.59 safety evaluation.

6.7.1.2 Industry Reports on Use and Installation of Freeze Seals

An INPO significant event report (June 1989) noted that freeze plug failure can result in the following:

(1) internal flooding of large volumes of water into plant equipment and structures

(2) damage to and contamination of equipment

(3) initiation of plant transients

In February 1989, the Electric Power Research Institute issued EPRI NP-6384 D, "Freeze Sealing (Plugging) of Piping," to guide nuclear power plant maintenance personnel in evaluating the use of freeze seals. The guide cautions personnel on the use of freeze seals and discusses contingency plans should freeze seals fail.

The Battelle Columbus Laboratories issued a final report, "Development of Guidelines for Use of Ice Plugs and Hydrostatic Testing," in November 1982; the Battelle report discusses the potential hazards associated with ice plugs and gives guidelines for plug slippage, restraint, pressure, impact loads and stress arising from handling. Defects and safety of personnel are also discussed.

6.7.1.3 NSAC-125, "Industry Guidelines for 10 CFR 50.59 Safety Evaluations"

NSAC-125, issued in June 1989 by the Nuclear Management Resources Council (NUMARC), gave the industry guidelines for performing 10 CFR 50.59 safety evaluations. The document provides industry guidance on the thresholds for unreviewed safety questions, the applicability of 10 CFR 50.59, and the procedures for performing 10 CFR 50.59 safety reviews for facility changes, tests, or experiments at nuclear power stations. The staff's review of NSAC-125 identified the following as appropriate guidance for the applicability of the

10 CFR 50.59 safety evaluation to the use of freeze seals as temporary modifications and the application of the 10 CFR 50.59 determination of whether an unreviewed safety question exists for the freeze seal installation: "Temporary changes to the facility should be evaluated to determine if an unreviewed safety question exists. Examples of temporary modifications include jumpers and lifted leads, temporary lead shielding on pipes and equipment, temporary blocks and bypasses, temporary supports, and equipment used on a temporary basis."

Although the use of freeze seals as a temporary block is not specifically identified, freeze seals perform the "temporary block" function and, therefore, the staff finds they conform with the NSAC-125 definition of "temporary modifications."

6.7.1.4 Results and Findings

° For BWRs, failure of a freeze seal in a system connected to the vessel's lower plenum region such as the reactor water cleanup (RWCU) system, could cause the core to become uncovered in less than 1 hour if the seal failed completely and makeup water was not added to the reactor.

° NSAC-125, industry guidance for application of 10 CFR 50.59, covers temporary modifications but does not discuss freeze seals specifically.

° Temporary modifications using freeze seals are not being evaluated per 10 CFR 50.59.

° Industry guidance exists for using freeze seals with contingency plans.

° Operating experience indicates that freeze seal failures could be safety problems.

6.7.2 Thimble Tube Seals

The arrangement of the in-core instrumentation assemblies in most PWRs may be visualized by considering one end of an approximately 1-inch tube as welded to the bottom of the reactor vessel and the other end welded to the seal table. This tube provides a penetration into the reactor from below, with the opposite end containing a high pressure seal during power operation. This "guide" tube is a permanent part of the reactor coolant system pressure boundary.

A thimble tube, which has a closed end, is inserted into the guide tube, closed end first, and pushed through the guide tube until it extends up into the reactor core. The thimble tube is then sealed to the guide tube by a high-pressure, Swagelok-type fitting at the seal table, thus forming a watertight assembly with the area between the tubes containing reactor coolant system water and the inside of the thimble tube open to the containment building. The space between the tubes is subjected to reactor coolant system pressure during power operation.

Preparation for refueling involves withdrawing the thimble tubes out of the core. Thus, the normal Swagelok thimble tube to guide tube seal at the seal table must be opened.

Once the thimble tube is withdrawn from the core region, the annular gap is closed by a temporary seal composed of split components and rubber gaskets. Temporary thimble tube seals have a typical design pressure of 25 psi so that a significant overpressurization could cause them to fail. This would cause a leak that is effectively in the bottom of the reactor vessel. The impact of the failure of these seals on RCS inventory has been analyzed and found to be significant, as discussed below.

The B&W thimble tubes terminate in an "incore instrumentation tank" that is open at the top, at the refueling floor level, with the bottom at roughly reactor vessel flange level. No temporary seals are used and the tank fills with water (or is filled) so that tank and refueling cavity water level remain the same. There are times during typical refueling outages when the tank is open to containment at the bottom and when some of the guide tubes are empty, thus providing a significant flow path between the reactor vessel and the incore instrumentation tank as well as to the containment.

Most CE units do not use such bottom-entering incore instrumentation of the above type. The staff understands that the few that do use a B&W-type arrangement to terminate the tubes in the refueling cavity rather than a separate tank.

Analysis of Instrument Tube Thimble Seal Failure

An analysis of the instrument tube thimble seal failure in a Westinghouse designed plant was performed to determine how long it takes to uncover the core when one steam generator is used to remove decay heat following a loss of RHR. This analysis is part of the transient thermal-hydraulic analysis of the loss of RHR in a PWR discussed in Section 6.6.1.2.

Thimble seal failure in the instrument tubes was assumed to occur when system pressure reached 20 psi. This value was chosen to investigate the consequences of failure of the thimble seals and may not reflect actual failure pressures for seals. For this analysis, it is assumed that there are 58 thimble seals and all of these seals fail, once the assumed failure pressure is achieved. The tubes have an inner diameter of 0.4 inch resulting in a total area of 0.0506 square feet. This "break" area provides an "upper bound," since the thimble tubes may remain in the guide tubes, making the flow area much smaller. The failure location is assumed to be at the seal table which is at the elevation of the reactor vessel flange. The tubes are connected to the vessel at the bottom of the lower head and are collected at the seal table resulting in an elevation difference between these two locations of about 22.5 feet.

The RCS was initialized with water at 90°F at a level at the centerline of the hot and cold legs. Air at 90°F and 100-percent relative humidity is present in all volumes above the centerline of the hot and cold legs. The decay heat power level corresponding to 1 day after shutdown was conservatively assumed for this analysis (11.5 MW).

Thimble seal failure is predicted to occur at about 1.6 hours after the RHR system is lost. Core uncovery in this conservative analysis is predicted to occur about 20 minutes later. These results demonstrate that very little time is available to establish a means for injecting water into the RCS to prevent heatup of the core.

6.8 Rapid Boron Dilution

The staff, with the assistance of Brookhaven National Laboratory (BNL), has completed a study of rapid boron dilution sequences which might be possible under shutdown conditions in PWRs (NUREG/CR-5819). Concerns relating to rapid boron dilution during a PWR startup were raised by the French regulatory authority in their shutdown PRA study. These sequences are the result of a two-step process. In the first step it is assumed that unborated (or highly diluted) water enters the normally borated reactor coolant system (RCS) while the reactor coolant is stagnant in some part of the primary system. This diluted water is then assumed to accumulate in this region without significant mixing. The second step is the startup of a reactor coolant pump (RCP) so that the slug of diluted water will rapidly pass through the core with the potential to cause a power excursion sufficiently large to result in core damage. Other variations to this two-step process include (1) having the slug forced through the core by the inadvertent blowdown of an accumulator and (2) having a loop isolated using loop stop valves and after the loop becomes diluted, opening the loop stop valves while the RCPs are running.

6.8.1 Accident Sequence Analysis

This study has considered both probabilistic and deterministic aspects of the problem and has focused on what is expected to be the most likely of the several sequences that have been identified as leading to a rapid dilution. This particular sequence starts (see NRC Information Notice 91-54) with the highly borated reactor being deborated as part of the startup procedure. The reactor is at hot conditions with the RCPs running and the shutdown banks removed. Unborated or diluted water is being pumped by charging pumps from the volume control tank into the cold leg. The initiating event is a loss of offsite power (LOOP). This causes the RCPs and the charging pumps to trip and the shutdown rods to scram. The charging pump comes bac' on line quickly whe diesel generators start up. Charging continues until the volume control tank is empty and it is assumed that there is little mixing with the water in the RCS so that a region of diluted water accumulates in the lower plenum. It is then assumed that power is recovered so that the RCPs can be restarted. This is assumed to occur after sufficient diluted water has accumulated so that the slug of diluted water which then passes through the core has the potential to cause fuel damage.

The probabilistic analysis was done for this scenario for a Combustion Engineering plant (Calvert Cliffs), a Babcock & Wilcox plant (Oconee) and a Westinghouse plant (Surry). The reactor systems and operating procedures ...ed in the scenario were reviewed and accident event trees were developed. The analysis focused on the specific arrangement of the makeup and letdown systems and the chemical and volume control system. The startup and dilution procedures were important, as were the procedures to recover from a LOOP.

