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Evaluation of Events Involving Decay Heat Removal Systems in Nuclear Power Plants

Prepared by J. A. Haried

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Prepared for
U.S. Nuclear Regulatory Commission

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FOREWORD

The Nuclear Safety Information Center (NSIC), which was established in March 1963 at Oak Ridge National Laboratory, is sponsored by the U.S. Nuclear Regulatory Commission's Office for Analysis and Evaluation of Operational Data. Support for the technical progress review *Nuclear Safety* (see last page of this report) is provided by both the Breeder Reactor and Light-Water-Reactor Safety Programs of the Department of Energy. NSIC is a focal point for the collection, storage, evaluation, and dissemination of operational safety information to aid those concerned with the analysis, design, and operation of nuclear facilities. The Center prepares reports and bibliographies as listed on the inside covers of this document. NSIC has developed a system of keywords to index the information it catalogs. The title, author, installation, abstract, and keywords for each document reviewed are recorded at the central computing facility in Oak Ridge.

Computer programs have been developed that enable NSIC to (1) prepare monthly reports with indexed summaries of Licensee Event Reports, (2) make retrospective searches of the stored references, and (3) produce topical indexed bibliographies. In addition, the Center Staff is available for consultation, and the document literature at NSIC is available for examination. NSIC reports (i.e., those with ORNL/NSIC and ORNL/NUREG/NSIC numbers) may be purchased from the National Technical Information Service (see inside front cover). All of the above services are available free of charge to U.S. Government organizations as well as their direct contractors. Persons interested in any of the services offered by NSIC should address inquiries to:

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PREFACE

The Nuclear Regulatory Commission (NRC) Division of Safety Technology in the Office of Nuclear Reactor Regulation assigned the project entitled *Special Studies of Reactor Operating Experience* to the Nuclear Safety Information Center (NSIC) in the early part of FY-1981. The object of this project was to identify safety-significant implications of current nuclear power plant operating experience by special studies of the following specific subsystems: compressed air and backup nitrogen, service water, decay heat removal, and boron dilution.

Two to three man-months of engineering assessment was devoted to each of the studies. The information used was basically that found in NSIC's files. The documents containing this information are available to the public in the NRC Public Document Room, 1717 H. Street, Washington, DC 20555. The scope of the project did not include visits to the plants or meetings with inspectors of the NRC Office of Inspection and Enforcement.

Project personnel for the studies were

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EVALUATION OF EVENTS INVOLVING DECAY HEAT REMOVAL SYSTEMS IN NUCLEAR POWER PLANTS

J. A. Haried

Abstract

This report reviews and evaluates events placed in the NSIC file involving the removal of decay heat in U.S. commercial boiling- and pressurized-water reactors from June 1979 through June 1981. The information was collected from operating experiences, licensee event reports, system designs in safety analysis reports, and other regulatory documents. The results were collated and analyzed according to safety significance and cause of event.

Thirty-eight reported events in these 2.1 years meet the criteria for safety significance. Steam bubble formation in the reactor vessel head during natural circulation cooldown at St. Lucie 1 was the most significant event; operator awareness of the possibility of this occurrence and preparedness for dealing with it was the most important recommendation. Cavitation of residual heat removal pumps during decay heat removal operation was the most common potentially significant event. Davis-Besse 1 had several instances in which an inadvertent signal to the safety features actuation system caused the operating residual heat removal pumps to align to the dry sump causing pump cavitation.

1. INTRODUCTION

In this study, reactor operating experiences with decay heat removal (DHR) systems are reviewed and evaluated to identify any possible significant implications for reactor safety, so that preventiv action may be taken. The results of this study will aid the Nuclear Regulatory Commission (NRC) in accomplishing the preventive action.

1.1 Background

The Office of Nuclear Reactor Regulation (NRR) has the responsibility for evaluating reactor operating experience to detect events or trends that may be of safety significance; the results of these evaluat are factored into the licensing process. To screen operating experience effectively, the adoption of a systematic approach in reviewing and evaluating this experience is essential. Therefore, these reviews and evaluations are applied to selected subsystems and may be extended to reviews on a plant-by-plant basis to complete the evaluation of reactor operating experience in a thorough and comprehensive manner.

Concern for the removal of decay heat from light-water-reactor (LWR) cores is not new. As far back as August 1974, the NRC expressed concern to all reactor vendors regarding boron concentration and precipitation following large-break loss-of-coolant accidents (LOCAs) and the possible consequent constriction of DHR flow over long time periods. In March 1980, the NRC designated shutdown DHR requirements as an unresolved safety issue.[1] This new designation signals an increasing realization that one of the important factors in the safety of nuclear reactors is the reliability of DHR methods following the shutdown of the reactor for any reason. Insufficient reliability of DHR methods, particularly in response to small-break LOCAs, was shown to be responsible for a substantial portion of the overall probability of core meltdown. Later reliability studies and related experience from the accident at Three Mile Island, Unit 2 (TMI-2) have reaffirmed that the loss of capability to remove heat through a steam generator is a significant contributor to the probability of a core-melt event.[1]

Note that following the TMI-2 accident, the NRC required reactor manufacturers to make many improvements to the steam generator auxiliary feedwater (AFW) system. Also, the NRC staff felt that upgraded and/or additional means of DHR could substantially increase a plants' capability to deal with a broader spectrum of transients and accidents and, therefore, could potentially reduce significantly the overall risk to the public. Consequently, the NRC is investigating alternative means of DHR in LWR plants, including but not limited to the use of existing equipment where possible.[1]

Before upgraded and/or additional means of DHR are considered, it is important to understand how well current systems cope with accident sequences that can jeopardize plant safety. If the present DHR methods are found to be unacceptable, design criteria for both existing and alternative DHR methods will be developed, as appropriate. Design criteria will have to consider both frequent events (such as loss of offsite power, loss of feedwater, small-break LOCAs) and special emergencies (such as seismic events, sabotage, airplane crashes) (Ref. 2 is an example). Because of the broad spectrum of LWR designs in currently operating plants and the wide variation in plant age, a considerable number of existing plants will be analyzed to complete the identification of new unresolved safety issues.[1] This study is one input among many for the resolution of these safety issues.

1.2 Removal of Decay Heat

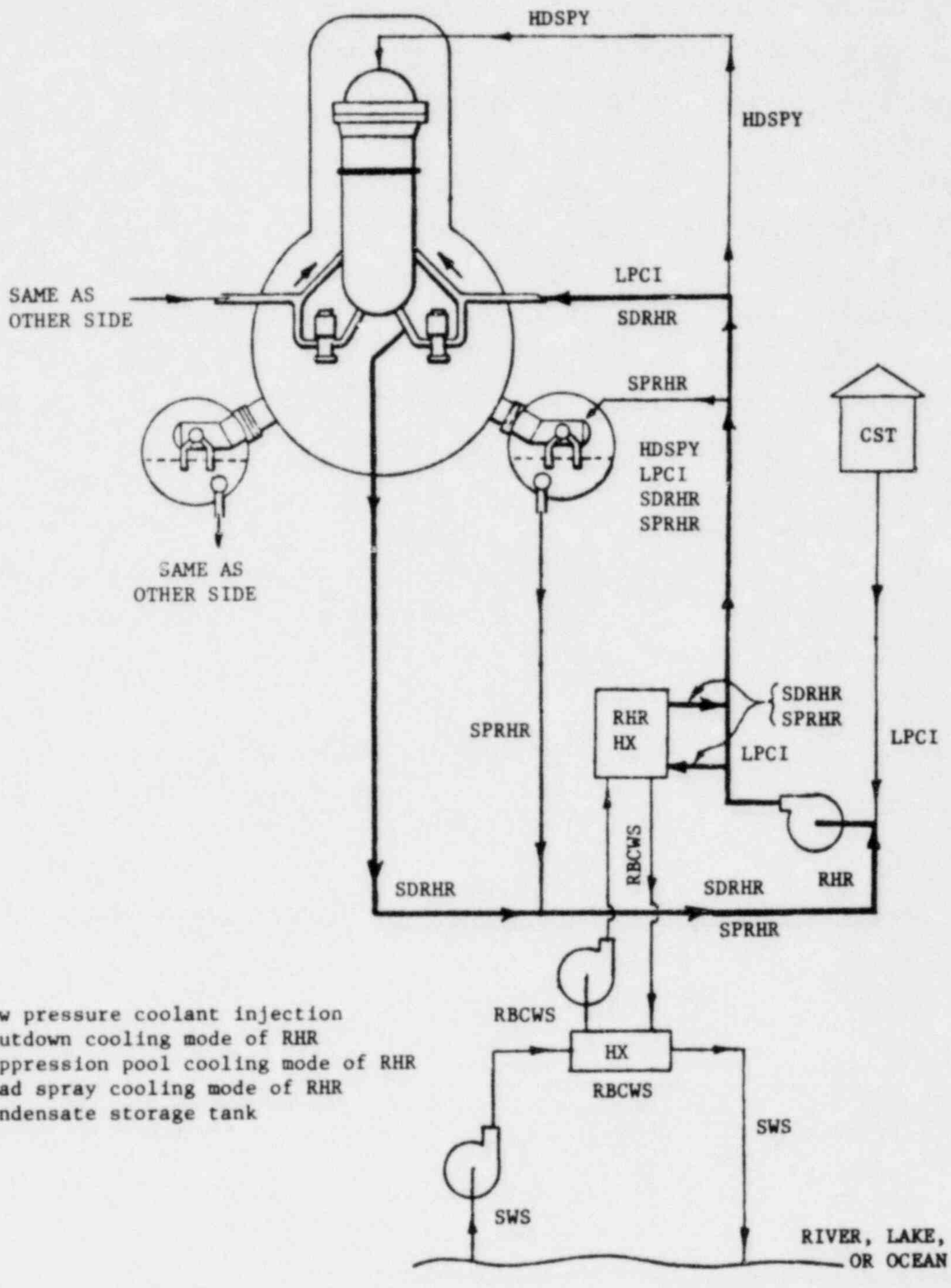
In this report, the phrase decay heat removal is used to represent the functional description of the systems, whereas the phrase residual heat removal (RHR) is used to denote the physical system(s) and component(s).

1.2.1 DHR in boiling-water reactors (BWRs) (Fig. 1)

The principal means for removing decay heat in BWRs during a normal shutdown while the reactor is at high pressure is via the steam lines to the turbine condenser. The condensate is returned to the reactor vessel by the feedwater system.

The emergency means for removing decay heat in BWRs during accident conditions varies among the different classes of General Electric reactor designs. BWR/1 (Big Rock Point, Dresden 1, and Humbolt Bay) differs significantly in many respects from the other BWR classes. (Because no DHR events reported herein occurred at a BWR/1, that design is not described here.) Also, plants within the same BWR product line usually vary somewhat. The description of DHR methods in Sect. 2.1 is useful for general concepts of DHR in BWRs. For descriptions of specific designs, consult the appropriate Final Safety Analysis Reports (FSARs).

In BWR/3 and 4, the high-pressure coolant injection (HPCI) system is the high-pressure emergency core-cooling system (ECCS). BWR/5 and 6



LPCI - low pressure coolant injection
SDRHR - shutdown cooling mode of RHR
SPRHR - suppression pool cooling mode of RHR
HDSPY - head spray cooling mode of RHR
CST - condensate storage tank

Fig. 1. Example of a BWR RHR system.

have no HPCI system; high-pressure core spray (HPCS) is the high-pressure ECCS in BWR/5 and 6. The primary source of water to HPCI and HPCS is the condensate storage tank (CST); the secondary source is the suppression pool. For some plants, raw water can serve as an emergency water source.

In BWR/3-6 the automatic depressurization system (ADS) is the second high-pressure ECCS. The purpose of the ADS is to reduce reactor pressure so that the large capacity, low-pressure ECCSs [low-pressure coolant injection (LPCI) and core spray] can operate. The major components of the ADS are relief valves on the main steam lines, which open and vent steam to the suppression pool.

When the primary system is at low pressure, decay heat is removed by the several modes of the RHR system. LPCI is the priority mode of RHR and is an ECCS. Core spray is a redundant, low-pressure ECCS, separate from RHR. More detailed descriptions of normal and postaccident DHR methods at high and low reactor pressure are provided in Sect. 2.1.

1.2.2 DHR in pressurized-water reactors (PWRs) (Fig. 2)

The primary method for removal of decay heat from PWRs for the first several hours after shutdown is via the steam generators to the secondary system, where the energy is transferred to the secondary system and is rejected as steam to either the turbine condenser or the atmosphere.

PWRs also have alternative means of removing decay heat at high primary system pressure if an extended loss of feedwater is postulated. This method, known as "feed and bleed," uses the high-pressure injection system or charging system to add water (feed) at high pressure to the primary system. As decay heat increases the system pressure, energy is removed (bleed) through the power-operated relief valves (PORV). This method relies on an operable and reliable pressurizer level gauge. Charging pumps can operate at full system pressure. Most PWRs incorporate high-pressure injection pumps that cannot operate at full system pressure (cutoff head about 10 MPa); PORVs can be manually opened,

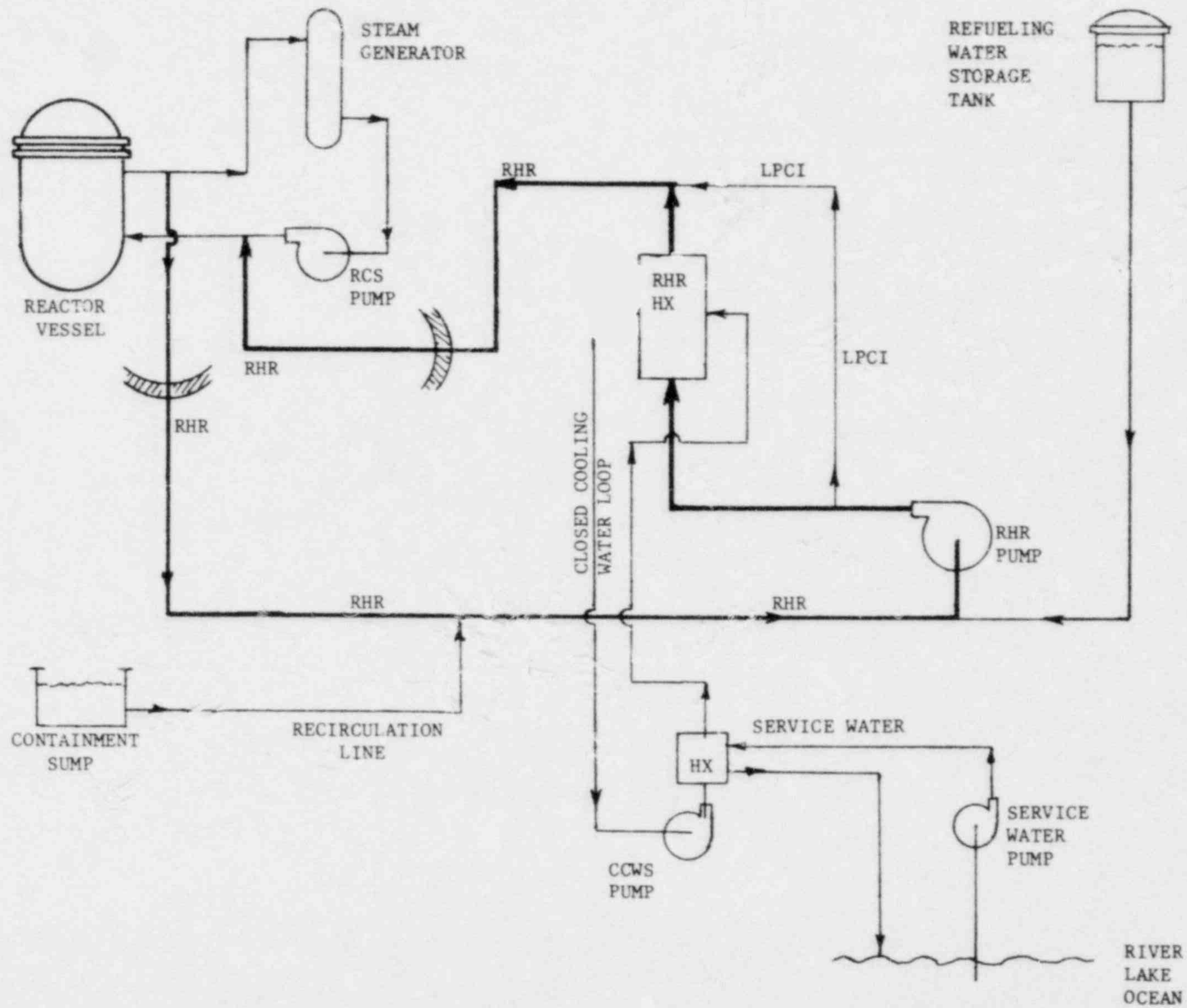


Fig. 2. Example of a PWR RHR system.

thereby reducing the system pressure to within the operating range of the high-pressure injection pumps. Vendor analyses have shown that the core can be adequately cooled by this means.

At low primary system pressure (below about 1.4 MPa) and low temperature, the long-term decay heat in PWRs is removed by the RHR system to achieve cold shutdown conditions. PWR systems designed for normal and postaccident DHR methods are described in Sect. 2.2.

1.3 Study Procedure

This report involves two types of incidents: those involving RHR system integrity and those resulting in partial or complete loss of safety-related components because of failures in the RHR system.

Reports of events by utilities in the form of Licensee Event Reports (LERs) were the major source of information in this report (although other sources mentioned below provided much useful information). A utility is required to submit an LER to the NRC each time a Technical Specification at a plant is violated. Technical Specifications are the plant-specific safety parameters for operation; they can be found in the FSAR for each plant. The FSARs were extensively utilized. NRC Regulatory Guide 1.16 (Ref. 3) defines event reporting requirements.

Computerized reference files of the Nuclear Safety Information Center (NSIC) (containing more than 24,000 LER descriptions plus abstracts of thousands of other operational and licensing documents) were systematically searched for those events associated with RHR. The search of LERs was keyed on the phrases decay heat removal, shutdown cooling, and residual heat removal; those abstracts containing these key phrases were found by this keyword process. The search provided 311 LERs from June 1979 through June 1981 pertaining to RHR systems at U.S. BWRs and PWRs. These LERs formed the backbone of information for this study. Other sources included Inspection and Enforcement (IE) Bulletins, periodic reports of utilities, and NUREG-0020 (Ref. 4). NRC corrective actions evolving from these events were not always available in NSIC files. In some instances, NRC actions were taken of which NSIC was unaware. When the information was available, it was included.

In this report, Chap. 2 describes typical RHR systems for BWRs and PWRs. Chapter 3 discusses particular potentially significant trends and problem areas. Conclusions and recommendations are found in Chap. 4.

Appendixes A through D fully describe and discuss 38 DHR events, all of which are listed in Table 1. The LERs generated by actual or potential RHR system failures with causes within the RHR system are summarized in Appendixes A and B for BWRs (three events) and PWRs (nine events), respectively. Twelve LERs reporting human error involving the RHR system are summarized in Appendix C. Ten LERs reporting DHR events with causes outside the DHR system are summarized in Appendix D. Each of the 38 significant events appears once in Appendixes A through D and once in Appendix E, which is a listing of titles of all 311 LERs found through the NSIC search for the keyword phrases decay heat removal, shutdown cooling, and residual heat removal.

Although Appendixes A and B contain three entries for BWRs and nine entries for PWRs (a 3:1 PWR to BWR event ratio), the total number of significant DHR events in Appendixes A through D is nearly proportional to the numbers of BWRs and PWRs in operation. Twenty-six BWRs and forty-six PWRs are listed in Table 1 (a 1.8:1 PWR to BWR ratio); 16 BWR and 23 PWR events are reported (a 1.4:1 PWR to BWR event ratio). These data indicate that there is very little difference in the frequency of occurrence of significant DHR events at BWRs and PWRs.

