NRC FORM 195 (2-76)

U.S. NUCLEAR REGULATORY CO[illegible]

DOCKET NUMBER: 50-261

FILE NUMBER

NRC DISTRIBUTION FOR PART 50 DOCKET MATERIAL

TO: R W Reid

FROM: Carolina Pwr & Light Co
Raleigh, NC
E E Utley

DATE OF DOCUMENT: 3-3-77

DATE RECEIVED: 3-7-77

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DESCRIPTION

Ltr re our 1-10-77 ltr.....trans the following:

1p

REACTOR VESSEL OVERPRESSURIZATION
DISTRIBUTION PER G. ZECH 10-21-76

PLANT NAME: H B Robinson

ENCLOSURE

Info concerning Reactor Vessel Over-Pressurization....furnishing appropriate design modifications......

(40 cys encl rec'd)

½inch

ACKNOWLEDGED
DO NOT REMOVE

SAFETY — FOR ACTION/INFORMATION 3-8-77 ehf

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2394 261 MA4 OD

NRC FORM 195 (2-76)

Carolina Power & Light Company

March 3, 1977

FILE: NG-3514(R) SERIAL: NG-77-236

Director of Nuclear Reactor Regulation
ATTN: Robert W. Reid, Chief
Operating Reactors Branch 4
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

RECEIVED
MAR 7 1977
U.S. NUCLEAR REGULATORY COMMISSION
Mail Section

H. B. ROBINSON STEAM ELECTRIC PLANT UNIT NO. 2
DOCKET NO. 50-261
FACILITY OPERATING LICENSE NO. DPR-23
REACTOR VESSEL OVERPRESSURIZATION

Dear Mr. Reid:

Your letter of January 10, 1977, requested a commitment from Carolina Power & Light Company (CP&L) for a specific implementation plan to resolve the overpressurization issue and asked several plant specific questions. Our responses to the questions are provided in an attachment to this letter.

CP&L shares your concern about the overpressurization incidents which have occurred at other PWR facilities, and we plan to make acceptable long term hardware changes as soon as practical. Your insistance on a commitment at this time makes it impossible to commit to a long term fix since the basic analytical work has not yet been completed. However, CP&L will install modifications at our next refueling outage, which is currently scheduled for November of this year. This modification will utilize the pressurizer relief valves and will be at least as acceptable as the "Reference Mitigating System" described in our letter of December 16, 1976. If the analytical work for a better modification is completed and the necessary equipment is obtainable, then our level of protection will be appropriately increased. A submittal detailing our proposed design modifications will be submitted in sufficient time to permit your review and approval prior to installation.

Yours very truly,

E. E. Utley
Senior Vice President
Power Supply

RHC/CSB/kr

Attachment

2394

336 Fayetteville Street • P. O. Box 1551 • Raleigh, N. C. 27602

Responses to NRC Additional Information Request
of January 10, 1977

1. The staff considers it essential that all plant operators (i.e., reactor operators, equipment operators, Instrument & Control personnel) be made aware of the details of the pressure transients which have taken place at all PWR facilities. POSITION: Formal discussions should be held with the operator to review the causes of past pressure transients that have occurred at other operating PWR facilities. Your discussions should include the plant conditions at the time, the mitigating action that could have been or was taken, and the preventive measures that could have been taken to avoid the event and the steps taken to prevent similar, further occurrences. Plant similarities and distinctions should be identified along with how these relate to plant startup, shutdown, and testing operations. With regard to this position, you are requested to provide the following information:

a. If you have not already completed the required formal discussion, when will you do so?

ans. The discussion will be held as soon as the necessary details of the pressure transients can be obtained from the appropriate utilities and reviewed.

b. How will the discussions be held?

ans. Formal discussions will be held either on shift or in the class room as necessary to reach the appropriate plant personnel in a timely manner.

c. Of the past PWR Appendix G violations that have occurred at PWR facilities and which are described in License Event Reports, identify which are not credible in your plant due to equipment differences. Provide a description of the distinctions.

ans. 1. Preoperational tests that resulted in overpressurizations are no longer credible events since this unit completed required tests in 1970.

2. Automatic isolation of RHR system while solid. The Robinson facility does not have the automatic isolation feature imposed by the NRC on newer plants.

d. Describe, in detail, how you are reducing the likelihood of the other remaining credible events. Furnish schematics, diagrams or procedural summaries necessary to support the effectiveness and reliability of these measures.

ans. The remaining credible events that could occur at H. B. Robinson Unit No. 2 are listed below. The major preventative measure is administrative controls over the operation of equipment throughout the facility.