The initiating frequency of the scenario was considered for both refueling and non-refueling outages and varied from 2.0E-4 per year to 5.0E-5 per year, depending on the reactor. The probability that the injected water would cause a region of diluted water before an RCP was started was treated as a time-dependent function. It was assumed that there was no mixing of a given injectant, but the core damage probability is not constant in time because it takes time to accumulate sufficient diluted water, and because after emptying the volume control tank, the suction from the charging pump switches to a source of highly borated water. The time dependence of the probability of restarting an RCP was also taken into account. The resulting core-damage frequency was found to vary from 1.0E-5 to 3.0E-05 per reactor year.

6.8.2 Thermal-Hydraulic Analysis for the Event Sequence

A key assumption in the probabilistic analysis is that the injectant does not mix with the existing water in the RCS so that a diluted region accumulates in the lower plenum. This assumption was tested by using mixing models to determine to what extent charging flow mixes with the existing water when it is injected into a loop which is either stagnant or at some low natural circulation flow rate insufficient to provide complete mixing. These mixing models are based on the regional mixing models that were developed to understand the thermal mixing of cold injectant into the "cold" leg which is at a much higher temperature. The thermal mixing problem was originally of interest for the problem of pressurized thermal shock.

The regional mixing model has been utilized to calculate the boron concentration in the mixed fluid when the unborated cold injected water mixes with the hot water in the cold leg which is taken to have a boron concentration of 1500 ppm. The model specifically considers the mixing region near the point of injection and at the end of the cold leg where the flow is into the downcomer, and ignores mixing in the downcomer or lower plenum.

The model was applied to the Surry plant under the assumption of no loop flow. The finding was that there is considerable mixing so that the water in the lower plenum would have a boron concentration that is only 200-300 ppm less than that originally in the core. On the basis of the neutronic calculations explained below, this is insufficient to cause a power excursion when an RCP is restarted. It is difficult to generalize these results as they are dependent on specific plant parameters defining the loop geometry and the charging flow.

6.8.3 Neutronics Analysis

The neutronics of this problem has been studied to understand the consequences of having a slug of diluted water pass through the core. In order to do simple scoping calculations, the staff took a synthesis approach. This approach combines steady-state, three-dimensional core calculations of boron reactivity worth under different configurations with point kinetics calculations of the resulting power transient.

The steady-state calculations were done with the NODEP-2 nodal code. The output from these calculations is the static reactivity worth of a diluted slug as a function of position of the slug as it moves through the core. The two basic shapes that have been considered are a semi-infinite slug (step function) and a finite slug (rectangular wave function) with a volume of 5[illegible] cubic feet. The calculations were done with different dilutions, relative to the 1500 ppm assumed as the initial state of the core. In addition to a radially uniform slug, two other geometries were considered. In one, the slug was localized in the center 49 assemblies and in another the slug was found at two peripheral locations affecting [illegible] assemblies. The calculations provided not only reactivity versus position of the leading edge of the slug but also Doppler weight factors for use in the kinetics calculations.

The dynamics calculations included the neutron kinetics as well as a simple fuel rod conduction model to calculate a more accurate fuel temperature than would be obtained by making an adiabatic assumption. The calculated peak fuel enthalpy was used as the criterion to judge whether [illegible]l damage had occurred.

If it was greater than 280 calories per gram then catastrophic fuel damage involving a change in geometry was assumed to occur. The peak fuel enthalpy was calculated using the time-dependent power and a power peaking factor taken from the static three-dimensional calculation at the condition corresponding to the time of the peak power.

The results show that fuel damage could occur if the boron concentration in a semi-infinite slug is reduced to 750 ppm corresponding to an equal mixing of injected water at 0 ppm and reactor coolant at 1500 ppm. These results are dependent on the worth of the shutdown banks and on the Doppler reactivity coefficient; calculations were done to determine this sensitivity.

6.8.4 Other Analyses

Transient calculations somewhat similar to these studies have been done by several other groups. Examples are

(1) Westinghouse [S. Salah et al., "Three-Dimensional Kinetic Analysis of an Asymmetric Boron Dilution in a PWR Core," Transactions of the American Nuclear Society, 15, 2, (1972)] performed calculations for a situation wherein the loop stop valves are both cold (down to 70°F from 547°F) and completely unborated due to an unknown mechanism. Westinghouse used a three-dimensional neutron kinetics analysis to asses the core response when the loop stop valves were assumed to open while the RCPs were running. All rods were assumed to be initially out of the core and hence, the worth of the scram reactivity (not including the assumed "stuck rod") would be about 6 or 7-percent delta-k. The result, for an initial 1500-ppm boron concentration, was (a) integrated core power not above normal core average power, but (b) localized fuel damage in the cold, unborated, stuck rod core region, involving only about 3 percent of the fuel and "not sufficient energy release to break the integrity of the primary system."

(2) Calculations performed as part of a thesis (S. Jacobson, "Some Local Dilution Transients in a Pressurized Water Reactor," Thesis No. 171, LIU-TEK-LIC-1989:11, Linkoping University, Sweden) examined similar transients with various dilution scenarios. The steam generator tube rupture/ accumulation of a diluted region during primary pump shutdown/rapid core dilution following pump turn-on was the most significant event found in the study. The conclusion drawn from this study was that the fuel failure criterion (similar to that used in the BNL studies above) is not exceeded.

The review and analysis of rapid boron dilution events during shutdown appears to indicate that core damage may occur for assumed extreme sets of event parameters, including a necessary assumption of minimal mixing of diluted and borated water, and may occur with a frequency of the order of 10^{-5} per reactor-year. These events can be prevented by the use of appropriate procedures which anticipate the possibility of dilution in various recognized situations and prevent it, or prevent the inappropriate starting of pumps until suitable mixing procedures are carried out.

6.9 Containment Capability

6.9.1 Need for Containment Integrity During Shutdown

The NRC staff performed scoping calculations of core heatup for a Westinghouse four-loop PWR to allow assessment of containment response and a potential release. For loss of residual heat removal during midloop operations, the time to heat the core to boiling was calculated to be 8 minutes. Once boiling began, the reactor vessel level could decrease to the top of the active fuel in as little as 50 minutes. This calculation assumes that the reactor had operated for a full cycle and had been shut down for 48 hours. Additionally, 35 percent of the reactor coolant inventory between the top of the active fuel and the middle of the hot leg was assumed to spill from the RCS.

PWRs have containment structures that are classified as large dry, subatmospheric or ice condenser. For any of these containment designs, the reestablishment of containment integrity before core damage occurs is important for reducing offsite doses. The effect of a containment in reducing the offsite dose consequences is evaluated by comparing what might occur if the containment were open to what might occur if the fission products remained within the closed containment. An open containment would allow direct release of steam and fission products to the atmosphere; holdup in the containment would allow plateout and decay to occur.

Offsite dose consequences from a postulated severe accident were evaluated with and without a containment in the NRC "Response Technical Manual RTM-91" NUREG/BR-0150. RTM-91 evaluates offsite dose at a distance of 1 mile from a typical site for varying degrees of core heatup and damage. The values used here are based on the assumption that the release occurs immediately after shutdown. In one case, the dose is evaluated for an accident causing damage only to the fuel cladding with release of the volatile fission products stored in the fuel pin gas space. The dose rate from further heating includes the release of the volatile fission products retained in the grain boundary regions within the fuel pellets and, finally, release following a postulated core melt is considered. Without the benefit of containment retention, the doses 1 mile from the plant would be high, ranging from 20 rem (whole body) and 2000 rem (thyroid) for a gap release to 1000 rem (whole body) and 100,000 rem (thyroid) for a postulated core melt.

A release 48 hours after shutdown would also have severe consequences since most of the thyroid dose results from inhaling iodine-131. Iodine-131 has a half-life of 8.1 days for a dose reduction by a factor of [illegible] after 48 hours. The whole-body dose would be somewhat more affected by a prior shutdown of 48 hours since short-lived isotopes make up about 80 percent of the whole-body dose following an immediate release. The whole-body dose 1 mile from the plant would be about 200 rem considering 48-hour decay. This would be principally from I-131 with its 8.1-day half-life. Further retention of the fission products prior to release would cause the offsite dose to be reduced by about 97 percent of the initial release value, with long-lived cesium isotopes as the principal contributors to contamination. These estimates assumed release of 25 percent of core iodine and 1 percent of particulates. The evaluations are appropriate for large dry PWR containments, subatmospheric containments, and ice condenser containments for which the ice bed was bypassed by the escaping steam. For releases through the ice bed, reduction factors of between 0.3 and 0.5 are expected.

The effect of holdup and plateout in containment on offsite dose was determined in RTM-91 to be significant. With a 24-hour holdup in containment and with design leakage assumed, calculated offsite doses are reduced to 5E-5 rem (whole body) and 4E-3 rem (thyroid) for the gap release case and 0.002 rem (whole body) and 0.2 rem (thyroid) for the core-melt case. Thyroid and whole body doses are further reduced by factors of 5 and 3 respectively, if the containment spray was operated during the event. Doses would of course be increased by any subsequent containment failure and revaporization of fission products that might occur following a hypothetical severe core damage accident.