Table 1. Listing of reports of safety-significant events by plant

Nuclear power plants[a]	Type of reactor	Architect-engineer (A-E)	Reports of safety-significant events[b]
Arkansas 1	PWR	Bechtel	
Arkansas 2	PWR	Bechtel	
Beaver Valley 1	PWR	Stone & Webster	IE information notice 81-09; 80-46, 80-31, 80-23, 80-22, 80-02
Big Rock Point 1	BWR	Stone & Webster	
Browns Ferry 1	BWR	Tennessee Valley Authority	
Browns Ferry 2	BWR	Tennessee Valley Authority	
Browns Ferry 3	BWR	Tennessee Valley Authority	
Brunswick 1	BWR	United Engineers & Contractors	81-32, 81-005
Brunswick 2	BWR	United Engineers & Conractors	81-59, 81-49, 80-33, 80-30, 80-01, 79-73, 79-50
Calvert Cliffs 1	PWR	Bechtel	
Calvert Cliffs 2	PWR	Bechtel	81-04
Cook 1	PWR	American Electric Power Service	
Cook 2	PWR	American Electric Power Service	
Cooper Station	BWR	Burns & Roe	
Crystal River 3	PWR	Gilbert Associates	
Davis-Besse 1	PWR	Bechtel	IE information notices 80-41 and 80-44; 80-58, 80-49, 80-29, 79-67
Dresden 1	BWR	Bechtel	
Dresden 2	BWR	Sargent & Lundy	
Dresden 3	BWR	Sargent & Lundy	
Duane Arnold	BWR	Bechtel	
Farley 1	PWR	Bechtel & Southern Services	80-57
Farley 2	PWR	Bechtel & Southern Services	
FitzPatrick	BWR	Stone & Webster	
Fort Calhoun 1	PWR	Gibbs & Hill	
Fort St. Vrain	HTGR	Sargent & Lundy	
Ginna	PWR	Gilbert Associates	
Haddam Neck	PWR	Stone & Webster	
Hatch 1	BWR	Southern Services	81-53, 80-62, 80-39
Hatch 2	BWR	Southern Services & Bechtel	
Humboldt Bay	BWR	Bechtel	
Indian Point 2	PWR	United Engineers & Contractors	
Indian Point 3	PWR	United Engineers & Contractors	
Kewaunee	PWR	Pioneer Service & Engineering	
La Crosse	BWR	Sargent & Lundy	
Maine Yankee	PWR	Stone & Webster	
McGuire 1	PWR	Duke Power Company	PNO-II-81-39, 81-72, 81-10
Millstone 1	BWR	Ebasco	
Millstone 2	PWR	Bechtel	
Monticello	BWR	Bechtel	81-02
Nine Mile Point 1	BWR	Niagara Mohawk	

Table 1 (continued)

Nuclear power plants[a]	Type of reactor	Architect-engineer (A-E)	Reports of safety-significant events[b]
North Anna 1	PWR	Stone & Webster	
North Anna 2	PWR	Stone & Webster	
Oconee 1	PWR	Duke Power	
Oconee 2	PWR	Duke Power	
Oconee 3	PWR	Duke Power	
Oyster Creek 1	BWR	Burns & Roe	
Palisades	PWR	Bechtel	81-30
Peach Bottom 2	BWR	Bechtel	81-30
Peach Bottom 3	BWR	Bechtel	
Pilgrim 1	BWR	Bechtel	
Point Beach 1	PWR	Bechtel	
Point Beach 2	PWR	Bechtel	
Prairie Island 1	PWR	Pioneer Service & Engineering	
Prairie Island 2	PWR	Pioneer Service & Engineering	
Quad Cities 1	BWR	Sargent & Lundy	81-07
Quad Cities 2	BWR	Sargent & Lundy	
Rancho Seco 1	PWR	Bechtel	81-24
Robinson 2	PWR	Ebasco	
Salem 1	PWR	Public Service Electric & Gas	79-59
San Onofre 1	PWR	Bechtel	
Sequoyah 1	PWR	Tennessee Valley Authority	81-21
St. Lucie 1	PWR	Ebasco	81-29
Surry 1	PWR	Stone & Webster	
Surry 2	PWR	Stone & Webster	
Three Mile Island 1	PWR	Gilbert Associates	
Three Mile Island 2	PWR	Burns & Roe	
Trojan	PWR	Bechtel	81-12
Turkey Point 3	PWR	Bechtel	
Turkey Point 4	PWR	Bechtel	
Vermont Yankee 1	BWR	Ebasco	
Yankee-Rowe 1	PWR	Stone & Webster	
Zion 1	PWR	Sargent & Lundy	
Zion 2	PWR	Sargent & Lundy	

2. DESCRIPTION OF THE GENERAL PLANT SYSTEMS TO REMOVE DECAY HEAT

This chapter describes normal ar/ postaccident DHR methods. The phrase residual heat removal is used to designate systems and components performing the general functions of DHR. These RHR system descriptions were obtained from various sources and do not apply specifically to any particular design. Significant exceptions to these general designs are noted where appropriate.

2.1 The BWR System to Remove Decay Heat

The BWR system to remove residual heat (Fig. 1) can operate in several modes as follows (not all BWRs have all the following modes):

1. shutdown cooling and reactor head spray,
2. steam condensing,
3. suppression pool cooling,
4. suppression pool drain to radwaste,
5. LPCI,
6. containment spray — dry well and suppression pool,
7. standby cooling supply, and
8. fuel pool cooling.

The purposes of the RHR are (1) to provide the capability to remove decay heat and sensible heat from the primary system so that the reactor can be shut down during a normal refueling and servicing operation and after accidents, (2) to provide and maintain an inexhaustible source of makeup water for vessel and containment flooding so that the core is adequately cooled, and (3) to limit the suppression pool water temperature and pressure.

The following paragraphs describe BWR operations involving the above eight RHR modes: first during normal shutdown, then during reactor vessel isolation, then during small-break LOCAs, and last during large-break LOCAs. Following this description of operations is a general description of each of the eight modes.

In a normal shutdown, power is reduced and steam from the low-pressure turbine is condensed in the main condenser. The turbine is tripped at less than 30% power, and steam for shaft sealing and steam jet air ejectors is supplied by an auxiliary boiler. Main steam is bypassed via the turbine bypass valves to the main condenser. Reactor pressure in a normal shutdown may be reduced (to about 0.7 MPa) by manually operating the safety/relief valves and venting main steam to the suppression pool, thus enabling the RHR shutdown cooling and reactor head spray mode (No. 1) to be functional. Reactor power is reduced to source range by inserting rods or scramming the reactor at about 20% power. Controlled cooling, at up to 38°C/h, continues using RHR mode 1.

If the reactor vessel is isolated [main steam isolation valves (MSIVs) closed, making the main condenser unavailable] or if feedwater is lost, then the steam condensing mode (No. 2) is used to control the pressure in the reactor until it is depressurized to a level where the shutdown cooling mode becomes operable. Mode 2 quenches main steam in the RHR heat exchangers and is also used to maintain a hot shutdown condition.

A related cooling system, though separate from and independent of RHR, is the reactor core isolation cooling (RCIC) system, a shutdown cooling system that performs under the same conditions as the steam condensing mode of the RHR system. RCIC is designed to provide adequate core cooling in the event of reactor isolation accompanied by loss of feedwater flow. RCIC also provides core cooling during normal reactor shutdown under conditions of loss of normal feedwater system by maintaining sufficient reactor water inventory until the reactor is depressurized to a level where the shutdown cooling mode becomes operable. Use of the RCIC system causes the suppression pool temperature to rise because the RCIC turbine discharges its steam to the suppression pool. When RCIC is operating, RHR is usually operated in the suppression pool cooling mode (No. 3) to keep the suppression pool temperature below about 35°C. When the RCIC is used during a vessel isolation, reactor steam is relieved to the suppression pool via the safety/relief valves, which both heat up the suppression pool and increase its level. In this case, also, the RHR system is required to operate in modes 3 and 4, suppression pool drain to radwaste. In these ways, then, the RCIC system supplements, relies on, and interacts with several modes of RHR operation. The isolation condenser (IC) on early BWRs serves the same function as RCIC (Table 2).

Shutdown cooling during small-break LOCA conditions is provided by the HPCI system. If vessel water level cannot be maintained using HPCI (and, if possible, RCIC), then the backup high-pressure ECCS ADS will initiate to reduce vessel pressure so that the low-pressure ECCSs can operate.

Shutdown cooling during large-break LOCA conditions is provided by two low-pressure, high-capacity ECCSs: LPCI and core spray, which initiate to flood the vessel and prevent core melting. LPCI is the priority mode of the RHR system (mode 5). Core spray is a separate system altogether, independent of RHR. The core spray system is discussed following the descriptions of the eight RHR modes.

Note that the ECCS's HPCI, LPCI, and core spray are not used during normal reactor startups or shutdowns; these systems are too large for vessel level control. Their purpose is entirely the removal of sensible and decay heat during a LOCA. During a design-basis LOCA, the suppression pool and dry well heat up and pressurize because of steam and coolant blowdown from the break.

To maintain containment integrity, RHR mode 6, containment spray — dry well and suppression pool, is provided to reduce containment pressure. In the unlikely event of loss of containment integrity and/or loss of RHR system function, RHR mode 7, standby cooling supply, is provided to supply raw water directly to the core, dry well, and/or suppression pool through the service water pumps.

The power supplied for the RHR pumps is usually the 4160-V shutdown boards. The shutdown boards are supplied by normal auxiliary power if available or by the diesel assigned to the bus. The buses are automatically energized by the diesels in the event of loss of normal auxiliary power.

Table 2. Variations in BWR design classes

Plant name	BWR design class	LPCI configuration	IC or RCIC
Oyster Creek	2	*a*	IC
Nine Mile Point 1	2	*a*	IC
Dresden 2 and 3	3	*b*	IC
Millstone	3	*b*	RCIC
Monticello	3	*c*	RCIC
Quad-Cities 1 and 2	3	*c*	RCIC
Pilgrim	3	*c*	RCIC
Browns Ferry 1, 2, and 3	4	*c*	RCIC
Vermont Yankee	4	*c*	RCIC
Duane Arnold	4	*c*	RCIC
Peach Bottom 2 and 3	4	*c*	RCIC
Cooper	4	*c*	RCIC
Hatch 1 and 2	4	*c*	RCIC
Brunswick 1 and 2	4	*c*	RCIC
FitzPatrick	4	*c*	RCIC
Nine Mile Point 2	5	*d*	RCIC

[a]Unique designs.

[b]No RHR head spray mode; head spray is supplied directly from the control rod drive pumps.

[c]Equipped with two RHR loops; LPCI mode injects to the recirculation system.

[d]Equipped with three RHR loops; LPCI mode injects directly to the vessel.

A minimum flow bypass line protects the RHR pumps from overheating when RHR is isolated by routing water from the pump discharge to the suppression pool. A single motor-operated valve controls the condition of each bypass line. The minimum flow bypass valve automatically opens upon sensing low flow in the discharge line from the associated pump. The valve automatically closes whenever the flow from the associated pump is above the low flow setting.

The major equipment for all RHR modes consists of heat exchangers, RHR pumps, and RHR service water pumps to cool the RHR heat exchangers. A general description of each RHR mode is covered in the next eight sections.

2.1.1 Shutdown cooling and reactor head spray

Shutdown cooling and reactor vessel head spray mode of RHR is an integral part of the RHR system and is placed in operation during normal shutdown and cooldown. The initial phase of nuclear system cooldown is accomplished by dumping steam from the reactor vessel to the main condenser. When nuclear system pressure has decreased sufficiently (to about 0.7 MPa), the RHR is placed in the shutdown cooling mode of operation. The shutdown cooling mode alone is capable of completing cooldown to 52°C in less than 20 h and maintaining the nuclear system at 52°C so that the reactor can be refueled and serviced.

Reactor coolant is pumped by the RHR pumps from one of the recirculation loops through the RHR heat exchangers, where cooling takes place by transferring heat to the RHR service water. Reactor coolant is returned to the reactor vessel via the recirculation loop. Part of this flow may be diverted to a spray nozzle in the reactor head. This spray condenses steam being generated by the hot reactor vessel walls, vessel internals, and decay heat, thus reducing vessel pressure. This ensures that the water level in the reactor vessel can rise. The higher water level provides conduction cooling to more of the mass of metal of the reactor vessel and therefore limits thermal stress in the reactor vessel during cooldown.

2.1.2 Steam condensing

In the RHR steam condensing mode, steam from the reactor is admitted into the RHR heat exchangers where it is condensed by the RHR service water. Condensate from the heat exchanger returns to either the torus or the RCIC pump suction. The later BWR/4 plants were the first with the steam condensing mode.

Most BWR plants have a shutdown cooling system, called the RCIC system, which is separate from the RHR system but operates under the same conditions as the steam condensing mode of RHR. The RCIC system is designed to provide adequate core cooling in the event of reactor vessel isolation (MSIVs closed) accompanied by loss of feedwater flow. RCIC also provides core cooling during normal reactor shutdown accompanied by loss of feedwater by maintaining sufficient reactor water inventory until the reactor is depressurized to a level where the shutdown cooling system becomes operable. Although RCIC is used during a LOCA, it is not an ECCS; its primary purpose is to supply water during hot standby, when MSIVs are closed. Steam produced by decay heat during such an isolation is relieved via the safety and relief valves to the suppression pool, and reactor water level is maintained using the RCIC system.

The RCIC system has small turbine-driven pumps, providing 25 to 38 L/s at full reactor pressure to either the feedwater line (BWR/3 and 4) or the head spray (BWR/5 and 6). The primary source of water to RCIC is the condensate storage tank; the secondary source is the suppression pool or the condensate in the RHR heat exchanger when the system is operated in the steam condensing mode.

In BWR/1 and 2 and early BWR/3, an RCIC system was not provided. For these cases, an IC was provided as a passive backup means for removing decay heat at high RCS pressure. The IC serves the same function as RCIC on later BWRs.

2.1.3 Suppression pool cooling

The suppression pool cooling mode of RHR limits the temperature of the water in the suppression pool so that immediately after the design-basis accident has occurred, pool temperature does not exceed about

77°C. In the suppression pool cooling mode, the RHR pumps are aligned to pump water from the suppression pool through the RHR heat exchangers where cooling takes place by transferring heat to the RHR service water. The flow returns to the suppression pool via the full flow test line.

2.1.4 Suppression pool drain to radwaste

The RHR system has flush and drain piping to permit flushing and filling the system with CST water. This piping can be used on most BWRs to adjust the suppression pool level (lowering the level serves to lower the torus pressure, which helps maintain suppression pool integrity during design-basis LOCA conditions, supplementing the containment spray mode of RHR). This procedure is therefore considered an RHR mode itself, called suppression pool drain to radwaste.

2.1.5 Low-pressure coolant injection

The LPCI subsystem is the dominant mode and normal valve lineup configuration of RHR. All other modes of RHR are submissive to LPCI, and the RHR will automatically align to the LPCI mode when ECCS initiation signals are sensed. LPCI is a low-pressure system.

During LPCI operation, the RHR pumps take suction first from the CST and discharge through the RHR heat exchangers to the reactor vessel via the recirculation loops on BWR/3-5 and on BWR/6 via a separate vessel penetration and a core shroud penetration directly into the core. Any spillage through a break in the lines within the primary containment returns to the suppression pool through the pressure suppression vent lines.

Service water flow through the RHR heat exchangers is not required immediately after a LOCA because heat rejection from the containment is not necessary during the time it takes to flood the reactor.

2.1.6 Containment spray — dry well and suppression pool

The water pumped through the RHR heat exchanger in the suppression pool cooling mode of RHR may be diverted to spray headers in the dry well and above the suppression pool. This is the containment spray mode of RHR, designed for postaccident operation to maintain containment integrity by reducing containment pressure. The spray headers in the dry well condense any steam that may exist in the dry well, thereby lowering containment pressure. The spray collects in the bottom of the dry well and drains back to the suppression pool. Approximately 5% of this flow may be directed to the suppression chamber spray ring to cool any noncondensible gases collected in the free volume above the suppression pool, thereby lowering torus pressure.

The spray headers of the RHR cannot be placed in operation if an LPCI auto-initiation signal is present, except under certain specific conditions (see individual plant Technical Specifications).

2.1.7 Standby coolant supply

Standby coolant supply connection and RHR crossties are provided to maintain a long-term reactor core and primary containment cooling capability independent of primary containment integrity or operability of the RHR system associated with a given unit. By proper valve alignment, the network created by the standby coolant supply connection and RHR crossties permits the RHR service water pumps and headers to supply raw water directly to the reactor core below 0.34 MPa. The service water pump and header can also be valved to supply raw water to the dry well or suppression chamber spray headers or directly to the suppression chamber of either unit.

2.1.8 Fuel pool cooling mode

The capacity of the fuel pool cooling system is based on normal refueling requirements. Provisions within RHR (primarily installation of a removable spool piece in most BWRs) allow the RHR pumps to move water from the fuel pool cooling system through the RHR heat exchangers and back to the fuel pool cooling system. This capability ensures fuel pool cooling capacity across the entire decay heat load spectrum.

2.1.9 Core spray

The core spray system is a low-pressure ECCS that operates independently of any mode of the RHR system. Core spray consists of two independent full-capacity loops that remove decay heat in postaccident low-pressure conditions. Core spray pumps take suction from the suppression pool and discharge to the reactor vessel through spray nozzles in ring spargers located within the inner shroud of the reactor vessel, directly above the fuel assemblies. BWR/3 and 4 have two low-pressure core spray (LPCS) spargers, and BWR/5 and 6 have one LPCS sparger as well as an HPCS system with electric motor driven pumps, a dedicated diesel, and an independent sparger. HPCS replaces the turbine-driven HPCI systems in earlier BWRs.

2.2 The PWR Decay Heat Removal System

The RHR system in PWRs (Fig. 2) is designed to remove decay and sensible heat from the reactor core and reduce the temperature of the reactor coolant system during the second phase of plant cooldown. During the first phase of plant cooldown, the reactor coolant temperature is reduced by transferring heat from the reactor coolant system to the steam and power conversion system (i.e., the main condenser via the steam generators).

Portions of the RHR system are used as LPCI, which is an ECCS, during accident conditions. The RHR pumps are powered from the engineered safety features electrical buses.

The RHR system also transfers refueling water between the refueling water storage tank (RWST) and the refueling cavity before and after refueling operations.

Generally, while at cold shutdown condition, residual heat from the reactor core is being removed by the RHR system. The number of pumps and heat exchangers in service depends on the RHR load at the time.

To ensure reliability, the pumps are connected to separate electrical buses so that each pump receives power from a different source. If a total loss of preferred power occurs while the system is in service, each bus is automatically transferred to a separate emergency diesel power supply.

2.2.1 RHR operation

To begin RHR operation, pressure-interlocked isolation valves are opened to connect the RHR pump suction to one of the reactor coolant system hot legs. Coolant flows from the reactor coolant system to the RHR pumps, through the RHR heat exchangers, and back to the reactor coolant system via cold legs. The residual heat exchangers are cooled either by a closed cooling water system or by raw water from the service water system circulating through the shell side of the RHR heat exchangers. The RHR system is generally placed in operation 4 to 8 h after reactor shutdown when the temperature and pressure of the reactor coolant system are approximately 177°C and 2.8 MPa, respectively. The RHR system is designed to reduced the temperature of the reactor coolant from 177 to 60°C in about 20 h. Steam generators steaming to the main condenser are often used in this temperature range to accelerate cooldown.