1. Manual isolation of the RHR system while solid conditions exist is procedurally prohibited. It is required that a steam bubble be established prior to isolation of RHR.

2. Overpressurization resulting from equipment tests are prevented through the use of special valve lineups. These lineups are a permanent part of the tests that are reviewed by Operations personnel and Management prior to approval. During the lineup, the operator is required to initial each valve after he confirms it to be properly aligned. This method allows verification of valve lineup completion prior to system test or operation.

3. Spurious operation of the safety injection system during cold shutdown is averted by procedurally requiring the SI accumulator isolation valves to be closed and deenergized, and the high head SI pumps to be deenergized. It should be noted that this equipment is operated for tests, filling of various systems, etc. during cold shutdown, but only with approved procedures and valve lineups.

4. The magnitude of pressure transients resulting from reactor coolant pump starts are minimized by requiring the initial pump start to be of short duration, (30 second "bump").

This "bump" allows a slower equalization of the stored energy in various system components, thus reducing the possibility of a transient. The procedure governing the inital starting of reactor coolant pumps is the fill and vent procedure.

5. Manual opening of the accumulator isolation valves at low pressure conditions is prevented by procedural restrictions. When the accumulators are pressurized, the isolation valves are opened only when the pressure of the primary system exceeds 1000 psig.

6. Personnel error or failure to properly follow established procedures will always be a credible event that cannot be easily avoided. H. B. Robinson has a continuous retraining program that stresses the proper use of our procedures. Our facility has a low incidence of operator error and no overpressurization events. This is a result of the constant emphasis on retraining and primarily on the amount of experience in plant operations. A large number of the overpressurization events have occurred at new facilities where the complete spectrums of operation have not yet been performed.

The above is a complete list of what we consider as credible events at H. B. Robinson Unit No. 2. Should you desire further information in the form of system schematics, we suggest that you consult your records under Docket No. 50-261 for the Robinson FSAR. It has sufficient information to fill your needs relating to system design.

2. The majority of the reported pressure transients events have occurred while the plants were operating in a water solid condition. POSITION: The staff will require that operations during which the plant is maintained in a water solid condition be minimized or if possible eliminated. Those operations in which the plant is in a water solid condition must be fully justified. Accordingly, please provide the following information:

a. Describe the procedures, evolutions or situations that require the plant be maintained in a water solid condition. Also provide reasons why a nitrogen, air or steam bubble cannot be maintained in these situations.

ans. The RCS is required to be solid when the Reactor Coolant Pumps (RCP's) are started and the reactor vessel head and pressurizer are to be vented. The RCP's are started in the following start, stop, vent sequence. An RCP is run for thirty seconds, then the reactor vessel head and pressurizer are vented. This short run will move most of the air out of the system and will assist in equalizing the loop temperatures. All three RCP's are run and vented in this manner. The RCP's are then sequenced through two minute runs followed by venting the reactor vessel head and pressurizer fifteen minutes after they are stopped. This will eliminate most of the entrapped air, since it will accumulate in the high spots. The RCP's are then sequenced through ten minute runs and are left running if the RCS pressure does not drop below 400 psig. If at anytime during the venting sequence, RCS pressure decreases to less than 400 psig, the RCP's are stopped and the RCS is again vented. The RCS must be solid during this process so air in the system can be readily identified and vented.

Nitrogen Bubble

If a nitrogen bubble was maintained in the pressurizer during pump startup and venting, it would be maintained at 400 psig to meet the startup requirements for startup or running an RCP. Pressure control would be unstable when the RCP's are started as the only control would be to add or vent nitrogen. At this pressure some nitrogen would be absorbed into solution. Radiation reverses the nitrogen, oxygen reaction forming some NO_3^- which would be detrimental to every surface in the RCS on a long term basis.

Air

The use of air at 400 psig in the pressurizers would raise the oxygen concentration of the RCS and we would be unable to maintain our

required 0.1 ppm oxygen limit with undesirable long term consequences.