BWRs are not typically operated in a reduced inventory condition as are PWRs. However, 2 days into an outage, a BWR/4 (such as Browns Ferry) may have as little as 205 inches of reactor coolant above the top of the active fuel. If shutdown cooling were lost, boiling would begin in 28 minutes. The reactor vessel water level would be at the top of the active fuel 308 minutes later. This corresponds to a steam flow rate of 24,800 cubic feet per minute into the Mark I secondary containment with the drywell head removed for refueling.

This flow into the secondary containment could increase the internal pressure to 0.5 psig in 5 minutes. This pressure is significant because the secondary panels are designed to blow out at 0.5 psig, releasing steam and fission products directly to the atmosphere. The calculation to determine the time to secondary containment failure was based on an energy balance after depositing 285,000 pounds of steam into the secondary containment. The heat sink inside the secondary containment is made up of structural steel and air. No secondary system leakage was assumed.

Two other calculations were performed to determine the secondary containment's sensitivity to changes in the mass of structural steel and air inside secondary containment. The first increased the mass of steel inside secondary containment by five times that used in the previous calculation. This increased the amount of time for secondary containment to reach 0.5 psig from 5 minutes to 6 minutes. The second decreased the volume of containment from 4 million cubic feet to 2 million cubic feet. That resulted in decreasing the amount of time from 5 minutes to 3 minutes for secondary containment to reach .5 psig. This sensitivity study was necessary because secondary containment designs and sizes vary from plant to plant.

RTM-91 also evaluated offsite doses at a distance of 1 mile from a typical BWR site for varying degrees of core heatup and damage. If the drywell head were removed, the release could go directly into secondary containment and through the blowout panels for Mark I and II containments, bypassing standby gas treatment. As in the PWR evaluation, the dose is calculated for releases from three cases: the fuel pin gap space, the grain boundary, and core melt. The BWR doses would range from 20 rem (whole body) and 2000 rem (thyroid) for a gap release to 1000 rem (whole body) and 100,000 rem (thyroid) for a postulated core melt. These are the same doses listed for the PWR case.

RTM-91 Table C-3 gives a reduction factor of 0.01 for dry-low-pressure flow and 1.0 for wet-high-pressure flow through the standby gas treatment system filters. Considering the fact that 24,800 cubic feet per minute of saturated steam is being deposited inside secondary containment and a typical standby gas treatment exhaust fan is only rated for 5000 cubic feet per minute, the flow through the standby gas treatment system will be closer to the wet-high-pressure case and the dose will not be significantly reduced.

6.9.2 Current Licensee Practice

Generic Letter 88-17 was issued to PWR licensees and required, among other things, implementation of procedures and administrative controls that reasonably assure that containment closure will be achieved before the time that RPV water level would drop below the top of the active fuel following a loss of shutdown cooling under reduced inventory conditions. The NRC staff assessed whether the requirements of GL 88-17 were in place by implementating special inspections at each site under the inspection guidance in TI-2515/101 and /103. The Vogtle IIT recognized the need to develop broader recommendations for low-power and shutdown operation. This led to the NRC staff's program to visit selected plant sites undergoing low-power shutdown operation (see Chapter 3). The staff also observed a variety of practices at the sites. For the PWRs, the staff noted that licensees were attempting to meet the recommendations of GL 88-17. Some licensees went beyond the recommendations of GL 88-17 by providing procedures for rapid containment closure for plant conditions other than reduced inventory.

Closure of the equipment hatch would be required for maintaining containment integrity. This operation required electric power at about 25 percent of the sites visited. In one case, use of the polar crane would be required. Some licensees utilized the equipment hatch as a passageway for electrical cables and hoses. At these sites, rapid removal of this equipment was provided for by the use of quick disconnects. Some plants also provided bolt cutters and axes for contingency. One of the sites visited demonstrated an equipment hatch closure capability requirement of within approximately 15 minutes of loss of residual heat removal. The onsite review report noted that this was more often the exception than the rule.

Several factors are key to ensuring that the equipment hatch is closed in a timely matter. These include accounting for radiological and environmental conditions that could result from reactor coolant being boiled into the containment, addressing the number and location of closure bolts, providing for the loss of ac power, keeping tools need for closing the equipment hatch near at hand, and finally, training and rehearsing personnel in the closure procedure. The closure of the equipment hatch in sufficient time is essential to keeping possible releases within established guidelines. These observations also apply to licensees with BWR Mark III containments. GL 88-17 was not sent to BWR licensees and the onsite review report noted that these licensees have not made provisions for rapid equipment hatch closure.

One licensee reporting a quarter-inch gap at the top of the equipment hatch when four bolts were used, found it necessary to use two more bolts to close the gap. GL 88-17 specified a no-gap criterion for hatch closure, but not every licensee confirmed that this condition was obtained. Tests or observations must be performed on internal equipment hatches to determine the location and minimum number of bolts needed to obtain an adequate closure. For external hatches, containment pressure effects on hatch closure must be considered along with the source term when evaluating the minimum number of bolts necessary to achieve an acceptable leak-tightness.

Procedures for controlling and closing containment penetrations varied widely. Some licensees did not initiate closure until temperatures exceeded 200°F. Above 200°F boiling might begin quickly. The licensees, however, had not evaluated the in-containment environment and the ability of personnel to work in that environment to perform the necessary containment closure operations. Some plants require that the containment always be closed during midloop operations. In one case, the licensee interpreted this as meeting GL 88-17 recommendations and, therefore, did not develop procedures for rapid containment closure. Water-filled, U-pipe loop seal configurations found at several plants provided containment entry for electrical cables and tubing. The water-filled U-pipes were judged in adequate for withstanding containment pressure conditions that might exist following a loss of shutdown cooling.

6.9.3 PWR and BWR Equipment Hatch Designs

In order to gain a better understanding of containment capability in PWRs and BWRs during an accident that occurs while a plant is shut down, the staff has gathered information regarding the design of equipment hatches. This was done by surveying resident inspectors at U.S. plants.

The hatch survey was conducted using a questionnaire on specific equipment hatch parameters. Answers to the questionnaire were tabulated and grouped under BWR or PWR. For BWRs, the survey asked for information on the equipment hatch that would be used only for removing a recirculation pump motor; the survey did not address removing and replacing a drywell head. The results of the survey are tabulated in Appendix C.

The majority of equipment hatches for both BWRs and PWRs were pressure seating hatch designs (59% for BWR, 82% for PWR). For BWRs, the resident inspectors polled indicated that the equipment hatch (either recirculation pump motor or CRD hatch) would generally be removed along with the drywell head, but that removal of the equipment hatch alone was unlikely.

PWR equipment hatches consisted of 9 of the pressure unseating type and 33 of the pressure seating type. Twenty four procedures required the use of ac and/or compressed air to install the hatch under normal conditions but five resident inspectors indicated that the licensee had a procedure to close the hatch manually. Four plants with pressure unseating hatches can use a truck-mounted crane to install the equipment hatch during a loss of normal ac power. Six PWR plants did not require the equipment hatch to be in place during fuel movement. They are: Braidwood, Byron, Cook, Palisades, San Onofre 1, and Zion. These have their hatches located so that they open to the fuel handling building which has HVAC to process contaminated air during a fuel drop event.

Three PWR resident inspectors and the licensees for Catawba, McGuire, and Salem have noticed that the minimum number of bolts as specified in the technical specification is not sufficient to bring all hatch sealing surfaces into contact. A noticeable gap was present with use of the minimum number of bolts. Two licensees (Palo Verde and Summer) ran successful leak tests, an Appendix J type A and a type B, with the minimum number of bolts installed. Discussion with two hatch vendors indicated that hatches have been designed so that the sealing surfaces should mate with the minimum number of bolts installed.

Ginna and Indian Point 2 have fabricated temporary closure plates which are used when the equipment hatch is removed but temporary services are run into the containment. The Indian Point 2 temporary closure plate is rated for 3 psid and has penetrations for fluid and electrical services.

6.9.4 Containment Environment Considerations for Personnel Access

6.9.4.1 Temperature Considerations

NRC staff estimated that approximately 50,000 pounds of steam could be deposited inside the containment 1 hour after loss of residual heat removal in a Westinghouse four-loop PWR occurring 2 days after shutdown. The steam is a result of boiling in the reactor coolant from the middle of the hot leg to the top of the active fuel and assumes 35 percent of the reactor coolant is spilled from the RCS. The staff assumed that the containment volume was 2 million cubic feet of dry air at 70°F and that the containment environment after the event would consist of air and structural steel at an elevated temperature, steam, and condensed steam in the form of water. The calculation did not consider the containment fan coolers and assumed no leakage from the containment. Under these conditions, the staff expects the containment atmosphere to go from 70°F and atmospheric pressure to 150°F and 5.9 psig in about 1 hour (see Figure 6-1).