After the reactor coolant system has been cooled and depressurized, the RHR system is continuously operated to dissipate decay heat and to ensure thorough coolant mixing for reactivity and plant chemistry considerations.

2.2.2 ECCS operation

The RHR system is used under accident conditions as the low-head high-capacity portion of the ECCS and is usually referred to as LPCI. At reactor coolant system temperatures above 177°C, the RHR system is aligned for ECCS operation. The suction valves from the RWST are opened to supply 2000 ppm borated water to the suction of the RHR pumps. All pump suction valves, heat exchanger throttle valves, and header isolation valves are aligned during power operation so that an injection path is provided from the RHR pumps discharge, optionally through the RHR heat exchangers, and into the cold legs. Bypass piping around the RHR heat exchangers is usually installed to allow low-pressure injection flow to bypass the heat exchangers. With the standard alignment, all that is necessary for low-pressure injection initiation is that the pumps start when reactor coolant system pressure decreases below the discharge pressure of the RHR pumps.

After the RWST has been emptied during the injection phase, the RHR system has the capability to take suction from the containment recirculation sump (except in Combustion-Engineering plants, in which the high-pressure injection pumps take over the low-pressure injection function). When the RWST has been emptied, high-pressure injection takes suction from the containment sump and low-pressure injection is isolated. The RHR system cools and recirculates the spilled reactor coolant and RWST water for long-term cooling of the reactor core after a large-break LOCA. This long-term cooling is designated the recirculation phase.

3. DISCUSSION OF POTENTIALLY SIGNIFICANT EVENTS

The safety significance of all DHR events was estimated. In general, a significant event is one that prevents or could have prevented the RHR system from fulfilling its design criteria. Of the 311 reports entered on the NSIC computer file with the DHR-related keyword phrases and dated between June 1979 and June 1981, 38 were selected as being potentially significant and are discussed in Appendixes A through D. Events were selected using the best engineering judgment of NSIC personnel to determine which events are potentially significant.

Safety-significant events in the RHR system are those events involving RHR system integrity which (1) cause failure of the system to perform its designed function and (2) result in partial or complete loss of safety-related components (in other systems) because of failures in the RHR system.

In the following discussion of specific events, numbers in parentheses represent Appendix and event numbers where more detailed discussions can be found (e.g., B.8 is Appendix B, event 8).

3.1 Concentration of DHR Events at Four Plants

Fifty-five percent or 21 of the 38 total potentially significant DHR events occurred at just 4 of the 72 operating plants. Table 1 vividly illustrates this concentration of events at these four plants: Beaver Valley 1, Brunswick 2, Davis-Besse 1, and McGuire 1. McGuire's record of three DHR events in the spring of 1981, the first months of its initial fuel loading, merits its inclusion in this group of plants.

Four of the six events at Beaver Valley 1 involved air-bound RHR pumps: two occurred while changing the RHR flow rate (B.8, B.9), one involved use of the reactor vessel vent eductor system (D.9), and one involved a failed reactor vessel water level indicator. The other two events at Beaver Valley 1 involved a false high-pressure signal that isolated both RHR loops (B.7) and a water hammer during the normal operating procedure of throttling the component cooling water (CCW) to the RHR heat exchangers (B.6). No trend or particular design error is found in these events, though plant engineers as of January 1980 were investigating a design modification to continuously vent the RHR pumps (D.9).

Three of the five events at Brunswick 2 were caused by human error: the wrong breaker was opened (C.10), the wrong pump was removed from service (C.12), and the reactor was incorrectly started with one LPCI loop inoperable in the recirculation mode (C.1). The other two events at Brunswick were RHR heat exchanger failures caused by oyster shell buildups in the service water side (D.2) and failure of redundant RHR coolers for the LPCI inverter room (D.8).

Four of the six events at Davis-Besse 1 involved failures relating to the safety features actuation system (SFAS). In three of these events (B.4, D.4, D.7), when RHR pumps were aligned to an empty sump, cavitation of RHR pumps resulted. In the fourth event, the RHR system was isolated because of an SFAS actuation (D.10). The other two events at Davis-Besse 1 were a check valve failure between the reactor coolant system and DHR systems (B.5) and a series of events involving loss of DHR flow (C.9).

The three events at McGuire 1 in 1981 included a broken air line, which caused the RHR to be throttled to 50% required flow (B.3); an SFAS signal, which isolated the RHR from the RCS (C.3); and a steam bubble under the reactor vessel head during cooldown (D.1, see Sect. 3.2.4).

3.2 RHR Pump Cavitation

A most frequent event causing a problem with the removal of decay heat was the cavitation of RHR pumps. Nine instances of this problem were reported in the 2-year period considered here. Less significant instances of RHR pump cavitation occur but are not reported in LERs and are not considered here.

Seven of these nine events occurred at Beaver Valley 1 and Davis-Besse 1, the two plants with the most events as reported herein. The other two events occurred at Trojan and Salem 1. At Beaver Valley 1, two instances of cavitation occurred because of mechanical failures (B.2, B.7) and two occurred in part because the particular design required more attentive procedures than were in use at the time (B.8, D.9). At Davis-Besse 1, three events occurred in which DHR pumps were aligned to a dry sump in the containment because of an SFAS actuation, resulting in no DHR suction (B.4, D.4, D.7). The events at Trojan and Salem 1 occurred because the reactor coolant system level dropped too low during reactor shutdown and the operating RHR pumps lost suction (C.2, C.11).

At Davis-Besse 1, the two-out-of-four SFAS logic is being examined for possible modification. More complete and explicit procedures are being recommended in most of these cases. It may be significant that at Davis-Besse no RHR pump cavitation events were reported in 1981 through June. Possibly these procedures are working.

3.3 SFAS-Initiating RHR System Failures

Seven significant events occurred in which the SFAS played a major role in causing the degradation or failure of the RHR system; five of these occurred at Davis-Besse 1. Four of these seven events led to cavitation of the RHR pumps because the SFAS caused the RHR to align to the recirculation mode when the containment sump was dry [three events (see Sect. 3.1) at Davis-Besse 1 (B.4, D.4, D.7) and the fourth at Beaver Valley (D.9)]. The fourth and fifth events involved the DHR system at Davis-Besse 1, which was isolated because of a loss of an essential bus supplying the SFAS (D.10), and the RHR system at McGuire 1, which was isolated because of an inadvertent reactor protection system signal to a reactor coolant system to RHR discharge isolation valve (C.3). The seventh SFAS-initiated RHR event occurred at Davis-Besse 1, where an RHR suction valve was aligned to no suction while maintenance was being performed (C.9); this event was one of three similar events within 10 d at Davis-Besse 1.

Davis-Besse 1 reported five events that involved SFAS-initiated DHR. IE Bulletin 80-12 required all PWR licensees to review their equipment and procedures relative to the DHR loss at Davis-Besse 1 on April 19, 1980 (D.7). Three factors contributed to this event: inadequate administrative control of valve alignment and SFAS logic during refueling, extensive and poorly coordinated maintenance activities, and two-out-of-four SFAS logic being served by one power source.

3.4 Steam Bubbles in PWR Reactor Vessel

Two events in the last 2 years occurred in which a steam bubble formed in the head area of a PWR reactor vessel (Fig. D.1) during DHR operation. In neither case did it restrict natural circulation cooling of the core, though this is the major safety concern. At McGuire 1 during a reactor cooldown prior to initial criticality, a steam bubble formed in the reactor vessel head when the reactor coolant system was vented with a reactor vessel head temperature of 121°C. The vent was closed and the bubble collapsed. The bubble formed again 8 h later, and the reactor coolant system was repressurized to permit operation of the reactor coolant pumps to sweep out the steam. In both cases, the cooldown of the head area lagged behind that of the rest of the reactor coolant system (D.1).

A more severe situation existed at St. Lucie 1 on June 11, 1980. The reactor was being cooled by natural circulation cooling following a shutdown from full power when a steam bubble formed because of the rapid depressurization in the reactor head area. Again the cause was a temperature lag in the head (D.6). Plant operators did not expect the bubble and therefore did not immediately recognize it. However, the reactor was brought to a cold shutdown after two leakage paths from the reactor coolant system to the RWST were discovered and isolated and the reactor coolant system pressure was increased sufficiently to collapse the bubble.

3.5 Seismic Design Deficiencies

Seismic design deficiencies were the most commonly reported failures, accounting for 36 of the 313 reports considered (Table 3). Most of these LERs resulted from reanalyses or tests in response to either IE Bulletin 79-02 or 79-14. None of these seismic design deficiencies, however, caused failures. For this reason, and because seismic designs have been covered in other regulatory initiatives, the seismic design DHR systems will not be discussed further in this report.

Table 3. Seismic design deficiencies

Abstract No.	Plant	LER No.	Description
34	Rancho Seco 1	81-10	DHR pipe stress exceeds limit (IE Bulletin 79-1 reanalysis)
46	Surry 1	80-10R	RHR pipe support found inadequate (reanalysis)
92	Turkey Point 4	80-14	RHR pipe support deficient (IE Bulletin 79-14 reanalysis)
106	Sequoyah 1, 2	80-180	Concrete block walls not seismically qualified (reanalysis)
110	Quad-Cities 1	80-27	Construction deficiency (IE Bulletin 79-14 reanalysis)
119	Pilgrim 1	80-67	RHR not designed to standard (utility reanalysis)
122	Peach Bottom 3	80-16	Snubber not up to design specification (A-E reanalysis)
134	Beaver Valley 1	80-67	CCW line inadequately supported (reanalysis)
144	Hatch 1	80-76	Pipe supports inadequate (A-E reanalysis)
145	Beaver Valley 1	80-49	CCW line inadequately supported (IE Bulletin 79-14 reanalysis)
175	Rancho Seco 1	80-31	DHR discharge pipe support inadequate (IE Bulletin 79-14 reanalysis)
178	San Onofre 1	80-19	RHR pump cooling pipe support missing (IE Bulletin 79-14 inspection)
203	Three Mile Island 1	80-14	DHR pipe underdesigned (reanalysis)
224	Hatch 2	80-12	RHR miniflow lines inadequately supported (reanalysis)
230	North Anna 1	80-02	Incorrect valve weights used (A-E reanalysis)
237	Surry 1	80-10	RHR discharge overstressed design (reanalysis)
239	Rancho Seco 1	80-02	DHR pipe supports underdesigned (IE Bulletin 79-14 reanalysis)
249	FitzPatrick	79-103	Thermal growth makes pipe hanger inoperable (A-E reanalysis)
252	Browns Ferry 1	79-32	Service water pipe to RHR heat exchanger unsupported (IE Bulletin 79-14 reanalysis)
253	Monticello	79-21	RHR snubber anchor bolts inadequate (IE Bulletin 79-02 reanalysis)
256	FitzPatrick	80-07	RHR pipe hanger inadequate (IE Bulletin 79-02 reanalysis)
268	Three Mile Island 1	79-17	DHR pipe support inadequate (IE Bulletin 79-14 reanalysis)
271	Robinson 2	79-39	RHR pipe supports inadequate (IE Bulletin 71-14 reanalysis)
272	Trojan	79-15	Weak wall holding engineered safety feature piping (reanalysis, see abstract 106)
274	FitzPatrick	79-87	RHR service water pipe supports inadequate (A-E reanalysis)
284	FitzPatrick	79-81	RHR pipe supports inadequate (A-E reanalysis)
286	Zion 1	79-68	RHR hot-leg injection pipe anchor inadequate (IE Bulletin 79-14 reanalysis)
288	FitzPatrick	79-57	RHR and core spray pipe supports inoperable (A-E reanalysis)
290	Connecticut Yankee	79-08	RHR seismic support missing (IE Bulletin 79-02 reanalysis)
296	Peach Bottom 2	79-43	Two HPSW pipe anchors inadequate (IE Bulletin 79-02 reanalysis)
302	Peach Bottom 3	79-24	RHR piping anchor bolts fail test (IE Bulletin 79-02 reanalysis)
306	Rancho Seco 1	79-07	DHR anchor bolts inadequate (IE Bulletin 79-02 reanalysis)
307	Hatch 1	79-64	RHR pump anchor bolts fail (RHR service water pumps modification)
314	Peach Bottom 3	79-23	Anchor bolts fail torque test (IE Bulletin 79-02 test)
319	Brunswick 1	79-42	RHR snubber support beam inadequate (IE Bulletin 79-07 reanalysis)
321	Pilgrim 1	79-22	HPCI and RHR piping supports inadequate (IE Bulletin 79-02 reanalysis)

3.6 RHR Minimum Flow Recirculation Line at Zion 1 and 2

Four failures of the isolation valve on the RHR minimum flow line were reported at Zion 1 and 2 (abstracts 12, 28, 36, and 87 in Appendix E). In each case, the micro switch on the valve was at fault and the valve failed a test to close, leading to a concern that during a LOCA the LPCI flow to the core would be reduced by the 10% that would recirculate through the failed-open valve. The faulty switch was replaced each time.

3.7 Check Valve Leaks and Failures

Leaking and failed check valves caused or were major factors in six of the significant events in this study (Table 4). At Davis-Besse 1, a check valve that became disassembled was the subject of Information Notice 80-41 to all licensees. The steam bubble event at St. Lucie was exacerbated by a leaking check valve (D.6).

In addition to these six significant check valve failures, several entries in Appendix E reported leaking check valves. This recurring problem was noted also in Task 3 on service water systems.

Table 4. Check valve leaks and failures

Appendix No.	Abstract No.	Plant	Event report	Description
A.1	29	Monticello	81-02	Check valve leak between RHR seal water loops
B.5	44	Davis-Besse 1	IE Information Notice 80-41	Check valve breaks between RCS and RHR system
	109	Kewaunee	80-26R	Check valve leak partially drains RWST
	115	Hatch 2	80-148	Check valve leak partially drains torus
	153	Browns Ferry 2	80-35	Check valve leak on connecting line between RHR heat exchangers
D.6	NA	St. Lucie 1	80-29	Check valve leak opens RCS to BWST

4. CONCLUSIONS AND RECOMMENDATIONS

The following conclusions and recommendations are drawn from this study of operating experiences. Note that this study was completed using only information that was placed in the NSIC file from June 1979 through June 1981, supplemented with other readily available information at NSIC. It was not possible to delve deeply into the events. However, a general conclusion can be drawn. A careful study of the information available on these potentially significant LERs involving DHR showed that no serious problems developed as the result of the events. In each case, decay heat removal was reestablished well before any serious problem could develop.

DHR events were found to be concentrated at just 4 of the 72 operating plants: Beaver Valley 1, Brunswick 2, Davis-Besse 1, and McGuire 1. Causes of the events at McGuire were not related, so no recommendation can be made for that plant. However, the event causes at the other three plants were often related, and conclusions and recommendations regarding these plants are part of the following items.

1. The most frequent event involving a significant problem with DHR was the cavitation of RHR pumps. Beaver Valley 1 experienced four significant RHR pump cavitations and Davis-Besse 1 experienced three. The causes of these events varied and followed no general trend (except at Davis-Besse 1, see below); therefore, no explicit corrections are recommended regarding RHR pump cavitations. In general, more complete and explicit procedures should be developed to ensure RHR availability.

2. Inadvertent emergency safety features actuations played a significant role in seven DHR events (five at Davis-Besse 1), often by aligning an operating RHR pump to an empty sump, resulting in RHR pump cavitation. The emergency safety features logic needs tighter administrative control (i.e., defeating or bypassing logic) during cold shutdowns and refueling. A review of any logic that allows RHR alignment to an empty sump is advisable.

3. Steam bubbles in the reactor vessel head threatened to inhibit natural circulation twice in these 2 years studied. Because

at St. Lucie the operators did not immediately recognize the situation, it is recommended that operators become aware of the possibility of bubble formation during natural-circulation cooldowns. Procedures should be developed to guide operators in responding to this situation. Methods to reduce the temperature variations in the reactor vessel should be pursued.

4. Procedures and tests are often directed to component unavailability. To ensure higher RHR availability, procedures and tests should be directed to flow path unavailability as well as component unavailability.

REFERENCES

1. U.S. Nuclear Regulatory Commission, *Identification of New Unresolved Safety Issues Relating to Nuclear Power Plants*, NUREG-0705 (March 1981).

2. L. E. Anderson et al., *Post-LOCA Long Term Cooling Evaluation Model*, CENPD-254P, Combustion Engineering, Inc. (June 1972).

3. U.S. Nuclear Regulatory Commission, *Reporting of Operating Information — Appendix A, Technical Specifications*, Regulatory Guide 1.16.

4. U.S. Nuclear Regulatory Commission, *Licensed Operating Reactors Status Summary Report*, NUREG-0020. (No one particular report in the NUREG-0020 series was used.)

Appendix A

DESCRIPTION OF EVENTS WITH CAUSES WITHIN THE DECAY HEAT REMOVAL SYSTEM — BWRs

Appendix A contains detailed descriptions of the following events:

A.1 Single Check Valve Failure Stops Both RHR SW Loops at Monticello (LER 81-02)

A.2 Operation With Potential Common-Mode Failure at Hatch 1 (LER 80-53)

A.3 RHR SW Pump Motor Cooling Does Not Meet Single Failure Criterion at Hatch 1 (LER 80-39)

A.1 Single Check Valve Failure Stops Both RHR SW Loops at Monticello

(LER 81-02, Abstract 29, March 3, 1981)

During a test of the No. 14 RHR service water (SW) pump, engineers discovered that the RHR SW was pressurizing the seal water supply system, which supplies seal water between inner and outer packing of the pumps in both loops. Two failures were responsible: (1) the No. 14 RHR SW pump shaft inner packing failed because of natural end of life, allowing pressurization of the seal water system, and (2) the check valve on the loop B seal water supply failed, allowing pressurization to extend to both loops. The pump packing and the check valve were replaced. Also, the seal water supply was modified to provide a check valve on the supply to each RHR SW pump.

The significance of this event is in the recurring check valve failure (Table 4). Task 3 of this project, on failures of service water systems, also found the performance of check valves to be generally unreliable (and recommended that testable check valves be installed where valve performance is essential to safety or where it is problematic).

A.2 Operation With Potential Common-Mode Failure at Hatch 1

(LER 80-53, Abstract 185, May 24, 1980)

The architect-engineer, Southern Services, Inc., discovered that Unit 1 had been operating within a single failure criteria region. A review of LER No. 80-41 revealed that the loss of power to essential motor control center 1B along with a recirculation discharge line break (a design-basis accident) would render both loops of RHR and the A loop of core spray inoperable. The A RHR inboard injection valve was leaking, pressurizing the RHR heat exchanger. At that time, closing the RHR outboard injection valve was chosen as an acceptable solution. The affected procedure was revised and the valve closed. After the discovery of the event, the unit was shut down, the valve repaired, and the procedure revised.

A.3 RHR SW Pump Motor Cooling Does Not Meet Single Failure Criterion at Hatch 1*

(LER 80-39, Abstract 192, April 11, 1980)

The cooling water supply line to the plant and RHR service water pump motors had a single pressure regulator that, if failed, would terminate cooling water to these pump motors, potentially rendering them inoperable. This event is significant because the failure of this single pressure regulator would be a common-cause failure for the plant and RHR pumps.

The initial design, by Southern Services, Inc., did not specify a divisional cooling water supply to these motors. A modification was made to create a divisional supply.

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*This discussion also appears in Appendix A of Task 3 on service water systems.