Steam Bubble

If a steam bubble is formed in the pressurizer before starting an RCP, the surge line thermal sleeve would experience a thermal shock, the effects of which have not yet been determined. The surge line will be hot from the outsurge of hot water from the pressurizer when the steam bubble is formed. When the RCP is started the insurge of colder water from the loops would produce the surge line thermal sleeve thermal shock.

b. Include sufficient background or supplementary information such as system diagrams, procedure summaries and descriptions of equipment operation to justify your need for operating the plant in a water solid condition.

ans. The FSAR contains background and supplementary data discussing all facets of plant operation, system diagrams and descriptions of equipment operation.

3. The inadvertent operation of SIS components during cold shutdown conditions has been responsible for a major portion of the overpressure incidents. POSITION: Based on the licensee submittals, the recent November 3-5, 1976 meetings, and discussions with NSSS vendors, the staff will require the deenergizing of SIS pumps and closure of SI header/discharge valves during cold shutdown operations. Those situations during which this position cannot be met must be described and be fully justified. Accordingly, please provide the following information:

a. A schematic diagram of the SIS showing the flowpaths into the RCS.

ans. This schematic is attached.

b. The head-flow characteristics of each of the SIS pumps.

ans. This curve is attached.

c. Identify on the schematic diagrams the pumps and the valves to be closed and deenergized.

ans. These pumps and valves are identified on the attached schematic.

d. Your time schedule for implementing these administrative and operating procedural changes to meet this position.

ans. After investigation of the overpressurization problem during cold shutdown conditions, present procedures are considered adequate. Since the SIS pumps' fuses are removed and the breakers racked out during this condition, closure of the SIS header/discharge valves is not considered necessary. No further administrative or procedural changes are scheduled at this time.

e. Indicate all circumstances for which the SIS pumps and valves may not be isolated and deenergized and for those situations, describe the manner in which SIS injection would be prevented.

ans. During cold shutdown conditions, there are only three cases where SIS pumps are not deenergized and the accumulators not isolated. 1) during refueling internal periodic tests (PT) 2.1 and 23.2 volume 10 of the Plant Operating Manual. These periodic tests are required to meet Technical Specifications requirements. SIS injection into the RCS is prevented during this test by having the SIS pumps header/discharge valves closed and the accumulators depressurized; 2) operational testing after maintenance on the pump would require operation. There would be similar safeguards as above in this situation; 3) the SIS pumps are also used to flood the refueling canal for refueling but this operation cannot create an overpressurization situation since the vessel head has been unbolted and the system is no longer solid.

f. The location of the breakers that will be opened, and the places from which they can be controlled.

ans. The breakers for the SIS pumps are located at and controlled from the emergency buses E1 and E2. The breakers for the accumulator

isolation valves SI 868 A, B & C are located at and controlled from the motor control centers (MCC) 5 & 6. These locations are remote from the RTGB.

g. Describe the position indication and status signals which will be lost as a result of deenergizing these components.

ans. The position indication lights for the accumulator isolation valves and the status signal lights for the SIS pumps are lost locally and at the RTGB.

h. Describe in detail, the administrative procedures which will be used to assure proper equipment alignment and the supervisory personnel responsible for maintaining control.

ans. Under present procedures, protection from overpressurization of the RCS is considered adequate. In the Plant Operating Manual volume 4, GP-1, paragraph 4.17 and 4.18 require closure of the accumulator isolation valves when RCS pressure decreases to 1000 psig. GP-1D, paragraph 4.37.1 requires when in the cold shutdown condition, RCS temperature at or below 200°F; the accumulator isolation valve breakers are to be opened; the fuses removed and the breakers racked out on the SIS pumps. The shift foremen are responsible for insuring that these procedures are followed.

i. Indicate the RCS temperature/pressure conditions for which the accumulator isolation valve will be closed, and describe the location of the breaker, and the places from which it may be controlled.

ans. As described in item 3h the accumulator isolation valves will be closed when the RCS pressure is reduced to 1000 psig and the breakers opened when RCS temperature is reduced to 200°F. The breaker location and control is at the Motor Control Center (MCC) 5 & 6 which are remote from the RTGB.

j. Describe the impact on overall plant operations if you routinely lowered accumulator nitrogen pressure when in a cold shutdown condition.

ans. The impact to overall plant operations for lowering the accumulator nitrogen pressure during cold shutdown conditions is to increase the plant operating cost due to increased nitrogen consumption and an increase in the activity releases due to the higher containment pressure attained.