This condition would be of concern because at about 160°F the air is hot enough to burn the lungs. Therefore personnel inside the containment would have to be equipped with self-contained breathing apparatues.

PWR Containment
Temperature vs. Time

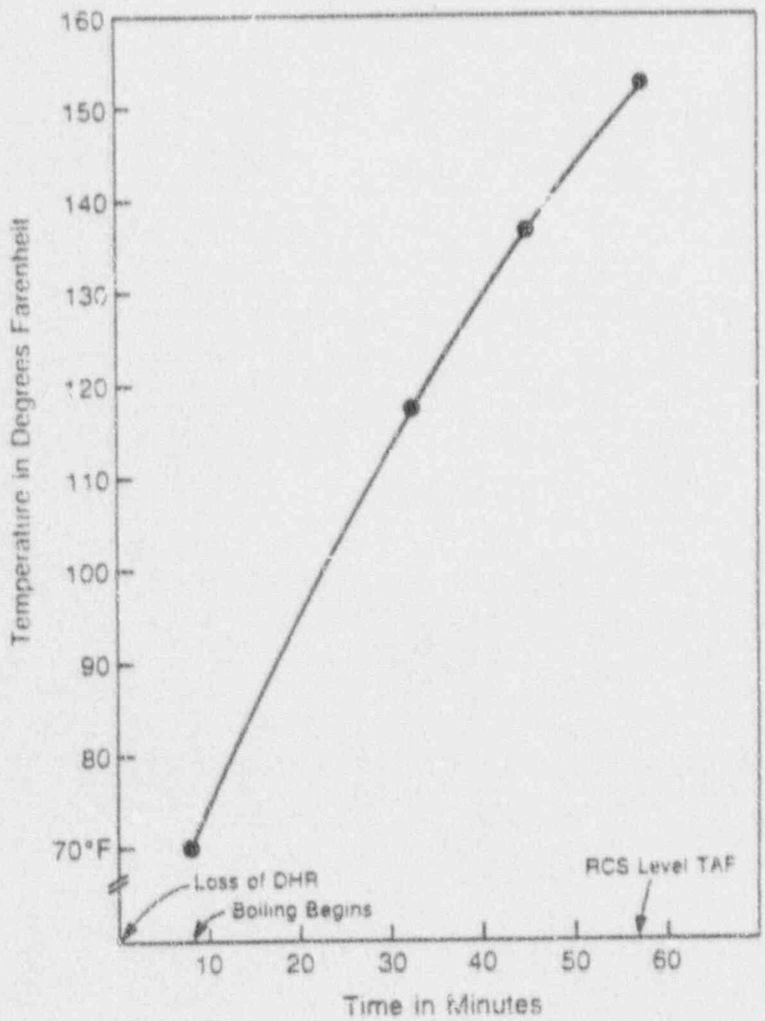


Figure 6-1

6.9.4.2 Radiological Considerations

Boiling of coolant within an opened reactor system following a postulated loss of shutdown cooling would release dissolved fission products within the containment atmosphere. If significant radioactivity were contained in the coolant, high-radiation-area alarms would be actuated. These are typically set at twice the background level. Health physics personnel would be expected to evacuate the containment until people could safely enter using the appropriate precautions and protective measures to perform any operation required to close the containment.

To assess the radiological conditions that workers might experience while closing the containment, the NRC staff performed scoping calculations. The staff assumed that the coolant contained the expected activity for a typical operating PWR and then for a BWR as given in RTM-91. Radioactive decay was assumed to progress for 48 hours before boiling began. Iodine decay into xenon was included. The resultant concentration for PWRs was about one-twentieth of the 1.0 microcurie-per-milliliter maximum equivalent of I-131 allowed in plant technical specifications. Although there is no specific requirement, PWR operators typically reduce coolant activity by two orders of magnitude using coolant cleanup systems before opening the reactor system. Additional reduction could be achieved, but the length of the outage might be increased. The scoping calculation should be considered conservative because it did not account for coolant cleanup.

The volatile fission products--noble gases and iodine--were assumed to be carried out with the boiled coolant. The particulates--cesium, strontium, and neptunium--were assumed to undergo a 1/100 partition. With these assumptions, the release of fission products to the containment was calculated concurrently with the steam released by decay heat boiling. The boiling rate was based on decay heat from a 3400 MWt plant shut down for 48 hours at the end of cycle. The steam was assumed to be mixed with the containment atmosphere (2 million cubic feet, PWR) and the mixture released through containment openings at a constant volumetric flow. Dose rates were derived from the guidance in the NRC

Site Access Training Manual which states that the risk of one Part 20 MPC-hour is approximately equal to 2.5 mrem of whole body dose.

The resulting PWR equivalent doses are depicted in Figures 6-2 and 6-3. (These ordinarily are conservative because they do not include the factor-of-100 reduction discussed in the preceding paragraph.) Inhaled iodine dose in the non-respirator case was computed using soluble MPCs, whereas the respirator case was computed using the insoluble MPCs for iodine. The calculated equivalent dose increases with time and approaches asymptotic values for a pure steam atmosphere. These calculations indicate that self-contained breathing apparatus would be required for an extended stay within the containment because of the dose and humidity, since the filtration type would not function adequately in high humidity above about 106°F. It may be difficult to perform containment closure operations in self-contained breathing apparatus because the air supply will limit how long personnel can stay on the job. In evaluating recovery actions following a potential loss of shutdown cooling, licensees should avoid plant conditions in which steaming could occur before the containment was closed, unless reduced coolant activities or limited requirements for personnel entry indicated that the associated risk was acceptable.

Using the expected coolant activities in RTM-91 for BWRs, the calculated equivalent dose with and without respirator protection was much less than for PWRs. See Figures 6-4 and 6-5. This is because BWRs do not retain volatile fission products in the coolant. The loss of shutdown cooling with subsequent boiling was assumed to occur in a typical Mark II containment 48 hours after shutdown with the drywell head removed. Perfect mixing was assumed in the secondary containment volume above the refueling floor (1.6 million cubic feet). Other assumptions were similar to the PWR calculation. The lower dose rates calculated for the BWR would allow for a longer stay within the containment than allowed for the PWR case, and the major concern may be the steam conditions in working areas. If practical, procedures for drywell closure under emergency conditions are desirable, since offsite releases from a severe accident could have unacceptable consequences as discussed in Section 6.9.1.