Appendix B

DESCRIPTION OF EVENTS WITH CAUSES WITHIN THE DECAY HEAT REMOVAL SYSTEM — PWRs

Appendix B contains detailed descriptions of the following events:

B.1 Unexpected Heatup While in Cold Shutdown at Palisades (LER 81-30)

B.2 RHR Pumps Cavitated Because of Low RCS Level at Beaver Valley 1 (IE Information Notice 81-09)

B.3 Broken Air Line on RHR Valve at McGuire 1 (LER 81-10)

B.4 ECCS Actuation Causes RHR Pump Cavitation at Davis-Besse 1 (IE Information Notice 80-44)

B.5 Check Valve Failure in RCS to RHR line at Davis-Besse 1 (IE Information Notice 80-41)

B.6 CCW Piping to RHR Heat Exchangers Embedment Plate Bows at Beaver Valley 1 (LER 80-46)

B.7 RHR Fails Because of False High-Pressure Signal at Beaver Valley 1 (LER 80-31)

B.8 Total RHR Flow Lost Because of Air-Bound Pumps at Beaver Valley 1 (LER 80-22)

B.9 Total RHR Flow Lost Because of Air-Bound Pumps at Beaver Valley 1 (LER 80-23)

B.1 Unexpected Heatup While in Cold Shutdown at Palisades*

[LER 81-30 (not included in Appendix E), July 15, 1981]

While the plant was operating in a shutdown cooling mode and the primary system level was drained near the hot-leg centerline for replacement of a reactor coolant pump (RCP) seal package, a loss of shutdown cooling capability occurred because of isolation of the single shutdown

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*This event is a late addition to this report; although it occurred after the cutoff date, it is included because of its potential generic safety significance.

heat exchanger outlet control valve (CV-3025). The outlet valve malfunctioned because of water accumulation in its control air system. This failure was not immediately evident because only the demand position for the valve is indicated in the main control room. The primary coolant temperature rose about 21°C, reaching a maximum temperature of 91°C over a 1.5-h period while the plant staff diagnosed and corrected the problem.

Figure B.1 shows the susceptibility of the shutdown cooling systems to (1) a single failure of the heat exchanger outlet valve (CV-3025), (2) a single failure of either of the common loop suction valves (MOVs-3015 or 3016), or (3) a single failure of the common heat exchanger inlet valve (CV-3055).

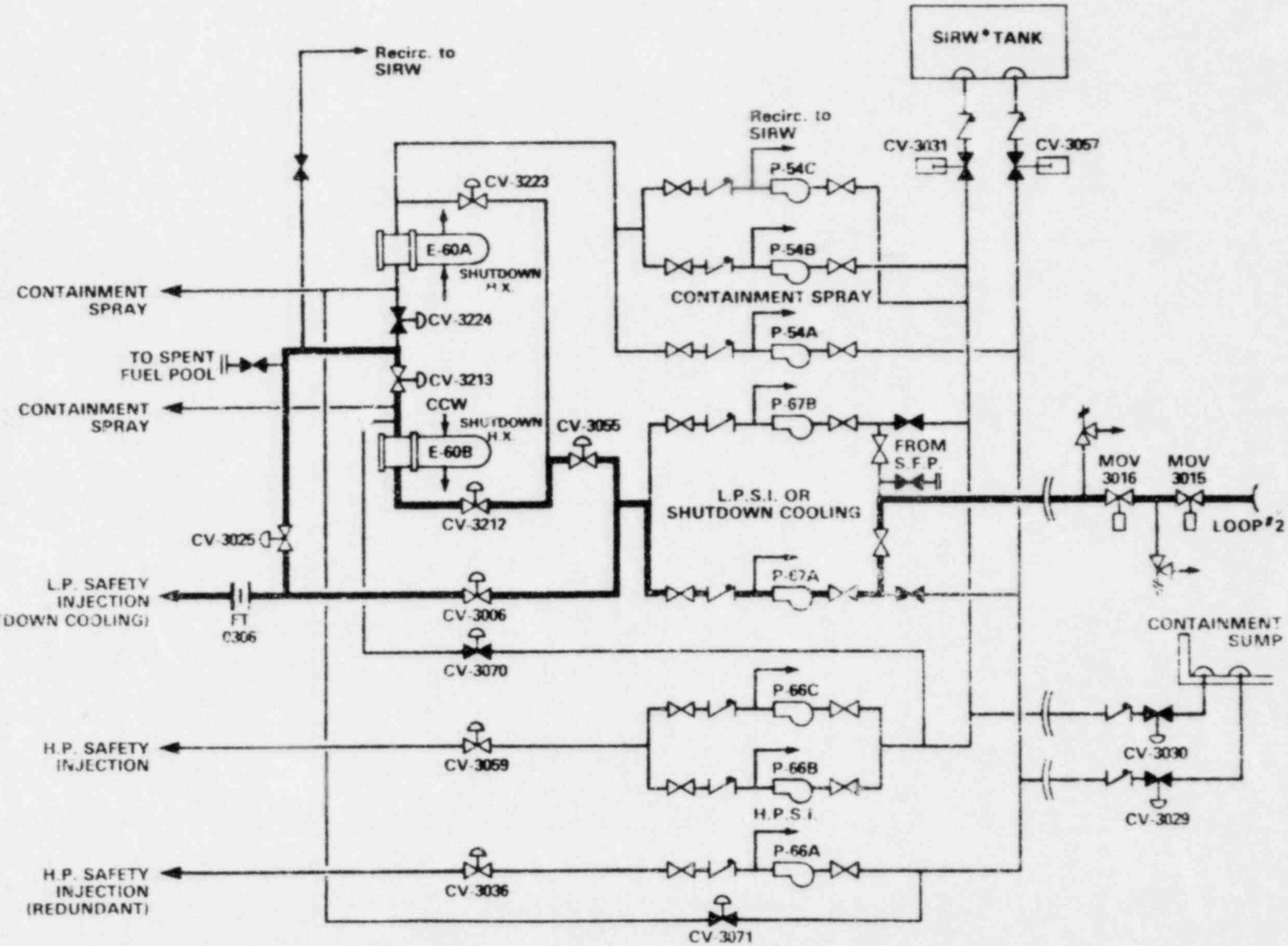
The pumps and heat exchangers at Palisades are arranged into two trains, and components of both trains were operable throughout the previously described events. However, both of these trains can be disabled by a single failure obstructing the required flow path. If such a failure could not be remedied in a timely manner, another method of shutdown cooling other than RHR would be necessary.

Prior to this event, the licensee's precedures were primarily directed to inoperability of pumps and heat exchangers rather than to flow path unavailability. After this event, the licensee has instituted the following actions to improve the reliability of the air-operated valves in the DHR system:

1. Low points in the air system are blown down once each shift.
2. The desiccant in the air dryers is checked once each shift and regenerated if necessary.
3. An operator observes valve motion during stroking tests to detect degraded valves.

The following lessons have been learned through the experience at Palisades:

1. A single failure can cause loss of shutdown cooling even if both trains of RHR are operable.
2. Moisture buildup in air lines can cause failure of one or more air-operated valves, particularly during periods of high air demand.



*SIRW SAFETY INJECTION REACTOR WATER

Fig. B.1. Palisades shutdown cooling systems.

3. Such valve failure may not be readily detectable if the control room has valve demand position as the only indicator of valve position.

B.2 RHR Pumps Cavitated Because of Low RCS Level at Beaver Valley 1

(IE Information Notice 81-09, Abstract 27, March 26, 1981)

During a shutdown on March 5, 1981, both RHR pumps were lost while the reactor water level was near the hot-leg midpoint. During the 54 min needed to vent both RHR pumps and add approximately 2.3 m^3 of water to the primary system, primary coolant temperature increased from 39 to 76°C.

This occurred when water was slowly lost from the reference leg of a temporary (tygon tubing) differential pressure system installed to measure water level in the primary system. There were no required surveillance procedures to check proper operation of the water level instrument. Consequently, the reference leg water loss was not detected until the actual primary system water level decreased to about 15 cm below the indicated level, low enough to allow air entrainment in the RHR suction line, which caused both RHR pumps to be air bound.

The significance of this event is the loss of the total RHR system function. The lack of surveillance procedures allowed the failure in the tygon tubing system to go unnoticed, resulting in the loss of the total RHR system function.

B.3 Broken Air Line on RHR Valve at McGuire 1

(LER 81-10, Abstract 56, February 7, 1981)

During routine surveillance, the control operator discovered total RHR flow return to be about 0.1 m^3/s (1500 gpm) rather than the 0.2 m^3/s required by Technical Specifications. An air line supplying the actuator on an RHR valve broke and allowed the valve to fail.

All rigid air supply lines are being replaced with flexible lines. Operating procedures for the RHR system will be modified, and more frequent surveillance of the system will be emphasized.

The cause of the break of the rigid line was not given in the LER, although excessive vibrations are often the cause (see Task 2 of this series on compressed air systems). The significance of this event lies in its potential for occurrence in other situations and at other plants. Rigid air supply lines should be considered for replacement with flexible lines.

B.4 ECCS Actuation Causes RHR Pump Cavitation at Davis-Besse 1

(IE Information Notice 80-44, Abstract 43, December 5, 1980)

In attempting to isolate electrical shorts and/or grounds, plant personnel removed ac power from channel 3 of the SFAS. When channel 3 was reenergized, an indicating lamp was out; therefore, an attempt was made to replace the failed lamp with a spare unit. While removing a lamp from a spare output slot in a channel 3 chassis, an arc was drawn between the lamp and the module chassis, apparently because of a combination of shorts or grounds in the SFAS. This arcing, coupled with a common connection between channels 1 and 3,* resulted in the loss of a power supply in channel 1. Because all the bistable trips in channel 3 had not been completely reset and because a power supply to channel 1 was lost, the two-out-of-four actuation logic actuated SFAS levels 1, 2, 3, and 5. Because SFAS level 5 indicated that the borated water storage tank (BWST) was at a low level, the ECCS was placed in a recirculation mode in which the ECCS suction was aligned to the emergency containment sump.

To place the ECCS system in the recirculation mode, the supply valves in the ECCS line and those from the containment emergency sump open, and then the supply valves leading to the RHR pumps from the BWST start to close. Thus, during this valve transition period, a flow path existed to the reactor coolant system (RCS) from the BWST via the ECCS pumping system (i.e., the RHR pumps); however, because the RCS pressure

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*The reason for the common connection between channels 1 and 3 is assumed to be the aforementioned shorts and/or grounds.

was higher than that of the pumping system (14.5 vs 11.0 MPa), no BWST water was pumped into the RCS. Rather, during the valve transition time ($\sim$1.5 min), $\sim$56,000 L of borated water was drained from the BWST to the containment emergency sump.

The major concern in such cases is that the RHR pumps could become air bound if their suction lines were aligned to a dry sump. At best, with the pumps air bound, the pump motor would trip automatically or could be tripped manually before any damage occurred, in which case flow could be established after the system is vented; at worst, the pumps could be damaged and become inoperable, in which case the active portion of the low-pressure ECCS would not be available.

The significance of this event lies primarily in the possible temporary loss of RHR flow because of pump cavitation or permanent loss of RHR flow because of pump damage. Also significant in this event is the common-mode failure mechanism, the shorts and/or grounds, that defeated the two-out-of-four logic of the SFAS.

B.5 Check Valve Failure in RCS to RHR Line at Davis-Besse 1

(IE Information Notice 80-41, Abstract 44, November 10, 1980)

The RHR system check valve CF-30 is the inboard one of two in-series check valves that is used to isolate the RCS from the low-pressure RHR system. A detailed investigation found that the valve disk and arm had separated from the valve body and were lodged just under the valve cover plate. The two 2-5/8 × 5/8-in. bolts and the locking mechanism for the bolts that hold the arm to the valve body were missing and have not been located. The CF-30 valve is a 14-in. swing check valve manufactured by Velan Valve Corporation. The cause of the failure has not been identified.

This recurring single failure event is significant. The hypothetical simultaneous failure of both these check valves would allow the high RCS pressure to be present in the RHR piping system, designed for about one-fifth that pressure.

B.6 CCW Piping to RHR Heat Exchangers Embedment Plate Bows at Beaver Valley 1

(LER 80-46, Abstract 167, July 1, 1980)

While performing an inspection in accordance with IE Bulletin 79-24, the embedment plate for CCW piping to the RHR heat exchangers was found to be slightly bowed, and the concrete around the plate was spalled in a few locations. An engineering review found the embedment plate to be structurally sound.

The cause was a failure to throttle the CCW flow to the RHR heat exchangers after intermittent isolation of service, causing water hammer and a resultant stress on the embedment plate. The operating procedure has been changed to require throttling of the CCW flow to the heat exchangers in this situation.

B.7 RHR Fails Because of False High-Pressure Signal at Beaver Valley 1

(LER 80-31, Abstract 190, May 21, 1980)

When a leaking RHR system vent valve weld while at zero power was found, a procedure was started to reenergize the RHR isolation valves to permit rapid isolation capability in case of a gross failure of the weld. A total loss of RHR occurred when the valves closed automatically because of a false high-pressure signal from a deenergized channel. Safety implications were minimal since the leakage was contained and the system was capable of total isolation. The false high-pressure signal (caused by a deenergized channel for a process control signal*) made this a significant event by causing both RHR loops to be isolated, resulting in a loss of total RHR system function.

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*Assumed to be SFAS channel input signal.

B.8 Total RHR Flow Lost Because of Air-Bound Pumps at Beaver Valley 1

(LER 80-22, Abstract 199, April 8, 1980)

While the RHR flow was being increased to the Technical Specification value required for dilution, the plant experienced a total loss of RHR flow caused by the pumps being air bound. [Resin had been replaced in a primary system demineralizer and a dilution of ~50 gpm was expected.] With RHR flow at 2500 gpm, the RHR pump (RHP-1A) ammeter began to fluctuate and then dropped to zero before the operator could reduce flow. Both RH-P-1A and -1B were found to be air bound. RHR flow was restored 33 min later after several attempts to restart the pumps resulted in no flow.

The cause was that the RHR pump was not vented when the flow began to increase. Operating procedures have been changed so that an operator is present while the flow is being changed in the RHR system. There have been losses of RHR flow in the past because the pumps were air bound, and methods are being investigated to improve the system design.

B.9 Total RHR Flow Lost Because of Air-Bound Pumps at Beaver Valley 1

(LER 80-23, Abstract 198, April 11, 1980)

The plant was in mode 5 with the steam generators drained and RCS level mid-span in the loops. The RCS temperature was 38°C and the B RHR pump was in service. A complete loss of RHR flow occurred while plant operators were increasing RHR heat exchanger flow by throttling the heat exchanger bypass flow. At 0020 h, when beginning this evolution, pump flow dropped to zero as the pump became air bound. At this time, the B RHR pump was shut down, and the A RHR pump was started. It also showed no flow and was shut down. Both pumps were then vented; the RCS level was increased. At 0108 h, the B pump was started satisfactorily, and normal conditions were reached at 0130 h.

This incident is not attributed to lack of venting because the flow was only being diverted, not increased. The cause for the air binding of the RHR pumps could not be determined from the available information; the cause is assumed to be within the system design or the event could have been caused by the standard operating procedures for that design.

This incident is significant because of the loss of total RHR system function for 48 min. A procedure has been implemented for response to a total loss of RHR flow. A continuous vent hose has been installed.

Appendix C

DESCRIPTION OF EVENTS INVOLVING HUMAN ERRORS AND THE DECAY HEAT REMOVAL SYSTEM

Appendix C contains detailed descriptions of the following events:

C.1 <u>Reactor Startup Commenced While One LPCI Loop is Inoperable at Brunswick 2</u>

[LER 81-59 (not in Appendix E), June 29, 1981]

During normal reactor startup, the valve lineup for the RHR loop A precluded its operation in the recirculation mode: the torus suction valve, F020A, was stuck closed. Three auxiliary operators were using a manual valve operator to try to break the valve disk off its seat. The control operator assumed that sufficient time had elapsed to open the F020A valve and thus initiated a reactor startup. He also initialed as complete the step on the startup procedure requiring the valve to be open.

Shortly after startup began, the Shift Operating Supervisor questioned the position of the F020A valve during a review of the control panel. The startup was secured with seven rods withdrawn, and the auxiliary operators assigned to open the valve were contacted to determine the actual valve position. Word was received in the control room that the F020A valve was still shut and could not be broken off its seat. The seven withdrawn control rods were immediately inserted and the reactor was switched to the refueling mode.

Lack of communication among station operators makes this a significant event.

C.2 RHR Flow Lost Because of Incorrect RCS Level Indication at Trojan

[LER 81-12 (not in Appendix E), June 26, 1981]

During an RCS level reduction, the operating RHR pump began cavitating and was stopped to prevent pump damage. The plant was in mode 5 with RCS temperature at 60°C and one train of RHR operating. The RCS letdown was being diverted to the hold-up tank to reduce RCS level below RCP height in preparation for working on its seal. Investigation revealed the RCS level standpipe to be indicating erroneously because of inadequate venting of the pressurizer. The RCS was properly vented, the level was restored, and the pump was restarted in about 75 min.

The event is significant because the erroneous indication of RCS level was caused by the inadequate venting of the pressurizer, a human error that is not directly associated with the operation of the RCS level standpipe and is thereby more difficult to detect as cause for the RHR pump to cavitate.

C.3 RHR Isolation from RCS During Initial Fuel Loading at McGuire 1

[LER 81-72 (not in Appendix E), May 27, 1981]

A normally energized relay in the solid state protection system (SSPS) logic was inadvertently deenergized during SSPS logic modification, closing the RCS loop 3 discharge-to-RHR containment isolation valve (1ND2A) and isolating the RCS from the RHR system.

It was not realized that work on the A train SSPS cabinets would affect B train operation of RHR. The A and B trains of RHR share a common suction from the RCS with two suction isolation valves in series (one train A powered and the other train B powered). Closing either valve isolates the RCS from the RHR system. As soon as 1ND2A closed, an operator was sent to deenergize the valve and open it manually.

At the time of the incident, only new fuel was in the core, so no decay heat load existed. Because no boron concentration changes were in progress, no mixing was required. For these reasons and because RHR flow was restored so rapidly, the health and safety of the public were not affected. The event is significant because of the loss of total RHR system function when the suction isolation valve was inadvertently closed. In this case, the operators have shown that the flow path could be restored quickly. However, if a similar incident were to occur when RHR was needed, the valves might not be opened so quickly and RHR total function would be lost.

C.4 RCS Exceeds 38°C Without Containment Integrity at Peach Bottom 2

(LER 81-31, Abstract 2, May 18, 1981)

Shutdown cooling was secured to permit maintenance of valve MO-17, a shutdown cooling suction isolation valve. The operator ordered restoration of shutdown cooling because of increasing reactor coolant temperature. Reactor coolant temperature exceeded 38°C before cooling was reestablished, which exceeded the Technical Specifications limit because primary containment integrity was lacking. The temperature exceeded 100°C for about 2.5 h.

This event was significant because of the lack of timely coordination between operations and maintenance personnel, which caused loss of RHR function.

C.5 Inadvertent Transfer of Reactor Coolant Water at Rancho Seco 1

[LER 81-24 (not in Appendix E), May 19, 1981]

While the unit was still shut down following refueling, ~ 15 m^3 of reactor coolant water was inadvertently transferred from the RCS to the reactor building emergency sump. A slight transient occurred because the RCS pressure dropped from 1.5 to 0.6 MPa. This resulted in refilling and reventing of the RCS.

At the time of the event, the A RHR system was in service and the reactor building emergency sump isolation valve to the B RHR system was to be tested. Both RHR systems have a common suction from the reactor vessel outlet, and isolation through either of two isolation valves is required when performing valve testing on one system while the other is in service.