4. The staff has noted that several Appendix G violations have occurred during component or system tests while in cold and shutdown conditions. In this regard, please address the following questions.

a. What components or systems that could cause overpressure transients, are routinely tested while in a cold shutdown condition?

ans. During cold shutdown, the safety injection system and, in some cases, the charging system (CVCS) are tested on routine intervals. Both the high head safety injection pumps and the charging pumps are capable of producing pressure transients. The SI accumulators are the only other source of high pressure fluid available for initiation of an overpressure transient.

b. What extra measures are taken to prevent an overpressure event during these tests?

ans. Valve lineups prevent the overpressurization of the primary system during pump tests. These lineups are a standard part of any approved system test procedure. During tests that require the functioning of accumulator isolation valves, the pressure of the accumulators is lowered prior to testing. Each test has various precautions to make the operator aware of the potential dangers associated with the testing of each system. For further information, refer to our response to question 1d.

5. The staff believes that a high pressure alarm used during low RCS temperature operations is an effective means to attract the operator's attention to a transient in progress.

POSITION: The staff will require that if it is not presently installed, such an alarm must be installed as soon as possible. Accordingly, please furnish the following information:

a. Your method to provide the alarm, and the associated time schedule, or your justification for why this cannot be done.

ans. A method for providing a high pressure alarm during low temperature operations is described as follows:

The proposal is to utilize (1) the existing Reactor Coolant System (RCS) C loop wide range pressure transmitter to supply a comparitor and (2) to utilize the existing RCS C loop wide range temperature transmitter to supply a comparitor. These two comparitors activated in coincidence will provide an alarm at a spare location in the existing annunciator system. The comparitor setpoints will be adjusted to values consistent with low RCS temperature operations. A modification is now being developed and the alarm will be installed as soon as possible after completion of the design and the procurement of the necessary equipment.

b. A synopsis of the system modifications that are necessary.

ans. The system modification will require the following items:

1. Addition of a comparitor in the RCS C loop wide range pressure transmitter instrumentation loop.
2. Addition of a comparitor in the RCS C loop wide range temperature transmitter instrumentation loop.
3. Two auxiliary relays to be activated by the comparitors.
4. Cabling from the relays to the comparitors and the annunciator panel.

c. The alarm setpoint, mode of annunciation and sensor.

ans. At the present the pressure alarm setting is under investigation. Preliminary indications are that the alarm will be set to activate at 450-475 psig when the primary loop temperature is below 350°F.

d. How you ensure that the alarm is available and operating properly during all water-solid operations and how you minimize its downtime for all other cold shutdown conditions.

ans. The reliability of the alarm system will be ensured by utilizing existing plant calibration and testing procedures which are conducted on a periodic basis. This calibration and testing will be scheduled at times when the plant is not operating at water solid conditions.

6. The RHR is normally connected to the RCS and operating when the plant is in a cold shutdown condition. The inadvertent isolation of the RHR system while water solid has caused a number of overpressure transients, and the RHR safety valve has actually terminated others. The RHR (or SCS) therefore plays an important part in the initiation and possible mitigation of the PWR overpressurizations. Accordingly, we request the following additional information:

a. RHR design pressure.

ans. See attached table.

b. A description of the system isolation valves and their arrangement (e.g., number and configuration of valves installed, and pneumatic or motor operated).

ans. See attached schematic of RHR system with isolation and safety valves identified.

c. Interlocks, interlock setpoints, and alarms associated with each isolation valve.

ans. See attached table.

d. Nominal stroke time of isolation valves.

ans. The nominal stroke time of the isolation valves is approximately five minutes.

e. The setpoint and capacity of RHR (or SCS) relief and safety valves.

ans. See attached table.

f. All pressure alarms, setpoints and associated annunciation for the system.

ans. See attached table.

7. Reactor coolant system heatups, resulting from improper operation of the reactor coolant pump (RCP) while in a cold, shutdown and water solid condition, have been responsible for approximately 15% of the RCS overpressurization events.

POSITION: We will require that all licenses include adequate provisions to prevent RCP starts while in a water solid condition unless such starts are absolutely necessary. In those cases where the RCP starts cannot be avoided, the licensee should take appropriate steps to determine and minimize the RCS temperature profile.

Based on the position stated above, provide the following information:

a. Describe the normal operating conditions during which the RCS is maintained water solid with all RCP's stopped (e.g. fill and vent, pressurizer cooldown).

ans. During the heatup process, we begin the RCP start, stop and vent sequence as soon as the RCS is solid and has been pressurized to 400 psig.