PWR DOSE RATES

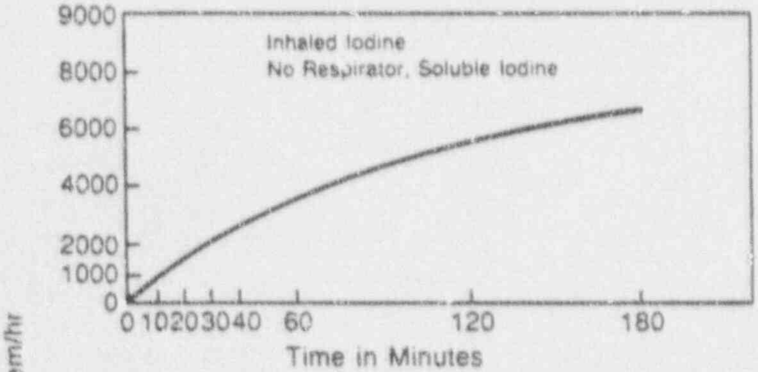


Figure 6-2

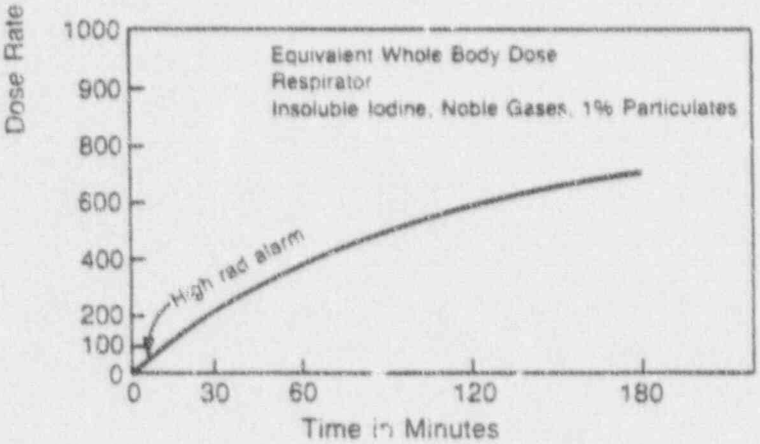


Figure 6-3

BWR DOSE RATES

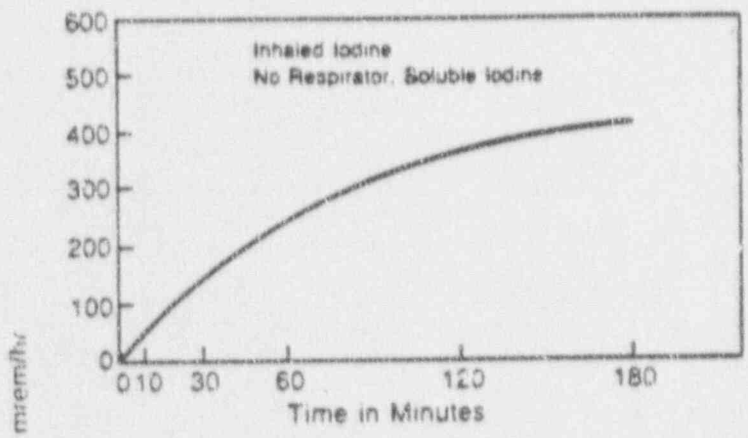


Figure 6-4

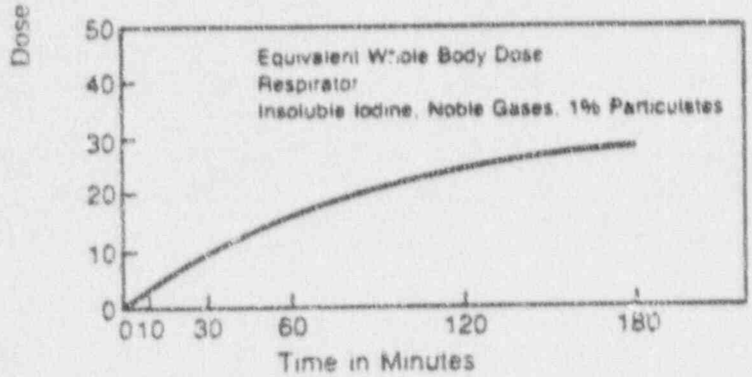


Figure 6-5

6.9.5 Findings

° The estimated dose from a core melt 2 days after shutdown with an open containment is roughly 80,000 rem (thyroid) and 200 rem at a 1-mile distance from the plant. A closed PWR containment with 24 hours (whole body) followed by design rate leakage reduces these to 0.2 rem (thyroid) and 0.001 rem (whole body).

° BWR secondary containments are anticipated to fail within a few minutes of initiation of bulk boiling if the steam is released into the containment. Boiling can begin a half-hour after RHR loss if the loss occurs 2 days after shutdown.

° The plant visit program (see Chapter 3) found no BWRs for which containment closure was considered if RHR were lost. Existing secondary containments were judged to be of little use if the reactor vessel and primary containment were open.

° PWR licensee response was mixed concerning recommendations in GL 88-17 regarding containment closure. Some licensees have not fully evaluated attaining a no-gap equipment hatch closure. Closure techniques for other penetrations were sometimes poor. No licensee fully addressed the containment work environment if it planned to close the containment while steam was being released into the containment. Most closure procedures were weak and few had been rehearsed.

° About half of the PWR sites require ac power to close the equipment hatch. Only one of these appears to have made provisions for closure if ac power were lost.

° Staff scoping analyses show that PWR containments probably require self-contained breathing apparatus within an hour of initiation of steam release into the containment due to the steam and temperature. (Localized heating and steam hazards were not considered.) Dose rates may not be serious if there are no fuel cladding leaks and if the licensee has significantly cleaned the primary system water, although breathing apparatus is likely to be needed. Airborne contaminants are of more concern with fuel leaks or contaminated primary water.

° Most containment concerns are eliminated if the containment is closed or if it is assured to be closed before to initiation of steam release from the RCS.

6.10 Fire Protection During Shutdown and Refueling

During shutdown and refueling outages, activities that take place in the plant may increase fire hazards in safety-related systems that are essential to the plant's capability to maintain core cooling. The plant technical specifications (TS) allow various safety systems to be taken out of service to facilitate system maintenance, inspection and testing. In addition, during plant shutdown/refueling outages, major plant modifications are fabricated, installed, and tested. In support of these outage-related activities, increased transient combustibles (e.g., lubricating oils, cleaning solvents, paints, wood, plastics) and ignition sources (e.g., welding, cutting and griding operations, and electrical hazards associated with temporary power) present additional fire risks to those plant systems maintaining shutdown cooling.

During plant shutdown, a postulated fire condition could potentially cause fire dmaage to the operable train or trains of residual heat removal capability. This fire damage could further complicate the plant's capabilitiy to remove decay heat.

In order to fully assess the fire risk during refueling conditions, the following action plan was implemented at a PWR and a BWR facility that the staff visited:

(1) Review the adequacy of current NRC fire protection guidance with respect to the protection of the systems necessary to perform the residual heat removal function during shutdown and refueling modes of operation

(2) Evaluate the fire protection requirements of Appendix R to 10 CFR Part 50 for cold-shutdown systems and determine if those requirements are adequate to assure the availability of residual heat removal capability under postulated fire conditions

(3) Review administrative controls and methods for reducing fire hazards during shutdown and refueling modes of operation.

The results of this review and evaluation in each of the three areas are discussed next.

6.10.1 Adequacy of Current NRC Fire Protection Guidance for the Assurance of Residual Heat Removal Capability

The NRC fire protection guidance (NUREG-0800 SRP 9.5.1) applied to ensure that an adequate level of fire protection exists, is a defense-in-depth approach. This approach is focused on the following programmatic areas:

(1) fire prevention through the use of administrative controls (e.g., good housekeeping practices, control of combustible materials, control and proper handling of flammable and combustible liquids, control of ignition sources)

(2) rapid fire detection through the use of early-warning fire-smoke-detection systems, fire suppression that occurs quickly through the application of fixed fire extinguishing systems and/or manual fighting means, and limiting fire damage through the application of passive fire protection features

(3) designing plant safety systems that provide for continued operation of essential plant systems necessary to shut down the reactor in those instances in which fire prevention programs are not immediately effective in extinguishing the fire

The defense-in-depth concept, as it applies to fire protection, focuses on achieving and maintaining safe shutdown conditions from a full-power condition. In addition, the SRP guidance given to licensees for conducting a fire hazard analysis, specifies that the analysis should demonstrate that the plant will maintain the ability to perform safe-shutdown functions and minimize radioactive releases to the environment in the event that a fire occurs anyplace in the plant. The SRP guidance established for the performance of a fire hazard analysis does not address shutdown/refueling conditions and the potential impact a fire may have on the plant's ability to remove decay heat and maintain reactor water temperature below saturation conditions.

The SRP establishes three levels of fire damage limits for safety-related/safe-shutdown systems. The limits are established according to the safety function of the structure, system, or components. The following material summarizes the fire damage limits: (1) one train of equipment necessary to achieve hot standby/shutdown from either the control room or emergency control stations must be maintained free from fire damage by a single fire, including an explosive fire; (2) both trains of equipment necessary to achieve cold shutdown may be limited so that at least one train can be repaired or made operable within 72 hours using onsite capability; and (3) both trains of systems necessary for mitigation of consequences following design-basis accidents may be damaged by a single fire. These damage limits are based on the assumption that full reactor power operation is the major limiting condition with respect to fire and its potential risk on reactor safety. The acceptable fire damage threshold for residual-heat removal functions has not been established in the SRP with respect to the various shutdown and refueling modes of operation.

6.10.2 Evaluation of Requirements for Cold Shutdown

The Appendix R fire protection criteria for the protection of safe-shutdown capability does not include those systems important to assuring an adequate level of residual heat removal during non-power modes of operation. Appendix R, Section III.G and III.L, allow certain repairs to cold-shutdown components to restore system operability and the ability to achieve and maintain cold-shutdown conditions. This repair provision includes the decay heat removal functions of the residual heat removal (RHR) system. Appendix R requirements focus on full-power operation and address the impact a fire may have on the plant's ability to achieve and maintain safe-shutdown conditions.

During plant shutdown conditions where the reactor head is removed, the RHR system and its associated support systems are performing the decay heat removal function (i.e. for PWR--component cooling water system, service water system, offsite/onsite ac/dc power train; for BWR--reactor building closed cooling water system, high-pressure service water system, offsite/onsite ac/dc power train). Depending on the specific mode of operation and the plant configuration (i.e., BWR/PWR--head off vessel water level at the vessel flange; PWR--head off in midloop operations), the plant TS may require both trains or only one train of decay heat removal capability to be operable.