Auxiliary operators were sent to close (or verify closed) one of the isolation valves, and subsequent communications between these auxiliary operators and the control room operator indicated that the valve was in the process of being closed and would soon be completely closed. The control room operator began the B RHR system test by stroking the emergency sump valve to the B RHR system after sending another operator to energize the electrical breaker for it, assuming the isolation valve was by then completely closed. The valve was not completely closed, however, and a flow path was established from the RCS to the reactor building emergency sump. This event is significant because ~ 15 m^3 of reactor coolant was transferred to the sump before the isolation valve completely closed and stopped the flow.

The licensee reviewed the incident and determined that the procedures used for testing valves were adequate and had appropriate limits and precautions. The cause was attributed to a breakdown in communications or a lack of complete communication. As a result, all shift supervisors were sent a memo discussing the event, reiterating the importance of complete and proper communication, and requesting that the event be reviewed by all operating personnel. The licensee considers this adequate corrective action to preclude a similar occurrence.

C.6 All Core Cooling Paths Temporarily Lost at Sequoyah 1

(LER 81-21, Abstract 51, February 11, 1981)

The unit was operating in cold shutdown (mode 5) with RHR pump A and RCS pumps 1 and 2 running with the RCS temperature at 82°C and pressure at 2.1 MPa. At 1931 h, the RHR containment spray was inadvertently initiated when an assistant unit operator (AUO) incorrectly opened valve 1-FCF-72-40, which connected the RHR system to the containment spray header. The spray started and continued for 35 min, releasing 150 m^3 of primary water and 250 m^3 of RWST water to the containment building.

AT 1981 h, the unit operator (UO) received alarms indicating a rapid decrease in pressurizer level and pressure. The UO notified the Shift Engineer of the condition and then tripped reactor coolant pumps 1 and 2 for pump protection (pumps 3 and 4 were not running). The situation was diagnosed as a possible LOCA, and emergency operating instructions 0 and 1 were consulted. The UO announced over the public address system that all employees should evacuate the containment. Health Physics and Public Safety were notified of the situation and their aid was requested.

Containment purge was stopped, and a path from the RWST to the charging pump suction was opened in an attempt to reestablish pressurizer level. RHR pump B was started at 1935 h with suction from the RWST, and then the pressurizer level started to increase rapidly. Whether the RHR pump A ran continuously throughout this event or was shut off at some time could not be determined from the available reports.

At 1948 h, the Radiological Emergency Plan, IP-4, was implemented. The evacuation alarm was sounded, and an announcement was made for all employees to assemble in designated areas. Accountability was initiated, and plant access control was established.

At 2009 h, manual isolation of the auxiliary building was initiated, and safety injection system pump A and centrifical charging pump B were started.

At 2014 h, the AUO, who opened the isolation valve, entered the control room with another UO, discussing the valve. At this time, a control room employee checked the indicator light and verified that the valve was indeed open. The valve was shut and IP-4 was terminated. The NRC duty officer was notified of the events at 2030 h. RCS pumps 1 and 2 returned to operable status at approximately the same time.

The primary cause was a lack of adequate oral communication. A secondary cause was the lack of sufficient training of the AUO on the particular work station to which he was assigned.

The event is significant because RCS coolant via the containment spray header was lost because of reactor coolant flowing out the active RHR injection line.

C.7 Shutdown Cooling Flow Lost at Calvert Cliffs 2

(LER 810-04, Abstract 9, February 4, 1981)

While conducting preventive maintenance, shutdown cooling flow was lost because of inadvertent deenergization of No. 21, 120-V vital ac bus. Deenergizing this bus caused a shutdown cooling return-header valve to shut. The bus was soon reenergized and flow restored.

The preventive maintenance procedure in use by plant electricians did not contain sufficient information. Procedures for vital ac inverters and back-up bus components are being revised to include specifying the required power source lineup necessary for conducting the maintenance.

The loss of shutdown cooling system function makes this a significant event.

C.8 RHR Incorrectly Isolated at Farley 1

(LER 80-57, Abstract 118, September 22, 1980)

While taking the reactor from cold shutdown to hot standby, the RHR system was removed from service and isolated from the RCS prematurely because of operator misinterpretation. The RCS temperature was 121°C, and Technical Specifications require that the RHR be operable at that temperature. Plant precautions and limitations stated that RHR must be removed from service prior to the pressurizer temperature reaching 246°C but did not caution against removing RHR if the RCS temperature is

≤154°C. A procedure change was generated to preclude future occurrences. The RHR was returned to operable status.

Incorrect procedures caused temporary loss of the total RHR system function.

C.9 RHR Flow Lost Three Times at Davis-Besse 1

(LER 80-58, Abstract 171, July 24, 1980)

At 0955 h on July 24, the control room operators observed a loss of RHR flow caused by valve DH12 closing. Decay heat pump 1-2 was stopped. Bypass valves DH21 and DH23 were opened and the pump restarted. The same day, at 2232 h, personnel were attempting to restore the SFAS CH4 cabinet to normal when valve DH11 was inadventently reopened with no suction available. The third event occurred on August 3, 1980, when Instrumentation and Control personnel removed bistable BA413 and caused valve DH11 to close, stopping flow. The bistable was reinstalled, and flow was restored.

The first loss of RHR flow was caused by construction electricians who pulled wires into a cabinet and shorted a fuse clip in the control circuit for DH12. The second event was caused by procedural deficiency in that the maintenance work order being used did not contain adequate restoration instructions. The third event was caused by an error by the maintenance specialist. These events are significant because Davis-Besse has been experiencing numerous losses of RHR flow and RHR function could be lost for the same reasons at times when flow would be desperately needed.

C.10 Incorrect RHR SW Pump Breaker Opened at Brunswick 2

(LER 79-73, Abstract 300, August 13, 1979)

The licensed operator was supposed to place loop B of RHR service water system under clearance for maintenance; it was inoperable at the time. However, a tag and tag-out sheet had been incorrectly prepared, and the operator opened the A loop service water pump breaker instead, making both loops inoperable for about 15 min. At Brunswick, this RHR service water system cools the RHR heat exchangers directly.

The clearance procedure is being revised to require that the tag-out sheets and tags for any clearance on a system covered by Technical Specifications be reviewed and approved by a licensed senior reactor operator prior to issue. This review must be independent of the person filling out the clearance.

The event is significant because of the loss of total RHR system cooling function. If a shutdown had been required and operators could not discover why both RHR service water loops were inoperable, shutdown cooldown cooling via the RHR exchangers would not have been available.

C.11 Reactor Low-Level Limit Too Low for RHR Suction at Salem 1

(LER 79-59, Abstract 207, June 30, 1979)

The level in the reactor vessel was lowered 3 cm above the low-level limit to support plant maintenance during reactor shutdown. The RHR pump started to lose suction and was secured. The level was raised 15 cm, and RHR flow was restored.

The low-level limit was raised 13 cm above the previous limit, ensuring sufficient suction to the RHR pump.

This event is significant because of the loss of total RHR system function. This is also significant because the Technical Specifications listed this low water level as being sufficient for safe operation.

Loss of RHR system function also occurred at Salem 1 on April 24 and May 8, 1979, because of relay testing; in these two cases, RHR was restored within 4 min.

C.12 Incorrect RHR Service Water Pump Removed from Service at Brunswick 2

(LER 79-50, Abstract 329, June 19, 1979)

Mechanics were supposed to uncouple and check alignment of 2B RHR service water pump. By mistake, they uncoupled the 2A RHR service water pump, thereby leaving both loops of RHR service water inoperable for

7 h. Plant management issued a memorandum requiring that future clearances be issued only to personnel who have had training in clearance procedures.

At Brunswick, the RHR service water cools the RHR heat exchangers directly, so during these 7 h the RHR heat exchangers were not functional. By the time the pressure was reduced enough for RHR cooling to begin, however, the 2A RHR pump could have presumably been recoupled and operated.

The event is significant because of the loss of total RHR system function.

Appendix D

DESCRIPTION OF EVENTS WITH CAUSES OUTSIDE THE DECAY HEAT REMOVAL SYSTEM

Appendix D contains detailed descriptions of the following events:

D.1 Development of Steam Bubble Under Vessel Head During Cooldown at McGuire 1

[PNO-II-81-39 (not in Appendix E), June 2, 1981*]

While reducing RCS temperature and pressure to achieve a cold shutdown condition, a steam bubble formed in the reactor vessel head area when the system was vented. At an RCS loop temperature of 71°C and a

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*See Appendix D.6 for a similar event at St. Lucie, LER 80-29.

pressure of 0.4 MPa, the RCS vent was opened and pressurizer level increased about 4%. The vent was closed and level returned to pre-vent conditions. A check of the reactor vessel head temperature showed the head temperature to be 121°C.

A similar event occurred again 8 h later at RCS loop temperature of 47°C. The highest recorded upper-head temperature then was 99°C. The system was repressurized to permit operation of the reactor coolant pumps to sweep the loops; a cold shutdown was then achieved.

Because the reactor had not achieved initial criticality, there was no decay heat and no natural circulation as there would have been in an operating plant. Also, the operating RHR resulted in recirculation of two of the four RCS loops.

D.2 RHR Heat Exchanger Failures at Brunswick 1 and 2*

(Brunswick 1: LER 81-32, Abstract 20, April 19, 1981)
(Brunswick 1: LER 81-00S, Abstract 21, April 25, 1981)
(Brunswick 2: LER 81-49, Abstract 10, May 6, 1981)
(Brunswick 2: LER 80-30, Abstract 126, April 12, 1980)

During a special inspection at Brunswick 1 on April 19, 1981, a baffle plate in the 1B RHR heat exchanger was found to be displaced 23 cm at the bottom, creating a service water flow path from the inlet to the outlet, bypassing the tubes. During the repair of the 1B RHR heat exchanger baffle plate, a loss of shutdown cooling occurred because of failure of the 1A RHR heat exchanger. This loss of cooling occurred immediately following the starting of an RHR service water pump providing water to the 1A RHR heat exchanger. An alternate shutdown cooling path was established using the RHR system, the fuel pool cooling system, and the core spray system. The baffle plate on the 1A heat exchanger was also found to be displaced at the bottom. The apparent cause of damage to the heat exchanger baffles was loading in excess of their design capability. Water hammer events were suspected, but no evidence was

*This discussion also appears in Appendix D of Task 3 on service water systems.

found. Later, buildup of oyster shells in the heat exchanger was discovered to be the cause.

Brunswick 2 (LER 81-49, May 6, 1981) reported that oyster shells were blocking and obstructing the heat exchanger tubes, producing excessive differential pressures across the divider plate (also called rib plate and baffle plate) during RHR pump operation. These differential pressures produced stresses greater than the divider plate could withstand, causing it to bow and be displaced. The divider plate was buckled in the center at the bottom and was displaced upward 8 cm. The welds along the top and sides of the plate remained intact. [This plate was replaced in April 1980 (Reference Brunswick 2, LER 80-30).] Shells of various sizes formed a layer averaging 5 cm thick with areas as thick as 13 cm on the side of the 2B RHR heat exchanger. Additional shell blockage was found in one-half of the tubes. The 2A RHR heat exchanger was similarly obstructed, even though the divider plate was not bowed or displaced and fewer shells were present because it is used less frequently than the 2B heat exchanger. The presence of shells in the heat exchangers resulted from a buildup of shells on the walls of the main service water piping. As the oysters died, their shells fell off and collected in the heat exchangers. The oyster buildup occurred when the chlorination system was out of service for an extended period because of operating difficulties.

When the chlorination system is inoperable for extended periods, differential pressure checks should be made periodically to ensure that design flow rate is available. Installed differential pressure gauges could be useful in identifying excessive differential pressure across heat exchangers or other system components [e.g., filters (see LER 80-103 for Hatch 1)] <u>before</u> damage or other problems occur.

This is a common-cause failure event that could eventually affect all heat exchangers and coolers on the service water side. If the heat exchangers were used on a rotational basis so each unit had the same amount of service, a gradual buildup in all heat exchangers could cause multiple failures all occurring about the same time, thus influencing plant safety.

D.3 LPCI Inoperable Because of Several Shorted Conductors at Quad-Cities 1

(LER 81-07, Abstract 32, March 13, 1981)

An operator observed that there was no position indication on 1A recirculation pump discharge valve MO-1-202-5A. The breaker for the valve motor was found tripped, and it could not be reset. This caused the LPCI mode of RHR to be inoperable. Cable number 12507 leading to this valve inside the dry well was tested electrically and found to have several shorted conductors. Reactor power was reduced to ~200 MW(e) for a dry well entry so that a temporary cable could be installed. The valve and its associated interlocks were tested satisfactorily.

These shorts caused the entire LPCI system to be inoperable, and therefore this event is significant.

D.4 RHR Flow Lost Because of Engineered Safety Features Actuation at Davis-Besse 1

(LER 80-49, Abstract 163, June 14, 1980)

During restoration of containment pressure inputs to SFAS cabinets, the station experienced a safety features actuation. The actuation caused decay heat suction to switch from the BWST to the emergency sump, which, being empty, caused the RHR pump to lose suction. The RHR pump was manually stopped. This loss of decay heat flow was a violation of Technical Specifications.

The cause of this event was a deficient procedure that neither required the mechanic to go to test-trip bypass while restoring an SFAS channel nor required him to reset the channel after completion of restoration. A modification to the procedure was added to provide these instructions.

D.5 Pavement Deflection Near Intake Structure at Hatch 1 and 2*

(LER 80-62, Abstract 49, June 12, 1980)

The weight of a crane caused some pavement to collapse over the plant and RHR service water piping for Units 1 and 2. This event caused no serious problems, but it has the potential for loss of all plant and RHR service water supply. Consequent investigation revealed two problems: (1) the separation of fill from underneath the piping (cause unknown) and (2) the existence of a temporary pipe support that should have been removed prior to backfilling.

If the plant were running at 100% power and a pavement collapse broke all the plant and RHR service water intake piping, there would be no cooling water for RHR shutdown heat exchangers and all the other service water cooling functions in the plant. Immediate shutdown of the reactor would be necessary. Reactor heat would be dissipated first through the turbine bypass, which at Hatch has 25% of full-load capacity. Other BWRs generally vary between 5% and 40% bypass capacity. The circulating water system is usually separate from these service water systems, and it would provide an ultimate decay heat sink via the cooling towers.

Also, steam may have to be vented to the torus. Eventually, the suppression pool would reach a temperature such that cooling would be required or no more heat could be added. At that time, relief valves would have to be opened venting steam to containment. If all these options were insufficient, this event would be the precursor of a core meltdown.

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*This discussion also appears in Appendix D of Task 3 on service water systems.

D.6 Steam Bubble in Reactor Vessel During Natural Circulation Cooldown at St. Lucie 1*

[LER 80-29 (not in Appendix E), June 11, 1980]

During full-power operation, the flow of CCW to the RCP seals was lost. This was initiated when moisture from a minor steam leak shorted the terminal board of a solenoid-operated air valve, which in turn caused the containment isolation valve for CCW to close. The isolation valve was on the CCW return line from the RCPs, and because the line was common to all pumps, all CCW flow to the pump seals was stopped. The operators tripped the reactor, and after unsuccessfully trying to restore CCW flow for 2 min, tripped all four RCPs. The RCPs were running ~8 to 9 min without CCW prior to being tripped. The plant operators, concerned over the increasing the hot-leg temperature T_H, jogged RCP 1B1 for 2 min after it was tripped.

About 30 min later, cooldown was started with natural convection by dumping steam via the atmospheric dump valves. The CCW flow was reestablished 1 h later, by bypassing the solenoid-operated air valve with a temporary air line. The rate of RCP seal leakage varied, but the seals did not fail. The RCPs were not restarted because RCP lower seal cavity temperature had exceeded the 121°C limit specified by the pump manufacturer (Byron-Jackson).

After about 3.5 h of normal natural-circulation cooldown, the RCS pressure was reduced from 7.86 to 4.76 MPa by directing the flow of charging water through the pressurizer auxiliary spray line. The pressurizer water level increased rapidly and then varied widely for about 5 h during cooldown and depressurization. The pressurizer water level increased at a rate approximately ten times as rapidly as could be accounted for by the charging flow rate when the charging pumps were in the spray mode and decreased rapidly when in the normal charging mode. This behavior is indicative of a steam bubble in the RCS. (Samples of

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*See also IE Circular 80-15; *Power Reactor Events* 2(4), July 1980; *Nuclear Safety*, 21(6): 782 (November-December 1980); NRC/AEOD report, *Saint Lucie 1 Natural Circulation Cooldown on June 11, 1980*, by E. V. Imbro; and similar event at McGuire 1, Appendix D.1.

reactor coolant indicated that there were not enough dissolved gases in the coolant to account for the magnitude of the level oscillations observed.)

The steam bubble formed in the reactor vessel heat due to a temperature lag (Fig. D.1). The bulk coolant temperature in the head region remained higher than the rest of the RCS because there was essentially no flow through this region during natural circulation. At the time the steam void developed, the subcooling margin (calculated using either T_H or the core exit temperature) ranged between 83 and 122°C. The minimum required subcooling region 28°C was not approached until about 9 h after natural-convection cooling started.

Forced circulation, using LPSI pump 1B, began about 8 h after cooldown by natural circulation. About 1.5 h later, LPSI pump 1A was started, taking suction from the RWST and discharging into the LPSI header (common to both LPSI pumps). The isolation valves on the common lines to the RWST were open as required, recirculation flow for warming the LPSI system had existed since the event began, and pump 1A was operating with the minimum flow recirculation line to the RWST open. (The LPSI pump 1B minimum flow recirculation line should have been closed but was later discovered to be one-half turn open.) While pump 1A was being used to inject water and maintain the system pressure near the shutoff head pressure of the pump, the operators tried unsuccessfully to raise the RCS pressure above 1.4 MPa by using charging pumps and pressurizer heaters and by securing letdown.

The cold-calibrated pressurizer-water-level instrument indicated a rise to 64% and stayed constant while the hot-calibrated channels reached 100%. Although the constant level on the cold-calibrated channel indicated that the pressurizer was solid, the continued charging flow did not cause the pressure to rise above 1.4 MPa, as it should have if the RCS were solid. The absence of a pressure rise indicated that there was a leakage path from the RCS.

