Attachment 2

NIAGARA MOHAWK

NMP-24179

NIAGARA MOHAWK POWER CORPORATION/301 PLAINFIELD ROAD, SYRACUSE, N.Y. 13212/TELEPHONE (315) 474-1511

April 15, 1987

Mr. Robert Keller
United States Nuclear
Regulatory Commission
Region I
631 Park Avenue
King of Prussia, PA 19406

Dear Mr. Keller:

This letter concerns the NRC Hot License examinations administered to four (4) Nine Mile Point Unit 1 Senior Reactor Operator candidates during the week of April 6, 1987. Mr. Frank Crescenzo of Region I was the lead examiner and was assisted by Mr. Alan Howe and Ms. Tracy Lumb.

The written exam was administered on April 7, 1987, with the operating exams on April 8 and 9, 1987. The exit meeting took place on Thursday, April 9, 1987.

We felt that this exam was well written and that in most cases, the questions and answer key accurately reflected the training material and plant procedures cited as references. Most of our comments were resolved during the exam critique. Please see attached for those unresolved comments.

We appreciated the efforts of Mr. Crescenzo, Mr. Howe and Ms. Lumb for ensuring that this examination was conducted in a very professional manner.

Please feel free to contact me if you need any additional information.

Very truly yours,

Thomas E. Lempges

Thomas E. Lempges
Vice President - Nuclear Generation

TEL/ph

Attachments: (3)

5.04	Part a	This question can be interpreted to discuss the effects of control rod worth instead of equilibrium reactivity conditions. Adequate discussion on control rod worth should be accepted for an alternate full credit answer.
	Part b.2	This part of the question can be interpreted to discuss the reverse power effect, a frequently observed condition at NMP1. This should be accepted as an alternate full credit answer.
5.08	Part d	Request wider tolerance band for answer to compensate the loss of accuracy in the event the candidate used the mollier diagram to obtain answer.
5.10	Part a	Explanation should also be accepted for full credit if candidate defends answer with respect to carry under relative to power level and water level.
6.01		The terminology in this question confused the candidates. The term 'Group' is not used in the stated reference, Operations Technology, chapter 9C. Recommend retaining parts A and E as valid question, and reject parts B, C, and D.
6.02	Part b.1	The Hi and Lo reactor vessel alarms do not come off of the Hi/Lo LO-LO Rosemount level instrumentation. They come off of the GEMAC level recorder. Refer to Operations Technology chapter 3, page 2. In addition, the question does not ask for the normal range. Therefore we recommend the normal range and alarms not be required for full credit.
	Part b.2	Candidates may provide additional plant conditions that occur at Lo level of 53 inches and LO-LO level of 5 inches and should not be penalized for providing additional information.

6.06	Part a.2	The "Condensate Storage Tank" should be accepted as an alternate full credit answer since this is the source of the condensate transfer system.
6.07	Part b	Question does not ask for how many ways to stop ADS blowdown. Recommend accepting any one answer for full credit.
6.09		Concerning "CRD Drive Water Pressure", OP-5 page 8 states the pressure as Reactor pressure plus 260 psi. Concerning "CRD Cooling Water Pressure", OP-5 page 8 states 17-40 psi above reactor pressure. The above should be accepted as full credit answers.
6.10		Current SOER 87-2 discusses draining of reactor vessel via test lines. Recommend accepting this as full credit answer. See attachment #2.
6.11	Part a	Although the answer key is valid, we are not sure the question solicits this much detail and therefore ask for redistribution of weighting values in answer key.
Section 7		General Comment Lesson objectives allow candidates to access operating procedures except for immediate operator actions and entry conditions to EOP's.
7.01	Part a	"Loss of cooling medium to pumps" should be considered a full credit answer (General answer versus specific).

7.04	Part c	Recommend considering the following alternate answers for full credit.
		1. Manufacture recommends minimizing the idle time to prevent seal damage. See attachment #3.
		2. Considerations for cold water accidents as stated in FSAR and Tech. Specs.
7.06		Recommend expanding key to accept <u>any</u> six of the nine steps in SOP-9.
		Recommend accepting as basis for isolating the reactor vessel as, "Conserve vessel inventory."
		Recommend accepting as basis for Reactor Scram as, "Placing reactor in safe condition prior to evacuating control room."
7.09	Part a	Recommend accepting any four parameters from OP-13 section D, start-up procedure.
8.01		Since candidates were not given 10CFR50.54(X), answer per A.P.-4.0 page 2 should be accepted. In addition key reflects 1.0 points but question is worth 1.5 points.