During cooldown we place the RCS on RHR, then collapse the bubble and take the pressurizer solid and continue running RCP's with spray valves in the full open position to cooldown the pressurizer. When the pressurizer is cooled to less than 200°F, the RCP's are stopped and the RCS is depressurized to 50 psig. The cooldown of the RCS continues on RHR to less than 140° and the pressurizer is cooled down by charging through the auxiliary spray line rather than loop 2. When the RCS is less than 140°, the RCS is depressurized and the charging pumps are stopped.

The ability to operate solid allows cooling of the steam generators to near 100°F thus reducing the possibility of later overpressurization from a hot steam generator. Cooling to a low level cannot be achieved if a steam bubble is maintained to prevent going solid.

b. For each of the above procedures, justify your inability to establish a N_2, air or steam bubble in the pressurizer prior to the start of the first RCP.

ans. See response 2a.

c. What are the limits associated with system temperatures before the first RCP can be started in a solid RCS?

ans. There are presently no temperature profile limitations explicitly related to the starting of the first RCP in a solid RCS.

d. Specify the instruments utilized to determine the RCS temperature profile.

ans. Instrumentation presently provided in the RCS to evaluate a temperature profile include a wide range (0-700°F) reading RTD in each loop cold leg and hot leg. These instruments are provided with readout on two strip-chart recorders located on the RTGB. Additionally, reactor vessel fluid temperature can be obtained from

the incore thermocouples located above the core. These thermocouples are provided with readout in the control room on the moveable detector panel.

e. Provide the necessary schematics and procedural descriptions that show what your actions would be to bring the RCS to an isothermal condition.

ans. Reference: FSAR, Section 4, Reactor Coolant System, Figure 4.2-1.

The RHR system is normally in operation and circulating the RCS prior to the starting of a RCP in a solid condition. The referenced figure locates the RHR system interface with the RCS loops. Flow is directed from the "B" loop hot leg through the RHR system and returned to loop cold leg via the low head SIS connection. Normal operating procedures and startup practices dictate that this circulation be in progress long before the first pump is started.

In addition to the above flow path, a relatively small flow is maintained to the loops by the CVCS. A small letdown is maintained from the RHR system into the CVCS for RCS pressure control. This flow (generally greater than 50 gpm) is returned to the RCS through the RCP thermal barrier and through the normal charging penetration in the "A" loop hot leg. The flow through the RCP thermal barrier is in-service since it accompanies the seal water injection flow which is required for RCP operation.

Since the above system configurations are routinely maintained during normal solid water conditions prior to starting a RCP, no procedures are provided that explicitly address a thermal profile. The procedures requiring these configurations are sufficient to create an equilibrium situation which approaches an isothermal condition.

f. Summarize any other measures you take to reduce possible RCS pressure spikes during RCP starts, (e.g. open all letdown orifice isolation valves, stop makeup flow, etc.).

ans. See response to Item 1d.

NOTES:

1. ISOLATION VALVE TO BE CLOSED AND DE-ENERGIZED
2. SIS PUMPS TO BE DE-ENERGIZED

REFERENCES:

1. AUXILIARY PIPING SHOP & FIELD FABRICATION WESTINGHOUSE E. SPEC. G676262
2. VALVE REFERENCE GUIDE WESTINGHOUSE E. SPEC. G676398
3. MATERIAL SPEC. PIPE & FITTINGS WESTINGHOUSE E. SPEC. G569866
4. DEFINITION OF SYMBOLS WESTINGHOUSE E. SPEC. G675176
5. INSTRUMENTATION & CONTROL STANDARDS WESTINGHOUSE SYMBOLS & APPLICATIONS FOR INSTRUMENT DIAGRAMS, SECTION 1.1 ISSUED AUGUST 12, 1966 WESTINGHOUSE INSTRUMENT INSTALLATION, SECTION 3.0, ISSUED NOV. 16, 1966

REFERENCES DRAWINGS:

RCS - REACTOR COOLANT SYSTEM WESTINGHOUSE DWG. 685J433-5379-971
CVCS - CHEMICAL & VOLUME CONTROL SYSTEM
SHEET #1 WESTINGHOUSE DWG. 684 J 753-5379-685
SHEET #2 WESTINGHOUSE DWG. 684 J 925-5379-686
SHEET #3 WESTINGHOUSE DWG. 541 F 055-5379-684
CC - COMPONENT COOLING WESTINGHOUSE DWG. 684 J 731-5379-376 (AUXILIARY COOLANT SYSTEM)
RHR - RESIDUAL HEAT REMOVAL WESTINGHOUSE DWG. 541F058-5379-1484 (AUXILIARY COOLANT SYSTEM)
SFPC - SPENT FUEL PIT COOLING WESTINGHOUSE DWG. 541F059-5379-1485 (AUXILIARY COOLANT SYSTEM)
SS - SAMPLING SYSTEM WESTINGHOUSE DWG. 541F056-5379-353
WDS - WASTE DISPOSAL SYSTEM SHEET #1 WESTINGHOUSE DWG. 684 J 721-5379-920 SHEET #2 WESTINGHOUSE DWG. 684 J 917-5379-921
IVSWS - ISOLATION VALVE SEAL WATER SYSTEM EBASCO DWG. G-190262

LEGEND:

R.R. - VALVE WITH REACH ROD
C.S. - CONTAINMENT SUMP
L.C. - LOCKED CLOSED
L.O. - LOCKED OPEN
F.C. - FAIL CLOSED
V - LOCAL VENT
S - SAFETY INJECTION ACTUATION SIGNAL
P - HIGH-HIGH CONTAINMENT PRESSURE SIGNAL
P (NO) - ITEM NO. OF PIPE WITHIN SLEEVE AT CONTAINMENT BUILDING
D - LOCAL DRAIN
IVSWS - ISOLATION VALVE SEAL WATER SYSTEM

NOTES:

A. PIPING IS SCHEDULE 140
B. LOCATE CONNECTION WITHIN 6 INCHES OF CHECK VALVE
C. 58 NOZZLES PER HEADER ARRANGEMENT NOZZLE TYPE - SPRACO MODEL 1713 (RAMP BOTTOM)
D. ALL WESTINGHOUSE ITEM NUMBERS IN SIS ARE SHOWN WITHOUT PREFIX CPLSI
E. LOCATE TAPS 2' APART
H. LOCATE CONNECTION CLOSE TO VALVES 860A, 868B AND 868C
J. FLOW UNDER SEAT VALVES OPENS AT 900 PSIG
K. VALVES 750 AND 751 CANNOT BE OPENED UNLESS VALVES 862A AND 862B, 863A AND 863B ARE CLOSED. VALVES 862A AND 862B, 863A AND 863B CANNOT BE OPENED UNLESS PRESSURE IN RESIDUAL HEAT REMOVAL LOOP IS BELOW 250 PSIG.
L. VALVE OPENING DELAYED 5 MINUTES.
M. HEAT TRACE PIPE AT LEAST THREE FEET BEYOND ISOLATION VALVE.
N. VALVE PROVIDED WITH MODULATING CONTROL FOR POST ACCIDENT RECIRCULATION FLOW CONTROL.