During the 90 min that the LPSI pump 1A was operating in the injection mode, the RWST gained about 17 m^3. A path from the RCS to the RWST existed through the LPSI pump 1B minimum flow recirculation line

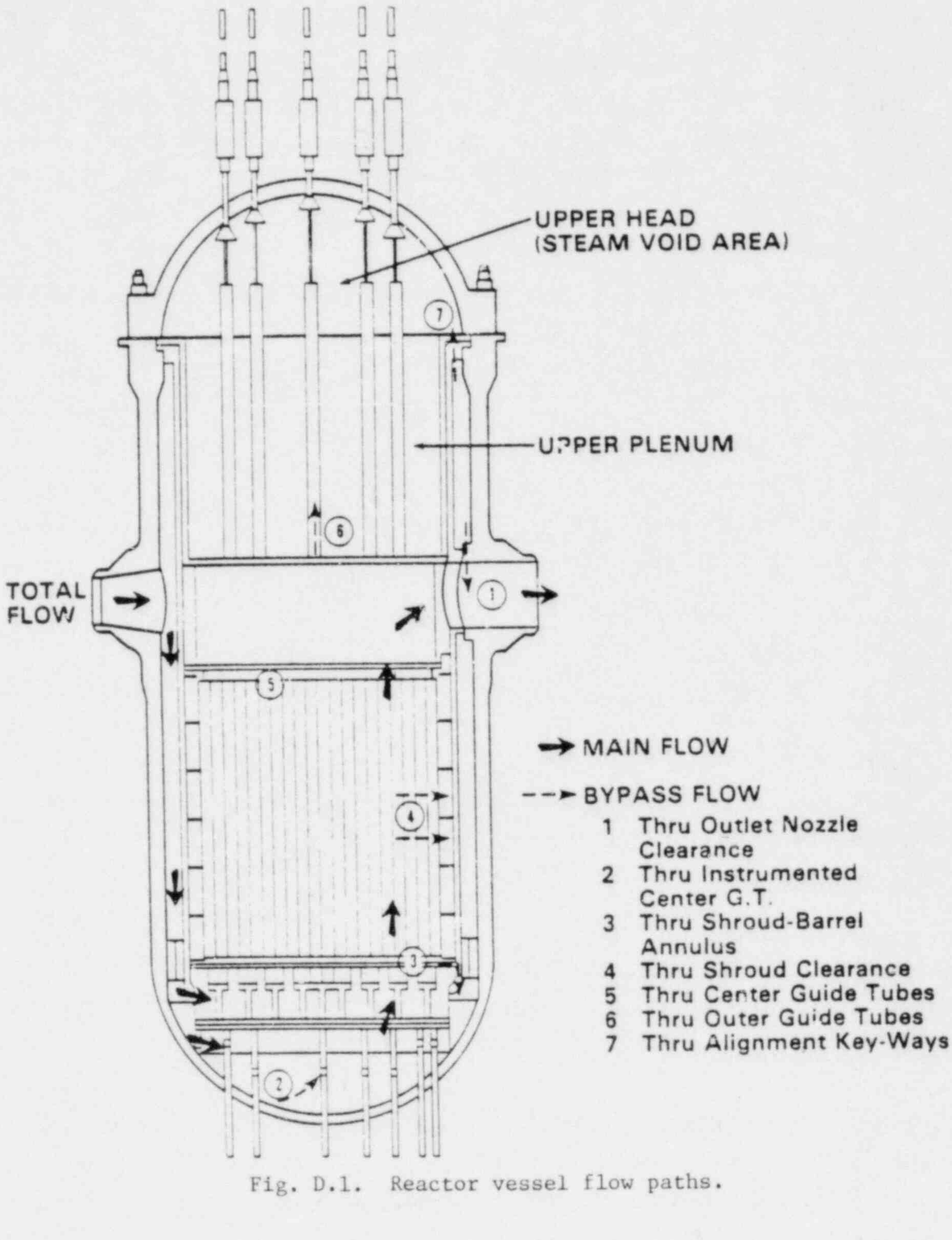

Fig. D.1. Reactor vessel flow paths.

(Fig. D.2). After the shutdown cooling system had been warmed, the pump 1B minimum flow manual isolation valve was closed; however, on a check subsequent to the discovery of an increasing RWST level, the valve was found to be one-half turn open, allowing the flow of water back to the RWST. Also, about 90 min after starting, pump 1A was stopped and its recirculation line was isolated. The LPCI pump 1A discharge check valve was a second suspected leakage path through the 1A minimum flow line to the RWST.

The continued use of both charging pumps caused slight rises in both the pressurizer pressure (to 1.8 MPa) and the pressurizer water level. Indications of a steam void in the reactor vessel head were no longer evident as the pressurizer became water solid and the RCS pressure increased, although the exact time it disappeared is not clear. Letdown from the RCS in excess of charging flow rate was then reestablished, and a steam bubble was drawn in the pressurizer.

The RCS was degassified over the next day, then depressurized and drained for inspection and replacement of all RCP seals. The seals were removed and visually inspected, but they showed no signs of damage.

Actions recommended in the NRC's IE Circular include: (1) informing all facility-licensed personnel of the possibility of a steam-void formation in the reactor vessel head during natural-convection cooling, even when a high subcooling margin exists in the reactor coolant loops; (2) reviewing and revising procedures for using natural convection for shutdown cooling and cautioning operators against the conditions that occurred, including appropriate recovery action should they occur again; (3) establishing natural-convection cooldown and depressurization rates that will preclude steam-void formation and ensure adequate core cooling; (4) evaluating the design of CCW systems to determine their vulnerability to single failures that could cause loss of RCP cooling, simultaneous common-mode failure of all RCP seals, and reactor-coolant-system leaks through failed seals at multiple locations; and (5) using a temperature-monitoring system for the metal of the reactor vessel head to aid the operator in preventing the formation of a steam void in the reactor head during natural-convection cooling.

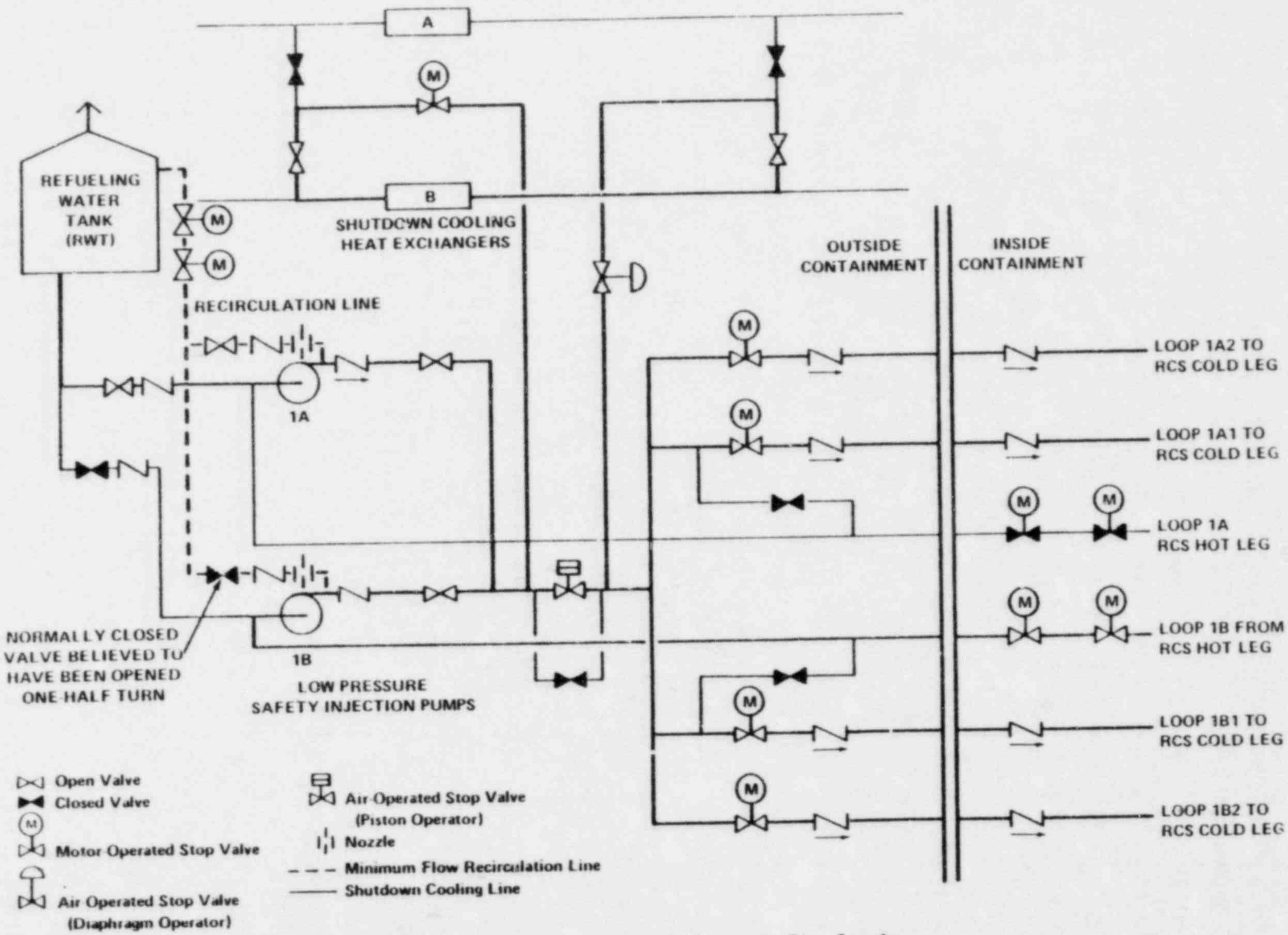

Fig. D.2. Shutdown cooling at St. Lucie.

The following numbered paragraphs are the findings and recommendations regarding this event taken from the NRC/AEOD report, *Saint Lucie 1 Natural Circulation Cooldown on June 11, 1980*, by E. V. Imbro.

Findings

1. The rapid depressurization of the RCS resulted in a plant condition unanticipated by the plant operators. Although the actual safety significance of drawing a steam bubble in the reactor vessel head during the natural-circulation cooldown appears to be small, the plant response did initially puzzle the plant operators. This could have resulted in the plant operators taking incorrect actions. Although this was not the case in this instance, operator guidance needs to be developed in this area.

2. The jogging of the RCP to aid in the establishment of natural circulation appears to have been unnecessary. The plant operators, apparently concerned over the increasing T_H, decided to jog RCP 1B1, and the first pump tripped. The RCP 1B1 pump had been run 8 min following the loss of CCW prior to its being tripped, 2 min less than the 10 min allowable time specified in plant procedures. Prior to jogging RCP 1B1, core differential temperature (ΔT) was approaching the normal full-power ΔT. Emergency operating procedures at the plant indicate that one of the criteria for ensuring that natural circulation has been established is that core ΔT is less than the normal full-power ΔT. The Combustion Engineering (CE) plants exhibit a characteristic increase in T_H during the establishment of natural circulation. Considering that T_H increased again at the same rate and stabilized at approximately the same temperature after RCP 1B1 was stopped would tend to indicate that jogging was unnecessary in establishing natural circulation. During the incipient stages of establishing natural circulation, operators need to be made aware that T_H will initially decrease then rise and peak suddenly. While jogging the pump caused no problem, it did increase the potential for seal failure. Operator guidance in recognizing natural circulation needs to be expanded.

3. The continued sloshing of the pressurizer eventually led to a condition that resulted in the simultaneous use of the LPSI pumps in the shutdown cooling (SDC) and injection modes to maintain an adequate subcooling margin. When aligned in this manner, the check valve on the discharge side of the LPSI pump in the injection mode becomes the only barrier between the reactor coolant fluid and the RWST. Since the RWST vents to the atmosphere, a leaky check valve in this system alignment creates an unmonitored leakage path for primary coolant activity. The leak tightness of the check valves on the discharge side of the LPSI pumps needs to be periodically verified.

4. The formation of the steam bubble in the reactor vessel did not inhibit natural circulation flow in the loops, although the licensee estimated that the size of the bubble was about 21.2 m^3. Information provided by the licensee indicates that the bubble extended about 25 cm below the reactor vessel closure flange. This left a 91-cm margin above the top of the hot leg, which corresponds to $\sim$8.5 m^3 of reactor vessel volume. The steam bubble size, therefore, would have had to be 29.7 m^3 before it would have reached the top of the hot legs. Intuitively, it would appear that if the RCS pressure is slowly decreased (causing a correspondingly slow expansion of the bubble), it is not likely that a bubble of this size (29.7 m^3) would be achieved, because as the size of the bubble is increased, the vapor liquid interface moves out of the upper-head region to a progressively cooler region of the reactor vessel. This would tend to condense the steam. Also, the liquid temperature approaches the measured T_H as the surface moves from a stagnant flow area to one that is in the natural-circulation flow path. This cooling effect also tends to inhibit further formation of steam. On the other hand, a very rapid decrease in RCS pressure will result in a rapid rise of pressurizer level and may result in the expansion of the bubble in the reactor vessel head into the hot legs. In this case, the dynamics of the situation may not permit sufficient time for condensation of the steam bubble in the reactor vessel. However, this may not be a problem because the vapor should condense either in hot legs or in the steam generator tubes. In any case, it may be desirable to maintain the

pressurizer level between specified bounds during the level oscillations to ensure that the vapor remains in the reactor vessel.

5. A rapid depressurization could be a problem for Babcock and Wilcox plants, particularly if they are cooling down one steam generator by natural circulation, because a steam bubble might form in the "candy-cane" region of the inactive hot leg. Once a bubble forms in this inactive hot leg, either due to flashing in the candy-cane region or due to vapor expanding out of the reactor vessel, nautral circulation could be precluded in the inactive loop. A steam bubble once formed simply by repressurization of the RCS may be difficult to totally condense: if the liquid surface is quiescent, the liquid acts as a piston and the increase in the pressure causes the bubble temperature to increase. The increase in RCS pressure causes a corresponding increase in the saturation temperature of the bubble, and if the process of repressurization is adiabatic, it is thermodynamically impossible to condense the vapor. The steam bubble can only be condensed by cooling of the bubble, which may be a relatively slow process because of the hot walls of the RCS piping.

6. During a natural-circulation cooldown with an idle steam generator, it may be possible to form steam bubbles in the upper regions of the U tubes. Although these bubbles can be condensed by cooling down the secondary side of the steam generator, the concern is that a rapid collapse of these bubbles could cause a large decrease in pressurizer level and possibly drain the pressurizer. Operators should be alerted to this possibility when starting an idle steam generator during a natural-circulation cooldown.

7. The RCS pressure boundary is an extremely reliable passive safety feature because, with the exception of the RCP seals, it requires no dependence on auxiliary systems to perform its function of containing the reactor coolant. The integrity of the RCP seals depends on cooling water. This cooling water is provided either by seal injection or CCW to the thermal barrier heat exchanger or integral seal cooler. Saint Lucie, as well as all other operating CE plants, does not have seal injection as a backup to CCW for seal cooling. In addition, the CCW is

supplied to the RCP so that a single failure can stop cooling flow to all RCP seals. Because the loss of CCW to the RCP seals may cause degradation of the RCS pressure boundary even if the RCPs are stopped, the CCW supply to the RCPs should be highly reliable even though it may not be from a safety grade source. Consideration should be given to upgrading the reliability of the system supplying cooling water to the RCP seals.

8. Saint Lucie was rapidly depressurized because of a valid concern over the capability of the RCP seals to maintain their integrity. Following an extended loss of CCW to the seals, the lower seal cavity temperature limit will be exceeded in about 25 min with the reactor at hot standby. At this point, the pump manufacturer recommends seal removal and inspection. This guidance, however, is not reflected in plant operating procedures. The manufacturer also recommends, as indicated in plant procedures, that the pumps not be run more than 10 min without CCW. In a similar loss of CCW in 1977, the RCPs were run for 12 min, and this resulted in failure of the lower stage of a seal in one RCP. The June 11, 1980, incident showed no damage to the two RCP seals inspected. These RCP seals were run for 8 to 9 min without CCW. Based on a single data point, the 10-min criterion appears to be a good one; but considering the seal stage failure in 1977 after 12 min, which clearly could have been influenced by other variables, the question of whether 10 min is close to a threshold is also raised. There appears to be a lack of data on the "off-design" performance of RCP seals.

9. The consequences of forming a steam bubble in the reactor vessel head on plants equipped with upper-head injection (UHI) were studied to determine possible adverse effects of UHI actuation to condense the steam bubble.

UHI is a passive injection system provided for core cooling during a LOCA. The injection pressure of the UHI was selected so that during a large-break LOCA the vessel head would be filled with water during injection. The UHI flows are high in this case due to the rapid depressurization of the RCS. During a small-break LOCA, UHI would be into a two-phase region but the flows would be low due to the slowly decreasing

RCS pressure. In either of these cases the potential for water hammer is minimized.

The UHI system is initially isolated during the cooldown and depressurization process as required by normal operating procedures. However, during natural circulation, if a steam bubble were formed in the vessel head, actuation of UHI would be one method available to the operators for collapsing the bubble. In this mode of operation, the UHI could be rapidly injecting into steam-filled piping which might produce water hammer be rapid steam condensation. This same situation could possibly occur over some spectrum of intermediate-size breaks.

In some plants equipped with UHI, the nozzles were added to the vessel head after the final heat treatment. Since the failure of a UHI line on the RCS side of the check valves results in a LOCA, it is important that the UHI piping will withstand any water hammer loads imposed by injection into the steam-filled lines. Consequently, it should be verified that the UHI piping will withstand potential water hammer loads associated with the use of UHI when injecting into steam-filled lines either to collapse a steam bubble in the reactor vessel head or during a LOCA.

10. One design feature provided at Saint Lucie is a valve closure on high ΔT for the CCW return valves provided for each RCP. The function of this automatic closure is to prevent reactor coolant from going into the CCW system in the event of a tube rupture in the RCP seal cooler. Therefore, if the CCW outlet temperature from the RCP seal cooler exceeds the inlet temperature by more than 95°C, the air-operated outlet valve will close.

The April 1977 natural circulation cooldown was caused by failure of a containment air compressor that caused these return valves to fail closed. After the air supply was restored, the operators were unable to open these valves because the high ΔT closure logic prevented the valves from being quickly reopened so that CCW could be reestablished to the seals. Following this incident, Saint Lucie has provided a reset feature on these valves incorporating a 10-s time delay. This permits the valves to be reopened after closure on high ΔT, but the valves will automatically close if the differential temperature does not go below 95°C within 10 s.

Incorporation of a reset feature on other plants that have an automatic isolation of CCW on failure of the RCP seal cooler would permit a more rapid restoration of CCW.

11. As the event progressed at Saint Lucie, the control room became increasingly occupied as plant personnel responded to the off-normal plant condition. Although the severity of the event did not warrant activation of the Technical Support Center (TSC), apparently the TSC could have been used to assist the operators in the evaluation of the event. This would have gotten some plant personnel out of the control room and permitted the operators to function in a quieter environment.

The TSC is officially activated in accordance with the Emergency Response Plan only during more serious events. Putting the Emergency Response Plan into effect at a level where the TSC is activated generally implies notification of the State, activation of the Offsite Support Center, and other functions geared toward an event of greater severity. Clearly, many events or situations exist where it would be beneficial to activate the TSC function at the lowest emergency action levels.

Criteria should be established to allow the use of the TSC function during events of less severity that progress over a relatively longer period of time.

12. The RCP manufacturer, Bryon-Jackson, recommends that the RCP seals be removed and inspected if the lower seal cavity temperature exceeds 121°C. At the Saint Lucie plant, the lower seal cavity temperature for each RCP is indicated on the main control board and activates an alarm when it reaches 77°C. During normal operation, this temperature is about 38 to 43°C. Although the lower seal cavity temperature is logged twice a shift, a situation could arise (i.e., failure of the high-temperature alarm) where this temperature could exceed the recommended 121°C limit for a period of time and then return to a temperature below 121°C without being noticed by the plant operators. This situation

could be avoided if, in addition to the high-temperature alarm at 77°C, a high-high-temperature alarm was also installed to actuate at 121°C. In reviewing the Saint Lucie procedure for RCP off-normal operation, no mention was made of the manufacturer's recommendation for RCP seal inspection if the lower seal cavity temperature exceeds 121°C.

Recommendations

1. During the incipient stages of establishing natural circulation, operators should be made aware that T_H will initially decrease and then rise fairly rapidly and peak suddenly. This guidance would preclude unnecessary concern or starting of RCPs.

2. Consideration should be given to an alarm for the lower seal cavity temperature if it exceeds the recommended limit of 121°C.

3. Cooldown procedures during natural circulation should be expanded to specify a nonmandatory rate of depressurization, which, if adhered to, would avoid formation of a bubble in the reactor vessel head.

4. Procedures should be developed to guide the operators in responding to a bubble formed in the reactor vessel head. These procedures should include some definite limits on the controlled oscillations of pressurizer level, if this procedure is recommended to aid in cooling the head. Emphasis should also be placed on the fact that it may not be possible to condense a steam bubble by repressurization without cooling.