8.07	Part b	Technical Specification 3.2.7 allows operation if inoperable MSIV is shut. Therefore, "If the Problem..." should not be required for full credit. "Shutting the MSIV" should constitute a full credit answer.
8.08	Part a	Bases for Condenser low vacuum scram is located in Tech Specs section 2.2.2 page 24 and does not discuss cladding safety limits being exceeded.
8.10		Lesson objectives state 10CFR50.72 would be given to candidates since 10CFR50.72 is provided in the control room.
8.11		Objective referenced allows candidates to reference tech specs. This table and supporting notes were not provided. Recommend deletion of this question.

ATTACHMENT 2

INPO

Institute of
Nuclear Power
Operations

Suite 1500
1100 Circle 75 Parkway
Atlanta, Georgia 30339

Significant Operating
Experience Report

87-2

☐ Green-
Normal Attention

■ Yellow-
Prompt Attention

☐ Red-
Immediate Attention

LIMITED DISTRIBUTION

March 19, 1987

INADVERTENT DRAINING OF REACTOR VESSEL TO SUPPRESSION POOL AT BWRs

EVENTS:

UNIT(TYPE):	PEACH BOTTOM 2 (BWR)	RIVER BEND (BWR)
DOC NO/LER NO:	50-277/85020	50-458/85008-1
EVENT DATE:	9/24/85	9/23/85
NSSS/AE:	GE/BECHTEL	GE/STONE AND WEBSTER
UNIT(TYPE):	SHOREHAM (BWR)	WASHINGTON NUCLEAR PLANT-2 (BWR)
DOC NO/LER NO:	50-322/85031	50-397/85030
EVENT DATE:	7/24/85	5/7/85
NSSS/AE	GE/STONE AND WEBSTER	GE/BURNS AND ROE
UNIT(TYPE):	LASALLE 2 (BWR)	GRAND GULF 1 (BWR)
DOC NO/LER NO:	50-374/84009	50-416/NA
EVENT DATE:	8/15/83	4/3/83
NSSS/AE:	GE/SARGENT AND LUNDY	GE/BECHTEL
UNIT(TYPE):	CAORSO (BWR)	
COUNTRY:	ITALY	
EVENT DATE:	3/29/83	
NSSS/AE:	AMN-GE/GIBBS AND HILL	

REFERENCES:

1. INPO Significant Operating Experience Report (SOER): 85-1

2. INPO Significant Event Reports (SERs): 37-83, 37-83 Supplement 1, 37-83 Supplement 2, 72-84 Supplement 1, and 72-84 Supplement 2

3. INPO Operations and Maintenance Reminder (O&MR): 231

4. NRC IE Information Notice: 84-81

5. General Electric Company Service Information Letter No. 388 and Application Information Document 67, Residual Heat Removal Valve Misalignment During Shutdown Cooling Operation for BWRs 3/4/5 and 6.

6. Nuclear Safety Analysis Center Report No. 88, Residual Heat Removal Experience Review and Safety Analysis - Boiling Water Reactors

SUMMARY:

Inadvertent draining of the reactor vessel to the suppression pool has occurred at boiling water reactors on several occasions. All of these events have resulted from improper valve lineups in the residual heat removal system during the shutdown and refueling modes of reactor operation. In each event, rapid draining of the reactor vessel occurred. In all cases, draining was stopped before fuel was uncovered.

With fuel in the vessel, a vessel draining event has the potential to uncover the fuel, causing fuel cladding failure due to overheating. If this type of event occurred while the reactor vessel head and the gates between the reactor cavity and the spent fuel storage pool were removed, extremely high radiation dose rates could occur on the refueling floor due to loss of water in the spent fuel storage pool. If such an event coincided with the movement of irradiated fuel between the reactor vessel and the spent fuel storage pool, uncovering and damage to the fuel being moved could also occur.

DESCRIPTION:

Peach Bottom 2

The plant was in cold shutdown. Reactor water level was being controlled automatically by the "C" reactor feedwater pump bypass level control valve. The "A" residual heat removal pump was operating in the shutdown cooling mode. Earlier, the "C" residual heat removal pump had been operated in the shutdown cooling mode. When "C" pump operation was terminated, the "C" shutdown cooling suction valve was erroneously left open (see Figure 1). Subsequently, the operators received a request to operate the "A" pump in the full-flow test mode (suppression pool to suppression pool) to support a pump problem investigation. The reactor operator shut off the "A" residual heat removal pump and closed the "A" loop shutdown cooling suction valve. Next, he opened the loop full-flow test return valve to the suppression pool. Because the "C" shutdown cooling suction valve was still open, a path was immediately established for gravity flow of reactor water through the "C" shutdown cooling suction valve, the "C" residual heat removal pump, and the full-flow test return valve to the suppression pool.