At one PWR facility visited, approximately 30 plant areas were associated directly with either the A or B train of decay heat removal. In 15 plant areas, both trains of residual heat removal were present. This facility elected to comply with the Appendix R requirements by utilizing damage control/repair procedures. Under the Appendix R damage control/repair approach, a postulated fire during shutdown/refueling conditions, in a plant area where both decay heat removal system trains are present, could cause fire damage to redundant trains resulting in a potential loss of decay heat removal capability. By contrast, if the plant was at 100% power operations at the time of the fire, the plant could be held in hot standby until the necessary repairs, allowed under Appendix R, could be made and subsequent cold shutdown could be achieved. For example, if the power cable to the RHR pump motor suffered fire damage, the plant maintenance staff estimated that it would take 16 hours to implement a repair and restore power to the pump. If this same postulated fire were to occur during shutdown/refueling, reactor coolant saturation conditions could potentially occur. As discussed in Section 6.6, there are several options available, depending on the plant configuration, for supplying water and/or providing limited RCS cooling. However, it should be noted that, without the performance of a detailed shutdown/refueling fire hazards analysis, the alternate RCS makeup and cooling options may have been affected by the same fire which caused the loss of decay heat removal.

During a BWR plant visit, it was determined that approximately 7 areas of the reactor building and 10 areas of the control building are associated with the decay heat removal function. Three areas in the reactor building and six areas in the control building contained both trains. In the areas containing both trains of decay heat removal, fire protection features in accordance with Appendix R, Sections III.G and III.L, were provided. Since this plant's capability to achieve cold shutdown complies with Appendix R, Sections III.G and III.L, RHR fire damage/control procedures were not required. However, by postulating a fire during shutdown and refueling conditions which required only one train of decay heat removal to be operable (the train provided with Appendix R fire protection is unavailable due to maintenance), in a plant area where the unprotected train is present, damage could could be sustained to the operable train resulting in a total loss of decay heat removal capability. Under these conditions, RCS heatup to saturation could occur There are several options available, depending on plant configurations, for supplying water to the RCS. These options include CRD pumps, standby liquid control system from test tank, condensate pumps, condensate or demineralized water via hoses from the service box on the fuel floor, core spray from the torus or condensate storage tank, refueling water transfer pump, high-pressure service water system, and makeup to reactor cavity skimmer surge tank and overflow into the reactor cavity. Alternate decay heat removal can be accomplished via the reactor cleanup or the fuel pool cooling systems. It should be noted that without the performance of a detailed shutdown and refueling fire analysis, the alternate RCS makeup and cooling options may not be available. The equipment and/or components associated with these options may be affected by the same fire that causes the loss of decay heat removal.

6.10.3 Review of Plant Controls for Fire Preventation

The staff reviewed fire prevention administrative and control procedures associated with the control of transient combustibles, and ignition sources, and also reviewed compensatory measures for fire protection impairment, and other measures. The fire prevention administrative control measures are applicable to both power operation and shutdown conditions. However, it was noted that in order to support certain work activities (e.g., welding and cutting) associated with maintenance or modifications, a temporary fire prevention administrative control procedure was changed. For example, a fire watch may be assigned to more than one welding or cutting operation; or increased combustible loading above that analyzed for full-power conditions may be introduced into safety-related areas to support maintenance operation. Fire prevention administrative control procedures did not provide enhanced controls or compensatory measures during shutdown conditions in those plant areas critical to supporting RCS makeup or decay heat removal.

During the PWR and BWR plant visits, when a plant walkdown was performed in areas that were associated with decay heat removal, an increase in fire hazards was noted. These fire hazards included temporary electrical and test wiring, increased transient combustibles (e.g., wood scaffolding, plastic sheeting and containers, lube oil, cleaning solvents, paper products, rubber products, and more) and increased welding and cutting activities. In addition, the staff noted that fire protection personnel at the site had not increased their inspections. The staffing level is limited and fire prevention inspections restricted because so much paper work was generated by activities associated with maintenance and modifications during an outage.

The lack of increased fire prevention/protection activities commensurate with the increased maintenance and modification activites during plant shutdown and refueling is reflected by the increased fire frequency which occurs. At the two facilities visited, reviewing the fire reports for a 18 month operating period, three fires occurred at the PWR and four fires at the BWR facility. Six of the seven total fires that occurred at these facilities were during refueling outages.

6.10.4 Summary of Findings

- A postulated fire could potentially damage the operable train or trains of decay heat removal systems during shutdown conditions. In addition, plant configurations can further complicate the plant's ability to remove decay heat.

- Increased transient combustibles, and ignition sources during outage activities present additional fire risks to their minimum required TS systems required to maintain shutdown cooling.

- SRP guidance established for the performance of a fire hazard analysis does not address shutdown and refueling conditions and the potential impact a fire may have on the plant's ability to maintain core cooling.

- 10 CFR Part 50, Appendix R, fire protection criteria for the protection of safe shutdown capability do not include those systems important to assuring an adequate level of decay heat removal during non-power modes of operation.

- Fire prevention administrative control procedures did not provide enhanced controls or compensatory measures during shutdown conditions in those plant areas critical to supporting RCS makeup or decay heat removal.

- The staffing level at the site for fire prevention is limited and inspection activities are restricted because so much paper work was generated by activities associated with maintenance and modifications during an outage.

- A majority of the fires at the facilities occurred during refueling outages.

6.11 FUEL HANDLING AND HEAVY LOADS

Mishaps in handling fuels and heavy loads during the refueling process can occur and have a potential for

(1) causing an array of new or spent fuel to become critical,

(2) damage to fuel assemblies which causes release of radioactivity, and

(3) overheating of spent fuel pool which causes damage to fuel cladding

6.11.1 Fuel Handling

In order to minimize fuel handling mishaps, the fuel handling equipment is designed and built in accordance with specified standards to prevent dropping fuel. In addition, fuel handling equipment is also tested before the fuel handling process to assure its proper operation. Design guidelines for such such equipment include the provision of high-temperature alarms and high-radiation alarms, should fuel damage or failures be imminent.

Criticality involved in the movement of a single fuel assembly is extremely unlikely with the greatest potential occurring in the case of misplacement of an element in the core or spent fuel pool (SFP). Proper planning and particular attention to details during the fuel handling process can minimize the probability of mistakes. In BWRs, the potential for criticality during refueling is minimized by starting the process with the mode switch in the refueling or shutdown position and with all rods in. In PWRs, the boron concentration in the reactor coolant and refueling canal is kept at a level sufficient to assure a K_{eff} equal to or less than 0.95 or, as an alternative, the boron concentration is kept equal to or greater than 1850 ppm. In addition, licensees are required to analyze the worst case of fuel mislocation and provide assurance that the concomitant fuel damage does not cause offsite doses in excess of specified criteria.

The licensee is also required to analyze the condition for an uncontrolled control rod assembly (a bank for a PWR and a single rod for a BWR) withdrawal at subcritical or low-power condition and to provide assurance that certain preset criteria, which includes thermal margin limits, fuel centerline temperatures, and uniform cladding strain for BWRs, are not exceeded.

Release of radioactivity from a spent fuel element may be caused by mechanical damage, such as dropping or striking it against some object. Dropping is minimized by proper design of handling equipment in accordance with specified criteria. Nevertheless, equipment has failed and fuel elements have been damaged. In order to minimize the radiation dosage as a result of such mishaps, all spent fuel must be moved under water during the refueling process. Current standard technical specifications (TS) for both PWRs and BWRs require that a specified level of water must be maintained above the reactor vessel head and spent fuel storage pools during refueling. This level of water is capable of acting as shielding for the handling of spent fuel and for absorption of the radioactivity that could be released should a spent fuel element be damaged. In addition, the fuel handling equipment is tested before being used in order to avoid using faulty equipment, and to assure load handling limitations as required by TS.

For PWRs, TS require that penetrations in the containment building be closed or be capable of being closed by an operable automatic valve on a high-radiation signal in the containment, before initiating the refueling process. For BWR, TS require that the integrity of the fuel handling building be assured before handling irradiated fuel.

As a final protection against the potential excessive radiation doses resulting from a fuel handling accident, the licensee must provide an analysis of the radiological consequences of a fuel handling accident to assure that results will conform to applicable dose limitations.

Spent fuel in the spent fuel pool is kept cool by a spent fuel pool cooling system. TS for PWRs and BWRs require that such system be operable in order to keep spent fuel cooled. TS also require that water level in the spent fuel pools and temperatures be maintained to minimize dose levels during fuel handling. Spent fuel cooling systems are analyzed to assure that proper spent fuel pool coolant temperatures are maintained at all times of storage of spent fuel so as to prevent overheating of the stored fuel.

6.11.2 Heavy Load Handling

In cases where access to the reactor core is required, it is necessary to remove the internal components. In doing so, the fuel elements could be damaged should a heavy load be dropped, resulting in the release of radioactive elements from damaged fuel. Relocation of damaged fuel into a critical mass is also of concern. Similar circumstances could occur upon lifting a heavy load over spent fuel elements stored temporarily in the containment or in the spent fuel storage pool.