5. Operator training should be expanded to allow operators to quickly recognize the symptoms of void formation of the RCS as well as the pressurizer.

6. Plant procedures should be formulated addressing the simultaneous use of the LPSI pumps in both the injection and shutdown cooling mode. Particular attention should be directed to any potential leakage paths from the RCS to the RSWT.

7. Leak tightness of the check valves in the LPSI pump discharge lines needs to be periodically verified. These valves should be included in the In-Service Test Program.

8. Analytical models used in accident and transient analysis should be examined to verify that they properly account for the observed thermal-hydraulic decoupling of the reactor vessel head region from the remainder of the reactor vessel.

9. Consideration should be given to the potential for the formation or accumulation of vapor in the candy-cane region of the Babcock and Wilcox reactors, particularly in the inactive loop when natural-circulation cooldown is being accomplished with a single steam generator.

10. Plants not using seal injection and having a high ΔT closure feature on the CCW discharge valves from the RCP should consider installing a time-delay reset that would permit temporary override of the closure feature.

11. Loss of instrument air procedures should be reviewed and revised as necessary to address the potential effects of extended loss of instrument air on RCP operation.

12. Consideration should be given to providing a supply of cooling water to the RCPs that will not be totally disabled by a single failure.

13. Consideration should be given to providing a means to measure temperature in the reactor vessel head.

14. The following definitive data based on operating experience or testing should be obtained from RCP vendors for pumps not provided with seal injection: (a) the time interval pump seals can survive without CCW at normal operating RCS temperature and pressure, if the pump is idle, and (b) the time interval in which RCPs should be stopped following the loss of CCW to preclude seal failure.

15. The potential for water hammer due to steam condensation in UHI lines should be evaluated.

16. Graduated criteria should be developed to allow activation of the TSC at the lowest levels of emergency response for incidents of less severity that progress over a significant period of time.

17. Plant procedures should be revised as necessary to include RCP manufacturers recommendation on seal inspection if lower seal cavity temperature exceeds 121°C.

Conclusions

The primary significance of the June 11, 1980, natural-circulation cooldown is that the formation of the steam bubble in the reactor head was unexpected by the plant operators and was not immediately recognized by them. This could have led to the operators taking improper corrective action, although this was not the case. However, operator training needs to be expanded so that the formation of a steam bubble in the reactor head can be promptly recognized by the operators during a rapid depressurization while the reactor is undergoing natural-circulation cooldown. Procedures should be developed to guide the operator in plant depressurization to avoid bubble formation when he recognizes that the reactor vessel head is not in good thermal-hydraulic communication with the remainder of the reactor vessel and that the formation of a steam bubble in the head is possible. Under certain conditions, rapid depressurization is necessary or desirable; therefore, procedures should also be developed to guide the operator in cooling down the plant by natural circulation with a steam bubble in the reactor vessel head.

Although further investigation, as enumerated in the recommendations, is necessary, the voiding of the reactor vessel head does not represent an immediate safety concern. Clearly, it is a plant condition that should be avoided, if possible. However, formation of a steam void in the reactor vessel head did not in any way impede natural circulation in the loops. Except for the problem caused by the leakage of reactor coolant to the RWST, the reactor was brought to a cold shutdown condition in an orderly manner, considering the new situation that confronted the plant operators.

D.7 DHR Flow Lost Because of Engineered Safety Features Actuation at Davis-Besse 1

(LER 80-29, Abstract 169, April 19, 1980)*

For 2.5 h during refueling with the RCS temperature at 32°C, Davis-Besse 1 lost capability to remove decay heat. Decay heat was being

*See also IE Bulletin 80-12 and IE Information Notice 80-20.

removed by decay-heat loop 2; the vessel head had been detensioned, but the bolts were still in place. The reactor coolant level was slightly below the vessel-head flanges, and the manway covers on top of the once-through steam generators had been removed.

Because the plant was being refueled, many systems or components were out of service for maintenance or testing. In addition, other systems and components had been deactivated to preclude their inadvertent actuation. Systems and components that were not in service or were deactivated included: containment spray system, high-pressure injection system, source range channel 2, decay-heat loop 1, station batteries 1P and 1N, emergency diesel generator 1, diesel generator 1, 4.16-kV essential switchgear bus C1, and 13.8-kV switchgear bus A, which was energized but not aligned.

The event occurred when a nonsafeguards feeder breaker for bus B in the 13.8-kV switchgear tripped. Because of the extensive maintenance and testing going on at the time, channels 1 and 3 of the reactor protection system (RPS) and the SFAS were being energized from only one source that, because of the ongoing maintenance, supplied power through the breaker that tripped. Because the SFAS logic used at Davis-Besse is a two-out-of-four scheme in which the loss (or actuation) of any two signals results in the actuation of all four channels (i.e., channels 1 and 3 and channels 2 and 4), the loss of power to the bistables of channels 1 and 3 also resulted in actuation of SFAS channels 2 and 4. The actuation of SFAS channels 2 and 4, in turn, affected decay heat loop 2, the operating loop.

Because the initiating event was a loss-of-power event, all five levels of the SFAS were actuated [i.e., level 1, high radiation; level 2, high-pressure injection; level 3, low-pressure injection; level 4, containment spray; and level 5, ECCS recirculation]. Actuation of SFAS levels 2 and 3 resulted in containment isolation and loss of normal RHR pump suction from the RCS hot leg 2. Actuation of SFAS level 3 aligned the RHR pump 2 suction to the BWST in the low-pressure injection mode. Actuation of SFAS level 5 represents a low level in the BWST; thus, when level 5 was actuated, ECCS operation was automatically transferred from the injection mode to the recirculation mode. Consequently, RHR pump 2, the operating pump, was automatically aligned to take suction from the containment sump rather than from the BWST or the RCS. Because the emergency containment sump was dry, suction to the operating RHR pump was lost. As a result, RHR capability was lost for 2.5 h, the time required to vent the system. Furthermore, since decay heat loop 1 was down for maintenance, it was not available to reduce the time required to restore RHR. The RCS temperature increased to 77°C during the incident.

The extended loss of DHR capability at Davis-Besse 1 was caused by three somewhat independent factors, any one of which, if corrected, could have precluded this event. The three factors follow.

1. Inadequate procedures and/or administrative controls were contributing factors that led to this extended loss of RHR capability. The HPI and containment spray pumps had been deactivated to preclude their inadvertent actuation during refueling. If SFAS level 5 had also been bypassed or deactivated, or if the emergency sump isolation valves had been closed and their breakers opened, this event would have resulted in a minor interruption of decay heat flow.

2. If maintenance activities had been less extensive, or possibly better coordinated, this event could have been ameliorated or avoided. If activities had been restricted so that two SFAS channels would not be lost by a single event (e.g., serving channels 1 and 3 from separate sources), the loss of DHR capability would not have occurred. In addition, if a backup RHR system had been readily available, consequences of the loss of the operating RHR loop would have been lessened.

3. The two-out-of-four SFAS logic used at Davis-Besse is also somewhat more susceptible to spurious actions than other logic schemes, such as a one-out-of-two-taken-twice logic. This susceptibility is increased when two SFAS channels are served from one source. As described in the event, when channels 1 and 3 were lost, all five levels of SFAS were actuated. This could have been avoided if channels 1 and 3 had

been served from independent sources, or by use of a one-out-of-two-taken-twice logic that requires coincident actuation or loss of power of one even-numbered and one odd-numbered SFAS channel.

The PWRs are most susceptible to losing DHR capability when steam g tors, or other means of removing decay heat, are not readily available. Such conditions often occur when the plants are in a refueling or cold shutdown mode, and when concurrent maintenance activities are being performed. The risk and frequency associated with this type of event dictates investigation of other PWRs. Actions required of PWR licensees are listed in IE Bulletin 80-12, *Decay Heat Removal System Operability*, dated May 9, 1980. These actions included (1) review of each facility for all RHR degradation events experienced, especially those similar to the Davis-Besse event; (2) review of the facility's hardware capability for prevention of RHR loss events; (3) analysis of procedures for safeguarding against loss of redundancy and diversity of RHR capability; and (4) analysis of procedures for adequacy of responding to RHR loss events.

D.8 Both RHR Room Coolers for LPCI Room Inoperable at Brunswick 2*

(LER 80-01, Abstract 210, January 17, 1980)
(LER 80-33, Abstract 212, February 15, 1980)

While the reactor was near full power and the 2B RHR room cooler for the LPCI room was under a limiting condition for operation (LCO), the 2A RHR room cooler, serving the same room, became inoperable when its circuit breaker tripped for an undetermined reason. Because both RHR room coolers were inoperable, both LPCI loops were declared inoperable. The 2A RHR room cooler was returned to service in 15 min; the breaker was reset and closed, and the room cooler successfully run.

The significance of this event lies in the loss of the total LPCI system function, though it was an administratively declared loss rather

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*This discussion also appears in Appendix D of Task 3 on service water systems.

than an actual loss caused by, for example, a mechanical failure. Had the LPCI been demanded, it could have started and run until the LPCI inverters failed because of overheating, though certainly not long enough to complete RCS cooldown from near full power.

D.9 Reactor Vessel Vent Eductor System Causes RHR Pump Cavitation at Beaver Valley 1

(LER 80-02, Abstract 257, January 17, 1980)

With RCS at 38°C and atmospheric pressure, the RCS level at midloop with RHR flow at 0.1 m^3/s, the reactor vessel vent eductor was placed in service in preparation for refueling. A low RHR flow alarm was received and low flow and low RHR pump motor current were indicated. A second RHR pump was started and became air bound, as had the first RHR pump. The pumps were vented, and core flow was rapidly restored.

Putting the vessel vent eductor system into service was the root cause of this incident. The eductor caused a negative pressure in the RCS because the eductor air volume was greater than the influent air. This overcame the differential pressure in the steam generator U-tube section, and the excess steam generator water and entrained air drained into the loops. This entrained air then caused the pumps to become air bound. The event is significant because the loss of the two pumps caused the loss of total RHR system function.

This problem was corrected by revising the procedure to include constant venting of the RHR pumps when putting the vessel vent eductor into service. This had been a recurring problem when the loops were drained to midspan while RHR flow rate was twice the flow rate in this event, but the lower flow rate had not (until this event) been a problem. Engineers began to research a method to constantly vent the RHR pumps automatically.

D.10 Loss of Essential Bus Isolates RHR System at Davis-Besse 1

(LER 79-67, Abstract 326, June 28, 1979)

An accidental short circuit caused the loss of essential bus Y4 which supplies power to SFAS channel 4 (and other systems) causing channel 4 bistables to trip as designed. It also closed valve DH 11, isolating the RHR system. With no RCPs in operation, this was a violation of Technical Specifications.

RHR flow was not essential because of low decay heat level. RHR pump 1-2 was manually shut down for its own protection, bus Y4 was supplied from an alternate source, and the affected safety systems were reset. DH 11 was reopened, and RHR flow was reestablished 18 min after the loss of bus Y4. A blown inverter fuse was replaced, and Y4 bus was returned to normal supply.

This event is significant because it resulted in the actuation of the SFAS and an 18-min loss of the total RHR system function.

Appendix E

LISTING OF NSIC LER SEARCH FOR KEY PHRASES FOR DECAY HEAT REMOVAL REPORT

Search abstract No.	Unit	LER No.	Title
1	Sequoyah 1	81-61	RHR Miniflow Valve Fails
2	Peach Bottom 2	81-31	RCS Exceeds 212°F Without Containment Integrity
3	Quad-Cities 2	81-09	RHR Containment Cooling Valve Fails to Close
4	Cooper	80-47	RHR Suppression Pool Cooling Valve Fails to Operate
5	Vermont Yankee	81-15	RHR SW Pump Breaker Fails to Close
6	Browns Ferry 3	81-22	Containment Cooling Inoperable When Valve Fails
7	Davis-Besse 1	81-23R	Update on Decay Heat Cooler Valve Failure to Open
8	Ginna	81-11	RHR Pump Seal Cooler Fitting Leaks
9	Calvert Cliffs 2	81-04	Shutdown Cooling Flow Lost
10	Brunswick 2	81-49	Two RHR Heat Exchangers Fail Flow Test
11	Kewaunee	81-13	Weld Cracks on Sump Vent Line
12	Zion 1	81-16	RHR Control Valve Fails to Close
13	Salem 2	81-05	All Reactor Coolant and RHR Pumps Deenergized
14	Davis-Besse 1	81-24	Decay Heat Flow Lost When Pump Power Interrupted

Search abstract No.	Unit	LER No.	Title
15	Brunswick 1	81-52	Corroded Contacts Cause RHR Pump Trip
16	Monticello	81-12	RHR Isolation Valve Exceeds Leak Rate Limit
17	Kewaunee	81-12	Containment Spray Train Inoperable Due to Breaker Trip
18	Browns Ferry 2	80-53R	Update on RHR Heat Exchanger Leak
19	Monticello	81-05	RHR SW Heat Exchanger Control Valve Fails
20	Brunswick 1	81-32	RHR Heat Exchanger Inoperable Due to Bowed Baffle Plate
21	Brunswick 1	81-00S	Shutdown Cooling Lost
22	Quad-Cities 1	81-09	RHR Service Water Pump Fails to Meet Flow Requirement
23	Rancho Seco	81-21	BWST Valve Fails to Open
24	Salem 1	81-37	RHR Pump Inoperable Due to Leak
25	Cooper	81-03	RHR Valve Starter Fails to Deenergize
26	Brunswick 1	81-46	RHR Snubber Shaft Breaks
27	Beaver Valley	IE 81-09	Degradation of RHR System
28	Zion 2	81-03	RHR Train Capability Degraded by Open Valve
29	Monticello	81-02	Single Check Valve Failure Stops Both RHR SW Loops
30	Browns Ferry 3	81-16	RHR Heat Exchanger Removed for Maintenance

Search abstract No.	Unit	LER No.	Title
31	No plant	NUREG-0705	Identification of New Unresolved Safety Issues Relating to Nuclear Power Plants - Special Report to Congress
32	Quad-Cities 1	81-07	LPCI Inoperable Because of Several Shorted Conductors to Recirculation Pump Discharge Valve Motor
33	Davis-Besse 1	80-30R	Update on Low RCS Water Level
34	Rancho Seco	81-10	RHR Piping Does Not Meet Stress Limits
35	Crystal River 3	80-36	RHR Pump Throttle Valve Fails to Control Flow
36	Zion 1	81-04	RHR Miniflow Control Valve Switch Fails
37	Crystal River 3	81-16	Decay Heat Pump Discharge Valve Breaker Trips
38	Hatch 1	81-15	Two RHR and One Core Spray Valve Leak Rates Exceed Limit
39	Hatch 1	81-14	Seven RHR and Two HPCI Valve Leak Rates Exceed Limit
40	Browns Ferry 3	81-13	Shutdown Cooling System Valve Breaker Trips
41	No plant	IE 80-21	Valve Yokes Supplied by Malcolm Foundry Company, Inc., Cracked
42	Salem 1	81-17	RHR Pump Removed for Test
43	Davis-Besse 1	IE 80-44	Actuation of ECCS in the Recirculation Mode While in Hot Shutdown
44	Davis-Besse 1	IE 80-41	Failure of Swing Check Valve in the Decay Heat Removal System

Search abstract No.	Unit	LER No.	Title
45	Brunswick 2	81-19	Shutdown Cooling Supply Isolation Valve Fails to Open
46	Surry 1	80-10R	Update on RHR Pipe Support Modifications
47	Duane Arnold	81-07	RHR Service Strainer Plugs Up
48	Peach Bottom 3	81-10	RHR Logic Control Fuse Blows Repeatedly
49	Browns Ferry 3	80-56	RHR Pump Breaker Control Relay Found Dropped Out
50	Hatch 2	80-149R	Update on RHR Pump Isolation Valve Leakage
51	Sequoyah 1	81-21	All Core Cooling Paths Temporarily Lost
52	Crystal River	80-58	Decay Heat Pump Discharge Throttle Valve Fails to Control Flow
53	McGuire 1	81-07	Draining SW Header Causes RHR Train To Be Inoperable
54	University of Missouri	None	Convective Cooling Loop Isolation Valve Fails to Fully Open
56	McGuire 1	81-10	Broken Air Line on RHR Valve
57	McGuire 1	81-06	Shutdown Cooling System Pump Seal Leaks
58	Browns Ferry 3	81-08	RHR Pump Inoperable
59	Kewaunee	81-03	RHR Heat Exchanger Inlet Valve Overloads Trip
60	Indian Point 2	81-03	Component Cooling Water Valve Fails to Open
61	Calvert Cliffs 2	81-03	Shutdown Cooling Flow Stopped

Search abstract No.	Unit	LER No.	Title
62	Brunswick 2	81-10	RHR Inoperable
63	Browns Ferry 3	80-27R	Update on RHR Pump Failure
64	Cooper	80-52	RHR Heat Exchanger Discharge Valve Inoperable
65	Davis-Besse 1	81-04	Decay Heat Pump Fails to Start
66	Farley 1	81-01	Shutdown Cooling System Flow Lost
67	Millstone 1	80-19R	Update on Condenser Nozzle Weld Cracks
69	Cooper	80-50	LPCI Injection Valve Fails to Open
70	Arkansas Nuclear 1	80-07R	Update on RHR Pump Material Discrepancy
71	Quad-Cities 2	80-39	RHR Valve Fails to Open
72	Brunswick 2	80-117	RHR Pump Inoperable
73	Browns Ferry 3	80-15R	Update Report on RHR Pump Seal Heat Exchanger
74	Browns Ferry 3	80-31R	Update Report on RHR Pump Seal Heat Exchanger
75	Dresden 3	80-48	Containment Isolation Valve Fails to Close
76	Peach Bottom 2	80-34R	Update Report on RHR Pump Room Cooler Failure
77	Farley 1	80-80	Spurious Signal Causes RHR Flow Loss
78	Cook 2	81-01	RHR Pump Control Power Lost
79	Cooper	80-22R	Update on RHR Radwaste Discharge Valve Failure

Search abstract No.	Unit	LER No.	Title
80	Brunswick 1	80-73	Remote Shutdown Panel RHR Flow Indicator Fails
81	FitzPatrick	80-38	Update Report on Containment Isolation Valve Failure
82	Ginna	80-10	RHR Thermowell Weld Leaks
83	Quad-Cities 2	80-33	RHR Logic Circuit Inoperable
84	Hatch 1	80-118	RHR Inboard Injection Isolation Valve Leaks
85	Peach Bottom 2	80-37	Maximum Cooldown Rate Exceeded by 2°F
86	Brunswick 2	80-107	Reactor Coolant Temperature Exceeds Limit During Shutdown
87	Zion 1	80-54	Shutdown Cooling Pump System Inoperable
88	Cooper	80-42	RHR Heat Exchanger Inoperable
89	Peach Bottom 2	80-34	RHR Pump Room Cooler Fails to Start
90	Farley 1	80-77	Shutdown Cooling System Loop Inoperable
91	Arkansas Nuclear 1	80-11R	Update Report on 46 Inoperable Pipe Hangers
92	Turkey Point 4	80-14	RHR Piping Support Structure Deficient
93	Quad-Cities 2	80-37	RHR Pump Inoperable
94	Brunswick 2	80-75R	Update on RHR SW Loop Inoperable
95	Browns Ferry 3	80-49	RHR Pump Trips on Overcurrent
96	Three Mile Island 2	80-49	Unauthorized ASME Stamp Found on RHR Valve