As reactor water level decreased, an automatic isolation of the shutdown cooling system occurred on low reactor water level, terminating the reactor water level decrease. The level had dropped about 45 inches during the transient to about 180 inches above the top of the active fuel.

River Bend

Prior to initial criticality, the plant was in the refueling mode with fuel in the vessel. Operators were performing the Division 1, 18-month emergency core cooling system surveillance test. During the process of restoring residual heat removal loop "A" to the low pressure coolant injection standby lineup, the handswitch for the "A" residual heat removal shutdown cooling suction valve was turned to the "closed" position (see Figure 2). The suction valve requires about 250 seconds to stroke closed. Immediately after this, the operator turned the handswitch for the "A" residual heat removal suppression pool suction valve to the "open" position. The suppression pool suction valve opens in a maximum allowable time of about 100 seconds. The valves have contacts that "seal-in" the open or closed commands so that once they are in motion they must stroke either to the fully open or closed position before they can be reversed. Therefore, a flowpath was established for approximately four minutes that allowed water to drain from the reactor vessel to the suppression pool until the shutdown cooling suction valve fully closed.

At the beginning of the valve transient, reactor level was approximately +28 inches. By the time the shutdown cooling suction valve had stroked closed, vessel water level had lowered 50 inches causing a reactor protection system actuation. Approximately 12 feet of water remained over the top of the fuel when the reactor vessel draining was terminated. The shutdown cooling isolation valves did not close on the low reactor water level isolation signal because the breakers to these valves had been tagged opened for maintenance purposes. This was allowed because the automatic isolation function is not required in the refueling mode.

Shoreham

With the unit in cold shutdown, both residual heat removal loops were lined up for the shutdown cooling mode of operation. The "B" loop was in operation. The operator was in the process of realigning the "A" residual heat removal loop to the standby low pressure coolant injection mode of operation (see Figure 1). He initiated a "close" signal to the shutdown cooling suction valve and, a short time later, initiated an "open" signal to the suppression pool suction valve. The stroke times for these valves are in the range of 90-100 seconds. Therefore, both valves were partially open long enough to allow approximately 7,000 gallons of reactor vessel water to drain to the suppression pool. Water level dropped to the low-level scram setpoint. The shutdown cooling isolation valves automatically isolated, terminating the event.

Washington Nuclear Plant 2

The control room operator was in the process of securing one of the residual heat removal pumps and switching from the shutdown cooling mode of operation to the low pressure coolant injection lineup (see Figure 2). The operator had reviewed the procedure and was aware

that having the shutdown cooling suction valve and the suppression pool suction valve open simultaneously would result in draining reactor pressure vessel water to the suppression pool. The operator initiated a "close" signal to the shutdown cooling suction valve and, approximately 30 seconds later, initiated an "open" signal to the suppression pool suction valve. The stroke times for both valves are in the range of 90-100 seconds. As a result, both valves were partially open for about a minute allowing water from the reactor vessel to drain to the suppression pool. Water level decreased to the low-level scram setpoint. A reactor protection system actuation occurred, and the shutdown cooling isolation valves automatically closed. This terminated the reactor vessel draining.

LaSalle 2

During preoperational tests prior to initial fuel loading, the control rod drive system was in service to supply water to the reactor. Reactor water level was being controlled by draining through the residual heat removal shutdown cooling "B" loop piping to the suppression pool (see Figure 2). The normal drain path through the reactor water cleanup system to the condenser hotwell was out of service.

To avoid inadvertent isolation of the temporary drain path, control for the residual heat removal shutdown cooling isolation valves was transferred to the remote shutdown panel. At LaSalle, the design for the remote shutdown panel bypasses some emergency core cooling system interlocks when control is switched from the main control room to this panel. This includes bypassing the shutdown cooling low water level isolation signal.

As a new drain path was being established through the "A" residual heat removal shutdown cooling loop, operators started opening the shutdown cooling suction valve with the suppression pool suction valve open. This action established a drain path between the reactor vessel and suppression pool, and the vessel level decreased rapidly. Normally, the shutdown cooling suction valve is interlocked such that with the suppression pool suction valve, the suppression pool spray valve, or the full-flow test valve open, the shutdown cooling suction valve cannot be opened. However, because the interlock between these valves was bypassed when control was transferred to the remote shutdown panel, the suction valve opened and the level decreased rapidly to 32 inches above core level before draining was terminated by operator action.