Any heavy load carried over redundant equipment used for removal of decay heat has a potential for damaging or destroying this equipment or other equipment involved in shutdown. Damage, in such case, is limited by following safe load paths or by minimizing the potential for damage, as noted below.

Risk associated with heavy loads can be minimized as outlined in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants": (1) by making the potential for a load drop extremely small, by utilizing a single-failure-proof lifting system in accordance with NUREG-0612 or (2) by evaluating a potential load drop accident and taking actions to ensure that damage is so limited that

(a) loss of coolant as can be replaced by normal makeup sources and

(b) there is no loss of capability for systems to maintain safe shutdown

In order to minimize the potential for a drop of a heavy load, the licensee were required to: (1) develop procedures for heavy loads handling, (2) train and qualify crane operators, (3) design special lifting devices in accordance with specified criteria, (4) design other lifting devices (other than "special") in accordance with specified criteria, (5) provide inspection, testing and maintenance of cranes in accordance with specified guidelines, (6) have cranes designed in accordance with specified criteria and (7) follow safe load paths, as noted above.

Three potential hazards regarding the handling of heavy loads are: (1) damage to surroundings in the improper design or use of handling equipment so as to permit swinging or rotating of the load, on breaking of one holding line, (2) improper handling of the internals of the MK-I BWRs and, by reference, of the internals of any reactor so as to damage the vessel, the core or other safety-related equipment, and (3) dropping of loads placed on the edge of the spent fuel pool.

A representative of each NRC regional office was contacted in an effort to determine whether they had observed problems in these areas. Item (3) (i.e., dropping loads from the edge of the fuel pool) was revealed and is discussed below.

There appears to be no special generic problem in handling heavy loads on a generic basis. The handling of heavy reactor internals can be done safely by adhering to the guidelines in NUREG-0612. The problem of load swing or rotation can be avoided by proper load handling. Since the staff has not identified such an event, they have concluded that load handling procedures are working successfully in the field.

6.12 ONSITE EMERGENCY PLANNING

The staff's technical evaluation of shutdown and low-power operation has shown that event sequences with potential offsite consequences can occur during cold-shutdown and refueling conditions. The plant configuration during shutdown and refueling conditions is significantly different from that during power operation. As a result, the sequence of events and the operator's ability to detect and respond to an event and mitigate its consequences may vary significantly during shutdown and refueling conditions. Therefore, the need for an operator to respond appropriately to an incident, including emergency classifications and notifications of offsite officials, still exists during cold-shutdown and refueling conditions.

6.12.1 Classification of Emergencies

Guidance for classification of emergencies for nuclear plants during power operation is found in Appendix 1 to NUREG-0654, FEMA-REP-1, Rev. 1 entitled "Criteria for Preparation and Evaluation of Radiological Emergency Response Plans and Preparedness in Support of Nuclear Power Plants." This guidance does not explicitly address the different modes of nuclear power plant operation. It is generally recognized, however, that the initiating conditions established in Appendix 1 to NUREG-0654 apply as a whole to a nuclear plant during its power operation and hot-shutdown modes. Some, but not all, of the initiating conditions in NUREG-0654 may apply to a nuclear plant during cold-shutdown and refueling conditions.

Because initiating conditions contained in Appendix 1 to NUREG-0654 were not intended to be directly and fully applicable to shutdown and refueling conditions and their unique characteristics, their use by the licensees has resulted in inconsistencies and often times excess conservatism in the classification of emergencies during shutdown or refueling conditions. For example, the loss of vital ac power and RHR at Vogtle Unit 1 in March 1990 was classified as a Site Area Emergency by the licensee but might have been classified as an Alert by a different licensee. In an event at Oyster Creek in March 1991 an Alert was declared when it was determined that both sources of onsite ac power were unavailable. However, offsite ac power was available at the time and the refueling cavity was flooded with water.

NUMARC has developed a method for defining emergency action levels which is referenced in NUMARC/NESP-007, Revision 1. Although the NUMARC approach is not considered complete in that regard. NRC will continue to work with NUMARC to issue the final guidance that will help licensees to identify initiating conditions and develop associated emergency action levels for shutdown and refueling conditions with a revised NUREG-0654 by spring of 1993. In the mean time, the staff will develop interim guidance for emergency classification during shutdown and refueling conditions to be issued within the next 6 months. The interim is discussed in Chapter 7.

6.12.2 Protection of Plant Workers

NRC regulations in 10 CFR 50.47(b)(10) require that a range of protective actions be developed for emergency workers and the public. In meeting this requirement as stated in Criterion J of NUREG-0654, the NRC expects each licensee to evacuate nonessential personnel and to account for onsite personnel within 30 minutes of the declaration of an emergency. During outage periods, hundreds of additional workers may be on site for maintenance, construction, and repairs. In addition to the presence of large numbers of workers on site during an outage, there will be many unusual activities taking place and normally available equipment and instrumentation may be lacking. These conditions, common during shutdown and refueling outages, can place an additional burden on the emergency response capability at the time of an accident. Emergency plans and procedures must address the evacuation and accountability of the large number of nonessential personnel on site should an accident occur during plant shutdown or refueling conditions.

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APPENDIX C

RESULTS OF EQUIPMENT HATCH SURVEY

Table C-1

Details of Equipment Hatch Survey: BWRs

Plant & OL date	Containment Type	Hatch Type 1/	No of bolts	Additional inspection for refueling closure	Temporary platform	Air or AC Needed	Bolt Pattern 2/	Comments
Big Rock Pt. 64	Sphere	in	Bayonet	App. J Type B	No	AC	Bayonet	TS requires containment when fuel is in Reactor. Double door.
Browns Ferry 73/74/76	Mark I	In 3/	12	None	Ladder	Manual	Hold-down clamp	
Brunswick 1&2 76/74	Mark I	In	12	None	No	Manual	B	
Clinton 87	Mark III	In	20	None	Yes	Manual	B	
Cooper 74	Mark I	In	8	None	No	None	A	
Dresden 2&3 69/71	Mark I	In	8	No	No	Manual	B	
Duane Arnold 74	Mark I	In	12	No	Yes	AC	B	Need AC for crane to install hatch.
Fermi 85	Mark I	Out, In	20/36	No	Yes	Manual	B	Two equipment hatches.

See footnotes at end of table.

Table C-1 (Continued)

Plant & OL date	Containment Type	Hatch Type	No of bolts	Additional inspection for refueling closure	Temporary platform	Air or AC Needed	Bolt Pattern	Comments
FitzPatrick 74	Mark I	In	8	None	No	Manual	B	
Grand Gulf 84	Mark III	In	20	None	No	AC	B	
Hatch 1&2 74/78	Mark I	In	8	None	Yes	Manual	B	Can close hatch without temporary platforms.
Hope Creek 86	Mark I	In	24	None	Yes	Manual	B	Hatch hasn't been removed since operations.
LaSalle 1&2 82/84	Mark II	In	16	None	No	Manual	B	
Limerick 1&2 85/89	Mark II	Out	80	None	Yes	AC	B	
Millstone 1 86	Mark I	In	8	None	Ladder	Manual	B	
Monticello 81	Mark I	In	8	None	No	Manual	B	

Table C-1 (Continued)

Plant & OL date	Containment Type	Hatch Type	No of bolts	Additional inspection for refueling closure	Temporary platform	Air or AC Needed	Bolt Pattern	Comments
Nine Mile Pt 1 74	Mark I	Out	36	None	Yes	Manual	B	Inspector noticed a gap with minimum bolts installed.
Nine Mile Pt 2 87	Mark II	Out	64	None	Yes	Manual	B	
Oyster Creek 69	Mark I	-	-	-	-	-		
Peach Bottom 2&3 73/74	Mark I	In	8	None	No	Manual	B	
Perry 86	Mark III	Out	72	None	Yes	AC	A	
Pilgrim 72	Mark I	Out	8	None	No	No	A	Licensee noted closing ASAP difficult due to temporary services.
Quad Cities 1&2 72/72	Mark I	In	8	None	Yes	Manual	B	
River Bend 85	Mark III	Out	64	None	No	Manual	A	

Table C-1 (Continued)

Plant & OL date	Contain-ment Type	Hatch Type	No of bolts	Additional inspection for refuel-ing closure	Tempo-rary plat-form	Air or AC Needed	Bolt Pattern	Comments
Susquehana 1&2 82/84	Mark II	Out	30	None	No	Air & AC	B	Can close manually.
Vermont Yankee 73	Mark I	Out	8	None	No	Manual	B	
Washington Nuc.2 84	Mark II	Out	64	None	No	Air	A	Licensee can close hatch manually.

1/ Hatch type: Out = pressure unseating design; In = pressure seating design.
2/ Bolt Pattern: A = bolt in threaded hole; B = bolt swing.
3/ Flat plate.