Search abstract No.	Unit	LER No.	Title
97	Calvert Cliffs 1	80-58	Shutdown Cooling Flow Lost
98	Hatch 2	80-152	HPCI and RHR Valve Leak Rates Exceed Limits
99	Robinson 2	80-29	Leak in RHR Pump Suction Isolation Valve
100	Cooper	80-37	RHR Valve Fails to Close
101	North Anna 1	80-93	Incorrect RHR Valve Weights Used in Stress Analysis
102	Calvert Cliffs 1	80-62	LPSI Pump Trips Off
103	Browns Ferry 3	80-48	EECW Valve Fails Closed
104	Brunswick 2	80-97	Torus Level Exceeds Limit
105	Hatch 2	80-155	RHR and Condensate Isolation Valves Leak
106	Sequoyah 1	80-180	Seismic Event Could Fail Containment Spray Pumps
107	Brunswick 2	80-75	RHR SW Pressure Alarm Fails
108	Hatch 2	80-149	RHR Pump Suction Valve Leaks
109	Kewaunee	80-26R	Update Report on Low RWST Level
110	Quad-Cities 1	80-27	Ten Pipes Not Supported Properly
111	Quad-Cities 2	80-24	RHR Containment Cooling Valve Fails to Open
112	Three Mile Island 1	80-17	Unqualified Brakes Used on Containment Purge Valve Operators
113	Humboldt Bay 3	80-06	Screen Wash Pump Discharge Valve Leaks

Search abstract No.	Unit	LER No.	Title
114	Duane Arnold	80-52	RHR Alarm Fails to Clear
115	Hatch 2	80-148	Torus Water Level Below Limit
116	Quad-Cities 2	80-27	RHR Pump Fails to Start
117	Pilgrim 1	80-82	Containment Isolation Valve Fails to Close
118	Farley 1	80-57	RHR Incorrectly Isolated
119	Pilgrim 1	80-67	RHR System Fails to Meet Safe Shutdown Earthquake Criteria
120	Brunswick 2	80-66	RHR SW Pumps Fail to Start
121	Peach Bottom 2	80-17	RHR Pump Room Cooler Fails to Start
122	Peach Bottom 3	80-16	Pipe Support Stresses Exceed Limits
123	Brunswick 2	80-59	RHR Flow Instrumentation Fails
124	Browns Ferry 2	80-34	RHR Heat Exchanger Leaks
125	Vermont Yankee	80-27	RHR SW Check Valve Fails to Seat
126	Brunswick 2	80-30	RHR Heat Exchanger Rib Plate Buckles
127	Beaver Valley 1	80-36	Weld Fails on RHR Piping
128	Crystal River 3	80-36	Automatic Control of DHR Pump Throttle Valve Fails
129	FitzPatrick	80-62	RHR SW Pumps Fail to Provide Sufficient Flow
130	Cook 2	80-32	Inadequate Venting Causes RHR Pump To Be Inoperable
131	Brunswick 1	80-69	RHR Flow Indicator Gives False Readings

Search abstract No.	Unit	LER No.	Title
132	Browns Ferry 1	80-72	RHR Torus Return Valve Fails to Operate
133	Hatch 1	80-105	RHR SW Pump Vibrates Excessively
134	Beaver Valley 1	80-67	Component Cooling Water Line Support Insufficient
135	Peach Bottom 3	80-19	LPCI Injection Valve Fails
136	Dresden 2	80-33	LPCI Heat Exchanger Tubes Leak
137	Cooper 1	80-34	RHR Pump Inoperable Because of Failed Breaker
138	Hatch 1	80-102	RHR Loop Inoperable When LPCI Trips Because of High Ambient Temperature
139	Farley 1	80-36	RHR HX Discharge Valve Fails to Stroke Closed
140	Three Mile Island 2	80-25	Reactor Coolant Inlet Temperature Meter is Inoperable
141	Browns Ferry 3	80-27	RHR Pump Motor Trips on Overcurrent
142	Point Beach 2	80-07	RHR Systems Potentially Inoperable
143	Davis-Besse 1	80-57	DHR Flow Control Valve Closes Too Far
144	Hatch 1	80-76	Inadequate Pipe Supports Found on RHR System
145	Beaver Valley 1	80-49	CCW Supply Line Support Deficient
146	Cooper	80-13	Primary Containment Isolation Valve Timing Requirements Not Met

Search abstract No.	Unit	LER No.	Title
147	Cooper	80-16	RHR SW Booster Pump Inoperable
148	Hatch 1	80-61	RHR Minimum Flow Valves Closed Inadvertently
149	Hatch 1	80-62	Pavement Deflection Near Intake Structure
150	Cooper	80-22	Inboard Throttle Isolation Valve Fails to Operate
151	Browns Ferry 1	80-61	RHR Area Cooler Fan Thermal Overload Trips
152	Browns Ferry 2	80-33	RHR Heat Exchanger Leaks
153	Browns Ferry 2	80-35	RHR Check Valve Leaks
154	Browns Ferry 3	80-31	Cooling Water Flow to RHR Pump Seal Falls Below Limit
155	Browns Ferry 3	80-33	RHR Heat Exchanger on Crosstie Leaks
156	Browns Ferry 3	80-34	RHR Pump Area Cooler Fan Thermal Overload Relay Fails
157	Hatch 1	80-92	LPCI Inverter Trip Due to Room Cooler Failure Causes RHR To Be Inoperable
158	Hatch 1	80-84	RHR Service Water Loop Inoperable
159	Hatch 1	80-95	RHR Inboard Injection Valve Fails to Open
160	Davis-Besse 1	80-60	Decay Heat Pump Stopped During Refueling
161	Cooper	80-30	Suppression Pool Cooling Inboard Throttle Valve Fails

Search abstract No.	Unit	LER No.	Title
162	Zion 1	80-34	Potential Exists for Inadvertent Dilution With Insufficient Indication
163	Davis-Besse 1	80-49	Loss of DHR Flow Because of ESF Actuation
164	Calvert Cliffs 2	80-08	Containment Spray Bender Isolated for Repair of RHR Valve
165	Davis-Besse 1	80-43	Loss of DHR Flow
166	Davis-Besse 1	80-44	DHR Flow Stopped Due to False Reading
167	Beaver Valley 1	80-46	Component Cooling Water to RHR Heat Exchangers Embedment Plate
169	Davis-Besse 1	80-29	DHR Flow Lost
170	Davis-Besse 1	80-30	DHR Flow Lost Due to Low RCS Level
171	Davis-Besse 1	80-58	DHR Flow Lost Three Times
172	Hatch 1	80-57	Shutdown Cooling Suction Valve Fails to Open
173	North Anna 2	80-01	RHR Inlet Valve Closes
174	Browns Ferry 1	80-49	RHR Heat Exchanger Leaks
175	Rancho Seco	80-31	Shutdown Cooling System Pipe Support Requires Modification
176	Salem 2	80-06	Improperly Assembled RHR Pump Discovered
177	Hatch 2	80-85	RHR Service Water Pumps Fail
178	San Onofre 1	80-19	RHR Pump Shaft Cooling Water Line Pipe Discovered Missing
179	Hatch 1	80-45	RHR SW Pump Bearing Cooling Water Valves Found Closed

Search abstract No.	Unit	LER No.	Title
180	FitzPatrick	80-38	RHR Containment Isolation Valve Fails to Close
181	Brunswick 1	80-47	RHR Flow Indicator on Remote Shutdown Panel Fails
182	Hatch 1	80-47	RHR Pump Operability Test Not Performed Correctly
184	FitzPatrick	80-35	RHR System Discharge to Radwaste Valve Fails to Open
185	Hatch 1	80-53	Operation Within a Single Failure Criteria Region Discovered
186	Peach Bottom 2	80-00S	RHR Pump Motor Fails
187	San Onofre 1	80-15	4-kV and 480-V Onsite and Offsite Power Lost
188	Hatch 2	80-83	Limit Switches on RHR Pump Valves Are Broken
189	Browns Ferry 3	80-15	Restriction in Cooling Water Line for RHR Pump Seal Heat Exchanger
190	Beaver Valley 1	80-31	RHR Fails Due to False High Pressure Signal
191	Browns Ferry 1	80-43	RHR Heat Exchanger Leaks
192	Hatch 1	80-39	RHR SW Pump Cooling Does Not Meet Single Failure Criterion
193	Peach Bottom 3	80-09	RHR Pump Drain Line Pipe Cracks
194	Rancho Seco	80-27	Low Oil Level in RHR Snubber Found
195	Monticello	80-19	RHR SW Pump Fails to Meet Requirements
196	Davis-Besse 1	80-35	DHR Cooler Outlet Valve Fails to Open Completely

Search abstract No.	Unit	LER No.	Title
198	Beaver Valley 1	80-23	Total RHR Flow Lost
199	Beaver Valley 1	80-22	Total RHR Flow Lost
200	Rancho Seco	80-17	BWST to DHR Isolation Valve Fails to Close
201	Maine Yankee	80-07	RHR Valves Left Open and/or Unlocked Following Refueling
202	Oconee 1	80-08	Unqualified LPI Valve Operator Installed in Containment
203	Three Mile Island 1	80-14	Errors Found in Seismic Analysis of Shutdown Cooling System
204	Arkansas Nuclear 1	80-07	Potential for Failure of RHR Pump Mating Wear Rings Discovered
205	Quad-Cities 2	80-08	Two RHR Delay Timers Fail to Actuate the Timer Contacts
206	Browns Ferry 1	80-25	RHR Injection Valve Fails to Close
207	Salem 1	79-59	RHR Flow Lost
208	Vermont Yankee	80-09	RHR SW Pumps Fail to Start
209	Cooper	80-04	Control Power to RHR Valve Fails
210	Brunswick 2	80-01	Both LPCI Loops Inoperable
211	Brunswick 1	80-08	RHR SW Pump Fails to Start
212	Brunswick 2	80-33	Both RHR Room Coolers Inoperable for 15 min
213	Quad-Cities 1	80-03	RHR Suppression Chamber Dump Valve Breaker Trips
214	Brunswick 1	80-16	RHR Pump Trips

Search abstract No.	Unit	LER No.	Title
215	Browns Ferry 3	80-06	Flow Through RHR Pump Seal Coolers Falls Below Limit
216	Oconee 1	80-06	DHR Cooler Outlet Valve Fails to Open
217	Browns Ferry 2	80-14	RHR Not Properly Tested
218	Crystal River 3	80-15	Decay Heat Suction Valve Closes Inadvertently During Testing
219	Hatch 2	80-24	RHR Service Water Pump Flow Restricted by Debris
220	Peach Bottom 3	80-03	LPCI Injection Valve Fails to Open
221	Kewaunee	80-15	Containment Sump Recirculation Valve Fails to Open
222	Brunswick 2	80-34	Three Snubbers Found Inoperable
223	Brunswick 1	80-21	RHR SW Pump Fails to Start
224	Hatch 2	80-12	RHR Minimum Flow Lines Not Adequately Supported
225	Rancho Seco	80-08	Pipe Support U-Bolt Bracket Found Missing
226	Browns Ferry 3	80-05	RHR Valve Found Inoperable
227	Crystal River 3	80-08	DHR Pump Discharge Valve Fails to Control Flow in Automatic
228	Calvert Cliffs 1	80-11	Shutdown Cooling Suction Header Relief Valve Lifts Below Set Point
229	Pilgrim 1	80-06	RHR System Snubber Fails
230	North Anna 1	80-02	Stress on Some Supports and Snubbers Exceed Design Allowable

Search abstract No.	Unit	LER No.	Title
231	Quad-Cities 2	80-02	RHR SW Vault Penetration Leaks
232	Rancho Seco	80-05	DHR Pump Seal Leaks
233	Brunswick 2	80-19	RHR SW Pumps Fail Flow Test
234	Hatch 1	8/29/79 LTR	Leak Rate of RHR Valve High but Below Limit
235	Brunswick 1	79-58	Core Spray/RHR Pump Running Annunciator Energized Incorrectly
237	Surry 1	80-10	RHR Line Outside Containment Fails to Meet Stress Criteria
238	Browns Ferry 3	80-20	RHR System Control Valve Trips in Mid Position
239	Rancho Seco	80-02	DHR System Pipe Support Fails to Meet Criteria
240	Hatch 1	80-03	Opening and Closing Time of RHR Valve Off Limits
241	Davis-Besse 1	79-125	Decay Heat Cooler Component Cooling Water Valve Fails
242	Brunswick 1	79-113	RHR Torus Suction Valve Left Closed During Reactor Startup
243	Duane Arnold	79-25	Piping Seismic Restraints and Hanger Found to be Damaged
244	Watts Bar 1	79-	RHR System Flanges Leak
245	Davis-Besse 1	79-117	A Snubber on Decay Heat System Found Upside Down
246	Davis-Besse 1	79-108	DHR Pump Fails to Start
247	Cooper	79-42	RHR Pump Breaker Found Not Charged
248	Crystal River 3	79-110	DHR Cooler Outlet Temperature Indicator Inoperable

Search abstract No.	Unit	LER No.	Title
249	FitzPatrick	79-103	RHR System Pipe Hanger Inoperable
251	Palisades	79-44	Two Shutdown Cooling System Valves Left Closed After Testing
252	Browns Ferry 1	79-32	RHR SW Pipe Supports Discovered Inoperable
253	Monticello	79-21	Snubber Anchor Bolt Safety Factor Less Than Required
255	Brunswick 2	79-98	RHR Logic Power Fails
256	FitzPatrick	80-07	RHR Loop Support Inoperable
257	Beaver Valley 1	80-02	Reactor Vessel Vent Eductor System Causes RHR PW Cavitation
258	Quad-Cities 1	79-35	Pipe Supports Discovered Inoperable
259	Hatch 2	79-118	Valve for RHR Pump Suction from Suppression Chamber Fails to Open
260	Peach Bottom 3	79-33	RHR Pump Trips with Two Others Blocked Out for Repair
261	Crystal River 3	79-104	Shutdown Indicator for DHR Cooler Outlet Temperature Fails
262	Cook 1	79-57	RHR Isolation Valve Fails
263	North Anna 1	79-145	RHR Flow Lost for 5 min
264	Arkansas Nuclear 2	79-36	Shutdown Cooling Heat Exchanger Tube Leaks
265	Quad-Cities 2	79-20	Hydraulic Snubbers Fail
266	Quad-Cities 2	79-25	RHR Pump Suction Valve Fails
267	Calvert Cliffs 2	79-38	Shutdown Cooling Stopped for 3 min

Search abstract No.	Unit	LER No.	Title
268	Three Mile Island 1	79-17	Piping Support Fails to Meet Seismic Criteria
270	FitzPatrick	79-91	RHR Pump and Valve Interlocks Fail
271	Robinson 2	79-39	Inadequate Pipe Supports Discovered
272	Trojan	79-15	Weak Walls Holding Pipe Supports Discovered
273	Calvert Cliffs 1	79-53	Containment Spray System Reduced to One Train
274	FitzPatrick	79-87	Inadequate Pipe Supports Identified
276	Salem 1	79-56	Update Report on RHR Pump Flow Exceeding Design Runout Flow
280	Crystal River 3	79-95	Decay Heat Cooler Outlet Temperature Indicator Fails
281	Crystal River 3	79-89	Decay Heat Cooler Outlet Temperature Indicator Fails
282	Farley 1	79-36	RHR Relief Isolation Valves Close During Testing
283	Pilgrim 1	79-37	RHR Valve Stem Guide Slips Upward
284	FitzPatrick	79-81	Four Pipe Supports Considered Inoperable
285	Brunswick 1	79-76	RHR SW Valve Fails to Open
286	Zion 1	79-68	RHR Cold Leg Crosstie Line Support Seismic Design Deficient
287	Peach Bottom 3	79-27	Two High Pressure Service Water Pumps Out of Service

Search abstract No.	Unit	LER No.	Title
288	FitzPatrick	79-57	Five Pipe Supports Found To Be Inoperable
290	Connecticut Yankee	79-08	RHR Piping Seismic Support Found To Be Missing
291	Salem 1	79-56	RHR Pump Flow Exceeds Design Run-out Flow
292	Maine Yankee	79-18	RHR System Secured Without an Operable Steam Generator
293	Brunswick 1	79-60	RHR SW Differential Pressure Transmitter Fails
294	Peach Bottom 2	79-41	RHR Injection Valve Breaker Inadvertently Opened
295	Hatch 1	79-75	AE Equations for RHR SW Pump Rated Flow Incorrect
296	Peach Bottom 2	79-43	Two HPSW Piping Support Anchors Installed Incorrectly
297	Pilgrim 1	79-30	RHR Valve Breaker Trips on Over-load
298	Zion 2	79-38	RHR Pump Suction Valve Fails to Open
303	Hatch 1	79-62	Operating Times of RHR Valve Exceed Limit
304	Hatch 1	79-51	Two RHR Isolation Valves Made Inoperable
306	Rancho Seco	79-07	Six DHR System Supports Fail to Meet Design Criteria
307	Hatch 1	79-64	RHR SW Pump Restraints Found to be Deficient
308	Quad-Cities 2	79-13	RHR Pump Suction Valve Thermal Overload Trips

Search abstract No.	Unit	LER No.	Title
310	Hatch 1	79-55	Relay for Starting RHR Pump Fails to Energize
311	Monticello	79-13	RHR Loop Torus Cooling Injection Valve Fails to Open
312	Beaver Valley 1	79-18	RHR Pump Trips While Transferring Buses
313	Hatch 2	79-73	RHR Flow Indicator in Remote Shutdown Panel Inoperable
314	Peach Bottom 3	79-23	Two Support Anchors Fail to Torque Properly
315	Pilgrim 1	79-28	RHR Valve Stem Guide Key Found To Be Sheared
316	Vermont Yankee	79-13	RHR Subsystem Logic Relay Fails to Energize
317	Hatch 2	79-72	Controller Fails to Automatically Operate RHR SW Valve
318	Hatch 1	79-50	Two of Three RHR Loops Inoperable
319	Brunswick 1	79-42	Support Beams for One RHR System Snubber Found to be Deficient
320	FitzPatrick	79-42	RHR SW Pump Output Less Than Limit
321	Pilgrim 1	79-22	Two Pipe Supports Fail to Meet Seismic Criteria
322	FitzPatrick	79-40	RHR Discharge Check Valve Not Fully Closed
323	Hatch 1	79-42	Standby Liquid Control Valve Leakage Exceeds Limit
324	Hatch 2	79-56	Torus Water Level Exceeds Limit

Search abstract No.	Unit	LER No.	Title
326	Davis-Besse 1	79-67	Loss of Essential Bus Isolates RHR System
327	Three Mile Island 1	79-13	Crack Found in Decay Heat Piping

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16. ABSTRACT (200 words or less)

This report reviews and evaluates events placed in the NSIC file involving the removal of decay heat in U.S. commercial boiling- and pressurized-water reactors from June 1979 through June 1981. The information was collected from operating experiences, licensee event reports, system designs in safety analysis reports, and other regulatory documents. The results were collated and analyzed according to safety significance and cause of event.

Thirty-eight reported events in these 2.1 years meet the criteria for safety significance. Steam bubble formation in the reactor vessel head during natural circulation cooldown at St. Lucie 1 was the most significant event; operator awareness of the possibility of this occurrence and preparedness for dealing with it was the most important recommendation. Cavitation of residual heat removal pumps during decay heat removal operation was the most common potentially significant event. Davis-Besse 1 had several instances in which an inadvertent signal to the safety features actuation system caused the operating residual heat removal pumps to align to the dry sump causing pump cavitation.

17. KEY WORDS AND DOCUMENT ANALYSIS

17a. DESCRIPTORS

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- Pump Cavitation
- Nuclear Power Plant Operating Experience

17b. IDENTIFIERS/OPEN-ENDED TERMS

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