Grand Gulf 1

The reactor was in cold shutdown. A shutdown cooling valve test had just been completed, and the residual heat removal system was being realigned to the low pressure coolant injection mode (see Figure 2). Reactor vessel water was at the normal level of 36 inches above reference zero, which is 160 inches above the top of the active fuel.

Loop "A" of the residual heat removal system was lined up in the low pressure coolant injection mode. Loop "B" was lined up in the shutdown cooling mode for a surveillance test. After completion of the test, the control room operator was in the process of returning the "B" loop to the low pressure coolant injection lineup. This required shutting the loop "B" shutdown cooling suction valve and opening the loop "B" suppression pool suction valve. Since a light bulb was burned out on the "open" indication for the shutdown cooling suction valve, the operator incorrectly assumed that it was already shut and opened the suppression pool suction valve. This opened a flow path from the reactor vessel to the suppression pool via the residual heat removal "B" loop. The reactor water level dropped approximately 50 inches before the shutdown cooling isolation valves closed automatically from the low reactor water level isolation signal.

Caorso-Italy

The unit was in cold shutdown following refueling. Maintenance work was being performed on the motor-operated "B" residual heat removal suppression pool suction valve (see Figure 1). Prior to beginning maintenance, the valve had been stroked to its normal full-open position from the control room, and the power leads had been disconnected at the motor operator for safety. Upon completion of the valve repairs, it was manually closed by maintenance personnel. Power leads to the motor operator were not reconnected at the time.

Sometime later, a valve lineup to the shutdown cooling mode was established from the control room in preparation for an overall primary containment leak rate test. The valve lineup for residual heat removal loop "B" required the shutdown cooling isolation valves, the shutdown cooling suction valve, and the residual heat removal low pressure coolant injection valve to be open; the residual heat removal low pressure coolant injection throttle valve to be partially open; and the suppression pool suction valve to be closed.

The control room handswitch for the "B" loop suppression pool suction valve had been left in the "open" position from the time maintenance on the valve had begun. Without the knowledge or concurrence of shift supervision, maintenance personnel restored power to the motor operator. With power restored, the valve commenced opening automatically and reactor water began draining through the "B" residual heat removal loop shutdown cooling isolation and suction valves and the suppression pool suction valve to the suppression pool. Reactor water level decreased 26 inches before automatic closure of the shutdown cooling isolation valves terminated the draindown.

SIGNIFICANCE:

These events are significant because of the potential for fuel damage and resulting high radiation fields, and because operator error was a common underlying cause. Operator inattention to the impact of unusual valve alignments and procedural deficiencies were contributors to the operator errors.

Rapid and uncontrolled draining of the reactor vessel could result in uncovering irradiated fuel in the reactor vessel. Such draining could also significantly reduce shielding over irradiated fuel stored in the spent fuel storage pool if the reactor vessel head and the gates between the reactor cavity and the spent fuel storage pool are removed. Radiation fields as high as several thousand R/hr could result. Additionally, irradiated fuel bundles could become uncovered if draining of the vessel occurred while they were being moved with the refueling bridge or while they were in the fuel preparation machines. Fuel clad failure and fission product release due to overheating can occur within one to two hours, depending upon decay heat generation rate.

ANALYSIS/DISCUSSION:

Frequency of Occurrence

A search of the LER and other data bases identified ten events since 1983 that have resulted in inadvertent draining of boiling water reactor vessels to the suppression pool. These events are broken down by year as follows:

Inadvertent Reactor Vessel Draindown Events

Year	Number of Events
1983	4
1984	2
1985	4
1986	0

Based on approximately 95 boiling water reactor years in the United States during this time period, these events result in an initiating event frequency of 0.09 per reactor year. This means that, with current reactor designs, procedure problems, and depth of operator training, each boiling water reactor (on average) could expect to experience one event of this type every 11 years.

Preventive Design Measures

One method to prevent most events of this type is to provide interlocks between the shutdown cooling suction and suppression pool suction valves, and between the shutdown cooling suction and the full-flow test return valves. The interlocks would prevent the

opening of either valve in the pair if the other is not fully closed.