Table C-2

Details of Equipment Hatch Survey: PWRs

Plant (Vendor) & OL date	Containment Type	Hatch Type [1]	No of bolts	Additional inspection for refueling closure	Temporary platform	Air or AC Needed	Bolt Pattern [2]	Comments
Arkansas 1 (B&W) 74	Large dry	In	4/24	None	No	Manual	B	
Arkansas 2 (CE) 78	Large dry	In	4/16	None	No	Manual	B	No procedure for temp. closing; just tighten bolt, close opening.
Beaver Valley 1&2 (W) 76/87	Sub Atmos.	In	4/24	None	Ladder	Manual	B	Emergency airlock inside hatch.
Braidwood 1&2 (W) 87/88	Large dry	In	0/20 [5]	None	Yes	AC	B	Opens to fuel handling bldg. Have look ISO valves, don't drain to midloop.
Byron 1&2 (W) 85/87	Large dry	In	0/20 [5]	None	Yes	AC	B	Opens to fuel handling bldg. Have look ISO valves, don't drain to midloop.
Callaway (W) 84	Large dry	In	4/20	None	No	AC	B	Special rigging needed to close during SBO.

See footnotes at end of table.

Table C-2 (Continued)

Plant (Vendor) & OL date	Containment Type	Hatch Type	No of bolts	Additional inspection for refueling closure	Temporary platform	Air or AC Needed	Bolt Pattern	Comments
Calvrt Cliffs 1&2 (W) 74/76	Large dry	In	4/20	None	No	AC	B	
Catawba 1&2 (W) 85/86	Ice	In	4/16 4/24	None	no	AC	B	U2 was mod. to add bolts to seal. Inspector notes # of bolts used for fuel move increased to close gap. U1 uses 10, U2 uses 15 bolts.
Comanche Peak (W)	Large dry	In	4/16	None	Ladder	Manual	B	
Cook 1&2 (W) 74/77	Ice	Out	0/32 [5]	None	No	AC	A	No requirement for hatch, but lic. maintans for fuel move & midloop.
Crystal River (B&W) 77	Large dry	Out	4/72	None	Yes	Air	B	Hatch can be closed manually w/truck-mounted crane.
Davis-Besse (B&W) 77	Large dry	In	4/12	None	Yes	Manual	B	

See footnotes at end of table.

Table C-2 (Continued)

Plant (Vendor) & OL date	Containment Type	Hatch Type	No of bolts	Additional inspection for refueling closure	Temporary platform	Air or AC Needed	Bolt Pattern	Comments
Diablo Canyon 1&2 (W) 84/85	Large dry	In	4/48	Daylight check	Ladder	Manual	B	Perform daylight check. One seal may be used for modes #5 & #6.
Farley 1&2 (W) 77/81	Large dry	In	4/28	None	Yes	Manual	B	
Fort Calhoun (CE) 73	Large dry	In	4/36	None	No	AC	B	
Ginna (W) 84	Large dry	Out	36/36	QC Metal	Yes	Manual	B	Lic. uses a temporary closure plate.
Haddam Neck (W) 74	Large dry	Out	18/92	None	No	AC	B	Mobile crane can be used to install hatch.
Harris (W) 87	Large dry	Out	4/36	None	Ladder	Manual	A	
Indian Pt 2 (W) 73	Large dry	In	20/20	None	No	AC [3/]	B	Lic. has a temporary closure plate for temp. services.

See footnotes at end of table.

Table C-2 (Continued)

Plant (Vendor) & OL date	Containment Type	Hatch Type	No of bolts	Additional inspection for refueling closure	Temporary platform	Air or AC Needed	Bolt Pattern	Comments
Indian Pt 3 (W) 76	Large dry	In	20/20	None	No	AC [3]	B	NYPA doesn't have temp. closure plate.
Kewaunee (W) 73	Large dry	In	12/12	None	4	AC	A	Uses boatswain chair to close hatch.
Maine Yankee (CE) 73	Sphere	Out	8/74	None	No	Manual	A	Need mobile crane for hatch.
McGuire 1&2 (W) 81/83	Ice	In	4/16	None	Ladder	Manual	hold-down clamp	Noticed gap w/4 & 8 bolts in place.
Millstone 2 (CE) 86	Large dry	in	4/20	None	Yes	Manual	B	
Millstone 3 (CE) 86	Sub Atmos.	In	6/16	None	No	Manual	B	
North Anna 1&2 (W) 78/80	Sub Atmos.	In	4/20	None	No	Manual	B	Lic. requires every 2nd bolt installed.
Oconee 1, 2 & 3 (B&W) 73/73/74	Large dry	In	4/48	None	No	AC	B	Can position w/o AC.

See footnotes at end of table.

Table C-2 (Continued)

Plant (Vendor) & OL date	Containment Type	Hatch Type	No of bolts	Additional inspection for refueling closure	Temporary platform	Air or AC Needed	Bolt Pattern	Comments
Palisades (CE) 72	Large Dry	In	0/24 5/	None	Ladder	Manual	B	Hatch opens to fuel handling bldg. Procedures to discontinue temp srvcs. on loss of shutdown cooling.
Palo Verde 1 2 & 3 (CE) 85/86/87	Large dry	In	4/32	Ran ILRT w/8 bolts	No	AC	B	Can close manually. Ran ILRT w/8 bolts. Lic. closes hatch on reduced inventory.
Point Beach 1&2 (W) 70/73	Large dry	In	66/66	None	No	Manual	B	
Prairie Island 1&2 (W) 74/74	Large dry	In	0/12	App. J Type B	Ladder	Manual	B	TS doesn't specify # of bolts. Ladders are secured near hatch.

See footnotes at end of table.

Table C-2 (Continued)

Plant (Vendor) & OL date	Containment Type	Hatch Type	No of bolts	Additional inspection for refueling closure	Temporary platform	Air or AC Needed	Bolt Pattern	Comments
Robinson (W) 70	Large dry	Out	8/48	None	Ladder	Manual & mobile crane		80-ton mobile crane for hatch. Has a hatch seal penetration press. system.
Salem 1&2 (W) 76/81	Large dry	In	4/16	None	Yes	AC	B	Lic. & inspector have noticed gap w/4 bolts installed.
San Onofre 1 (W) 67	Sphere	In	0/12	None	No	Manual	B	U1 refuels through hatch (new fuel). Close quickly w/SBO.
San Onofre 2&3 (CE) 82/83	Large dry	In	4/16	None	No	AC	B	4 hrs to close w/SBO.
Seabrook (W) 90	Large dry	In	4/32	None	Yes	AC crane	B	Recently completed 1st refuel.
Sequoyah 1&2 (W) 80/81	Ice	In	4/20	None	No	AC winch		Can use chain fall in place of winch.
South Texas 1&2 (W) 88/89	Large dry	In	4/28	None	No	AC	B	

Table C-2 (Continued)

Plant (Vendor) & OL date	Contain-ment Type	Hatch Type	No of bolts	Additional inspection for refuel-ing closure	Tempo-rary plat-form	Air or AC Needed	Bolt Pattern	Comments
St. Lucie 1&2 (CE) 76/83	Large dry	Out	4/12	None	No	AC	B	
Summer (W) 82	Large dry	In	4/30	App J Type B	Ladder	AC	B	LLRT w/4 bolts. Could close w/o AC.
Surry 1&2 (W) 72/73	Sub Atmos.	In	4/36	None	No	Manual	B	Lic. has temp. cover plate use for aux. services.
TMI 1 (B&W) 74	Large dry	Out	4/72	None	Yes	Manual	B	Emergency hatch common w/equip. hatch and mounted on carriage.
Trojan (W) 75	Large dry	In	4/20	None	No	No	B	Procedure to close in SBO.
Turkey Pt. 3&4 (W) 72/73	Large dry	In	4/58	None	No	Air	A	Hatch can be positioned manually.
Vogtle 1&2 (W) 87/88	Large dry	In	4/30	None	No	AC	B	Can close hatch during SBO.
Waterford (CE) 85	Large dry	In	4/16	None	Yes	Manual	B	

Table C-2 (Continued)

Plant (Vendor) & OL date	Contain-ment Type	Hatch Type	No of bolts	Additional inspection for refuel-ing closure	Tempo-rary plat-form	Air or AC Needed	Bolt Pattern	Comments
Wolf Creek (W) 85	Large dry	In	4/20	None	No	AC	B	
Yankee Rowe (W) 63	Sphere	In	4/56	None	[illegible]	AC	B	
Zion 1&2 (W) 73/73	Large dry	In	0/12 5/	Seal press. system	No	AC/Air	B	Lic. can install 2 hrs w/SBO.
								Installed during midlo
								Hatch connects w/fuel handling bldg.

1/ Hatch Type: Out = pressure unseating design; In = pressure seating design.
2/ Bolt pattern: A = bolt in threaded hole; B = bolt swing.
3/ Polar crane.
4/ Crane & boatswain chair.
5/ Hatches open to fuel handling building.