Most boiling water reactors have a one-way interlock to prevent opening the shutdown cooling suction valve if the suppression pool suction valve is open. However, the interlock does not prevent the suppression pool suction valve from being opened if the shutdown cooling suction valve is open. This was demonstrated by several of the events described in this SOER. In addition, the majority of boiling water reactors have no interlocks to prevent the opening of the shutdown cooling suction valves and the full-flow test return valves at the same time. Such an interlock would have prevented the type of event that occurred at Peach Bottom. Some plants have already installed the interlocks described. Names of these plants are available from the information contact.

Bypassing Interlocks

Typical boiling water reactor technical specifications allow the isolation signal that closes the shutdown cooling inboard and outboard isolation valves on low reactor water level to be bypassed in the cold shutdown and refueling modes. This is because the containment isolation function of the valves is not needed in these modes. However, to ensure protection against inadvertent vessel draining, the low reactor water level isolation signal and the valve interlocks, such as those discussed under preventive design measures above, should remain in effect during all modes of operation. Controls should be established in residual heat removal system operating procedures to prevent bypassing the isolation signal and the valve interlocks.

If exceptions arise that make it necessary to bypass the isolation signal or the valve interlocks, contingency measures should be taken in the form of having other emergency core cooling system trains, such as one or more low pressure core spray trains, available to provide water to the vessel. Calculations should be performed to confirm that the rate of draining through the largest potential drain path would not exceed the rate of addition by the emergency core cooling system equipment. Special monitoring and pre-planned isolation actions should also be considered. Nuclear Safety Analysis Center Report 88 addresses other contingency measures that should be considered. In addition, senior plant management review and approval should be obtained before bypassing any isolation signals or interlocks that protect against vessel draining.

Procedure Improvements

Several vessel draining events have highlighted the need for explicit operating procedures to control valve lineups on the residual heat removal system. Caution statements should be inserted in these procedures immediately prior to steps where deviation could lead to the initiation of a draining event. Procedures should provide specific information regarding valve travel times. Cautions should explain why it is important to allow time for a particular

valve to go full stroke before operating another valve. Procedures should describe how the valve position indication lights can be used to verify that a valve has moved full stroke. Cautions should point out that both the closed and open lamps will be lit during midstroke.

The event at LaSalle 2 demonstrates what can happen when the shutdown cooling valves are operated from remote shutdown panels. These panels are typically designed to bypass various system interlocks. The procedures governing use of the remote shutdown panel should contain caution statements that indicate when pertinent interlocks are being bypassed and the potential effect. In addition, human factors measures, such as permanent caution labels on controls or the use of switch covers, should be employed to help prevent misoperation.

Training Enhancements

Licensed operators should receive training on events involving inadvertent draining of the reactor vessel to the suppression pool. The training should address the major causal factors, including human errors. The case study concept should be utilized on selected events. The training should also address plant procedures for performing operations associated with the shutdown cooling mode of operation and steps that could lead to opening a drain path from the vessel. The functioning of system interlocks and the risks of bypassing them in terms of maintaining reactor water level should also be addressed. The objective of the training should be to provide licensed operators with an understanding of reactor vessel draining events and how to prevent the occurrence of similar events.

Maintenance personnel should receive training on the impact that draining the vessel could have on the plant. Specific events in this SOER in which maintenance activities contributed to the event should be discussed along with how the events may have been avoided. The training should also address the need for following applicable maintenance procedures and the importance of communicating the scope of work and final system status when the work is completed. The objective of the training should be to provide maintenance personnel with an understanding of reactor vessel draining events that have occurred and how they affect plant operations.

Refueling Considerations

If a vessel draining event occurred during a refueling outage when irradiated fuel was being moved between the vessel and the fuel storage pool, it would be possible to uncover a fuel bundle being transported by the refueling bridge or in the raised position on the fuel preparation machines in the fuel storage pool. The overall effect would be the same as a large-volume failure of a reactor cavity seal with the exception that the water goes to a confined volume, i.e., the suppression pool, and is available for reflooding via residual heat removal or core spray pumps. Plants should again

review SOER 85-1 and SER 72-84 Supplements 1 and 2 to ensure that adequate measures have been prescribed to reposition bundles that may be potentially uncovered and to provide evacuation instructions to personnel in the primary containment and on the refueling floor.

The low reactor water level isolation signal that closes the shutdown cooling isolation valves does not prevent uncovering fuel in transport during a vessel draining event. The water level at which this isolation signal is actuated is well below the flange of the reactor vessel. By the time the water level reached that elevation, fuel in transport would already be uncovered. The valve interlocks discussed previously are the best means for protecting against draining during refueling operations. Additional measures in the form of procedural requirements that preclude changing from the shutdown cooling to the standby low pressure coolant injection valve lineup during refueling operations can also be taken.

RECOMMENDATIONS:

Design

1. Provide interlocks between the suppression pool suction valve and the shutdown cooling suction valve on each residual heat removal pump so neither valve can be opened if the other is not fully closed. Provide similar interlocks between the shutdown cooling suction valve on each pump and the full-flow test return valve. (Some BWRs have already installed these interlocks. None of the plants that have installed the interlocks have had vessel draining events. The information contact shown on this SOER can be contacted for further information.)

Procedures

2. Provide cautions and instructions at appropriate steps in residual heat removal system operating procedures relative to the mitigation of inadvertent reactor vessel draining. This should include procedures that pertain to the following operations:

   a. switching the residual heat removal system between shutdown cooling and low pressure coolant injection valve lineups

   b. transferring control of the residual heat removal system from the control room to the remote shutdown panel

3. Provide controls in residual heat removal system operating procedures that preclude bypassing the low reactor water level isolation signal and the valve interlocks. If special exceptions occur that require bypassing the isolation signal or the valve interlocks, the procedures should specify

special requirements to be followed. These should include the following:

a. senior plant management review and approval to bypass the interlocks

b. contingency methods to supply sufficient makeup water if a draining event occurs while the interlocks are bypassed

Training

4. Provide initial and continuing training to operations and maintenance personnel on industry reactor vessel draining events. This SOER, including detailed discussions of the events that are described, should be made an integral part of that training (the case study approach). The training should stress the role of human error in these events, potential consequences in terms of fuel damage, and the risks associated with bypassing interlocks.

INFORMATION CONTACT: Adam H. Geesey, INPO, 404/953-5318

MGC/2/IDO

ATTACHMENT 3

AECL PROPRIETARY
COMMERCIAL - NMPC

CRNL-2994

DRAFT

INSTRUCTION MANUAL

FOR THE NINE MILE POINT-1

REACTOR RECIRCULATION PUMP SEAL

by

R. Lesco and R. Metcalfe

FLUID SEALING TECHNOLOGY UNIT
Atomic Energy of Canada Limited
Chalk River Nuclear Laboratories
Chalk River, Ontario Canada K0J 1J0

1986 February

T_2 increases while T_1 remains essentially stable, high friction from the No. 2 stage is indicated.

Although temperatures are important signs of abnormality, they are superfluous to the "Failure Boundary" that should be applied when operating with CAN seals. This is given in Figure 7.2 for nominal operating pressures (1050 psig). Pump speed has no effect, but the depicted boundary must be adjusted in order to interpret seal condition at lower pressures.

Long before the CAN seal cartridge is "failed", one seal stage (or both) will be "deteriorated" such that if full system pressure (1050 psig) were applied across it (that is, the other stage failed completely), the combined outleakage would project to more than 1.5 US gpm upper limit (see Figure 7.2). This illustrates the potential loss of lifetime through premature replacement of "just deteriorated" seals.

In summary, Figure 7.2 should be used to monitor the trend of CAN seal cartridge condition, starting within the shaded region of "normal performance" and moving out towards the "failure boundary", as shown for example by the test data plotted in Figure 7.3. These seals are designed to give ample warning through a gradual approach to failure.

7.4 Abnormal Conditions

7.4.1 Hot Standby

When an RRP is on hot standby, there is no forced circulation through its heat exchangers. Hot water ingress to the seals then depends on the amount of leakage through the No. 1 stage. In the CAN seal, this has been designed to stabilize at less than 0.2 gpm in order to maintain clearance in the carbon bearing. The critical temperature calculated to correspond to this is about 300°F at the pump bearing.

CRITICAL TO BEARING

Based on the above, an RRP must not be restarted after hot standby (unless the reactor system has been cooled down) under either of the following conditions:

(i) the No. 1 stage seal leakage exceeds 0.2 gpm
(ii) the No. 1 stage seal cavity temperature exceeds 250°F

Note: The No. 1 seal leakage must be interpreted from measured interseal pressure and combined outleakage, as shown by L_1 (Point A) in Figure 7.2.

Either condition indicates No. 1 seal deterioration or failure and necessary replacement of the cartridge.

7.4.2 Loss of Cooling Water

An RRP must not be operated without cooling water. Staging flow (and any seal leakage) will draw hot reactor system water through the pump. Elastomers in the CAN seal are designed to resist failure by extrusion, but will never-the-