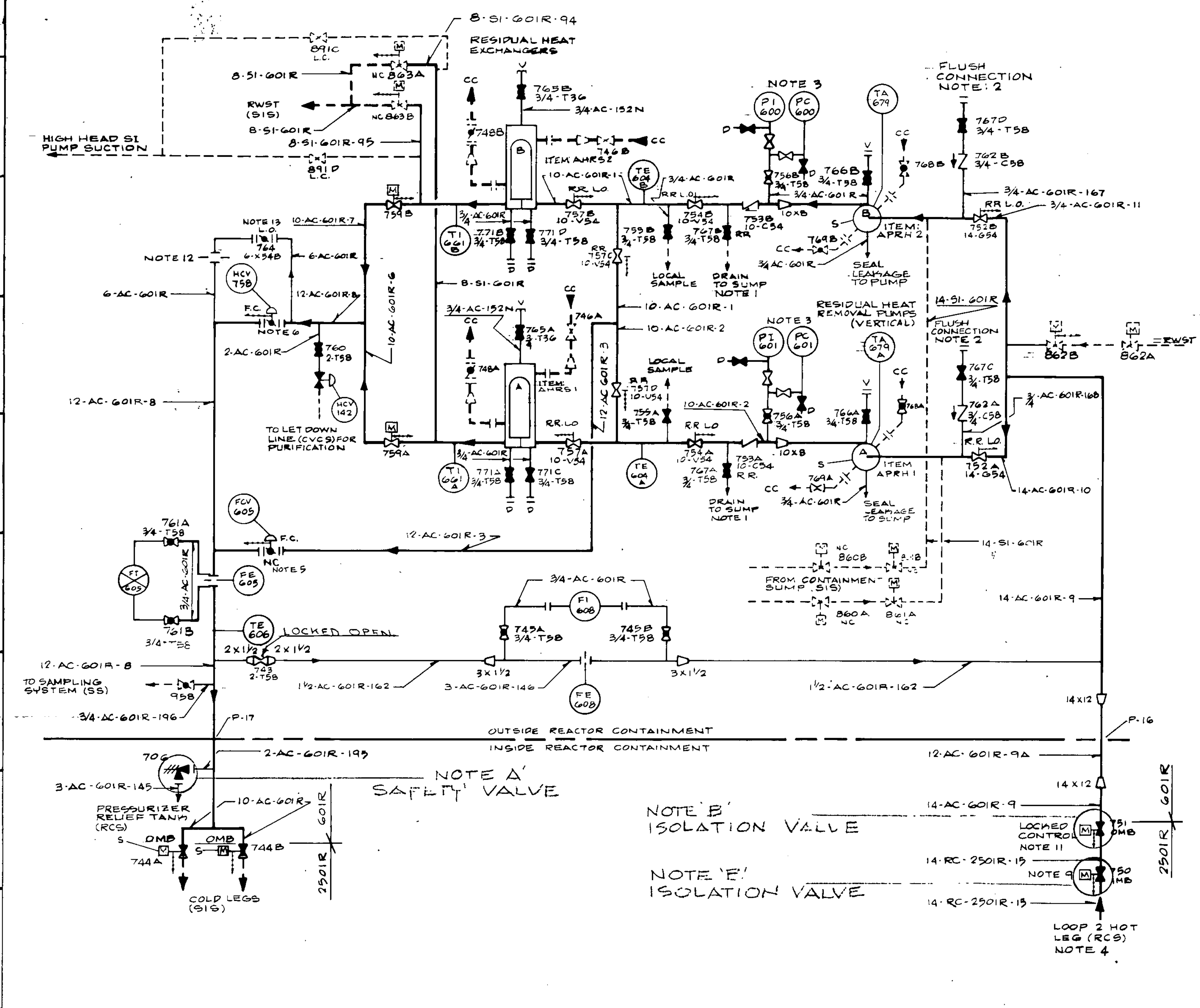
NOTE 1
ORIFICE PLATE AND SPACER ARE TO BE INSTALLED WITH 304 STAINLESS STEEL WIRE-WOUND ASBESTOS FEXITALIC GASKETS AND 304 SS NUTS AND BOLTS

NO	DESCRIPTION	BY	APPROVED	DATE
1	W-709878			
2	ECN-4285 NOTE N ADDED, NOTE F+G DELETED, NOTE L CHANGED, SPRAY ADDITIVE TK PIPING REVISED, REMOVED BIT ACCUMULATOR			3/13/69
3	CHANGED LI-950 TO LI-949A ECN-5105			10/7/69
4	REVISED PER FIELD CHG DCN-1-8 ECN-7262			1/28/71
5	INSTALL ORIFICE IN ACCUM RELIEF/INDICATING LINES MOD.#252			3/27/75
6	INSTALL FW FLUSHING OF BORIC ACID IN SAMPLING LINES MOD.#79			4/1/75
7	MODIFICATION NO. 252			
8	CORRECTED + VERIFIED TO "AS BUILT" CONDITIONS BY HBR-2 PLANT PERSONNEL		G.R. STOTT	6/1/76
9	CORRECTED PER TELEPHONE CONVERSATION: KEN DRIPPS/GR STOTT; REVERSED VALVE NOS. 870A & 870B (VAC. BKRS. TO 870C & D), SHOWED FLOW INDICATIONS ON 876A & B, & 883R	mtr	GR STOTT	12/1/76

CP&L ROBINSON STATION
UNIT NUMBER 2
SAFETY INJECTION SYSTEM
SHT 1 OF 1 ENGINEERING FLOW DIAGRAM
684J878 | CP-200-5379 | 1082

Response 3a&c

SIS FLOW DIAGRAM



REFERENCES:

1. AUXILIARY PIPING & FIELD FABRICATION
   Ⓦ E. SPEC. G676262
2. VALVE REFERENCE GUIDE
   W E. SPEC G676398
3. MATERIAL SPEC. PIPE & FITTINGS
   Ⓦ E. SPEC. G569866
4. DEFINITION OF SYMBOLS
   Ⓦ E. SPEC. G675176
5. INSTRUMENTATION & CONTROL STANDARDS
   Ⓦ SYMBOLS & APPLICATIONS FOR INSTRUMENT DIAGRAMS.
   SECTION 1.1., ISSUED AUGUST 12, 1966

REFERENCE DRAWINGS:

CVCS	CHEMICAL & VOLUME CONTROL SYSTEM		
	SHEET 1.	684 J 753	5379-685
	SHEET 2	684 J 925	5379-686
	SHEET 3	541 F 055	5379-684
RCS	REACTOR COOLANT SYSTEM	685 J 443	5379-1971
SS	SAMPLING SYSTEM	541 F 056	5379-353
SIS	SAFETY INJECTION SYSTEM	684 J 878	5379-1082
CC	COMPONENT COOLING (AUXILIARY COOLANT SYSTEM)	684 J 731	5379-376
SFPC	SPENT FUEL PIT COOLING (AUXILIARY COOLANT SYSTEM)	541 F 059	5379-1485
WDS	WASTE DISPOSAL SYSTEM		
	SHEET 1	684 J 721	5379-920
	SHEET 2	684 J 917	5379-921

LEGEND:

- RWST - REFUELING WATER STORAGE TANK
- V - LOCAL VENT
- D - LOCAL DRAIN
- F.C. - FAIL CLOSED
- F.O. - FAIL OPEN
- L.O. - LOCKED OPEN
- N.C. - NORMALLY CLOSED
- RR - VALVE WITH REACH ROD
- IMB - INSIDE MISSLE BARRIER
- OMB - OUTSIDE MISSILE BARRIER
- S - SAFETY INJECTION ACTUATION SIGNAL
- P(NO) - ITEM NO. OF PIPE WITHIN SLEEVE AT CONTAINMENT BUILDINGS

NOTES:

1. LOCATE IN LOW POINT IN LINE
2. LOCATE ABOVE RESIDUAL HEAT REMOVAL PUMP SHIELDING & CLOSE TO VALVES 752 A & B
3. INSTALL AT AN ACCESSIBLE AND CONVENIENT LOCATION
4. LOCATE SUCTION CONNECTION IN BOTTOM HALF OF REACTOR COOLANT PIPING ON 30° ANGLE TO VERTICAL.
5. VALVE IS NORMALLY CLOSED
6. VALVE IS NORMALLY CLOSED
7. VALVES ARE NORMALLY INSTALLED WITH FLOW UNDER SEAT EXCEPTIONS ARE 767 C & D
8. VENT AND DRAIN REQUIREMENTS DEPEND ON ON PIPING LAYOUT TO BE SPECIFIED BY A.E.
9. REACTOR COOLANT PRESSURE INTERLOCK PREVENTS ISOLATION VALVE OPENING ABOVE SET PRESSURE.
10. ALL Ⓦ ITEM NUMBERS ARE SHOWN WITHOUT PREFIX CPLAC
11. INTERLOCK PROVIDED SO THAT VALVES 750 & 751 CANNOT BE OPENED UNLESS VALVES 862A&B AND 863A&B IN THE SIS ARE CLOSED
12. FLANGES AND BLANK ORIFICE PLATE TO BE SUPPLIED BY EBASCO
13. VALVE NORMALLY OPEN DURING PLANT OPERATION AND CLOSED DURING SHUTDOWN. INSTALL IN CONVENIENT OPERATING LOCATION.

NO.	REVISIONS	DATE	DRW'D	APPR'D
6	SYSTEM REVIEWED/UPDATED BY PLANT PERSONNEL	10/20/76	M.R.	G.R.S.
5	SYSTEM UPDATED BY PLANT PERSONNEL	6-27-76	M. REEVE	G.R. STOTT K.E. BROM
4	ADDED TIE-IN TO RWST, CONTAINMENT SUMP AND HIGH HEAD SI PUMP SUCTION	1-30-76		FCL
3	CHANGE NOTE 6 ADDED NOTES 12 & 13			

NO.		REVISIONS		DATE	DRW'N	APPR'D
3		CHANGED HCV 758 TO FC. ADDED BY-PASS AROUND HCV 758 WITH FLG ORFICE & VALVE 764				
3	S_0	CPL 200 ECN 5105	UPDATED LINE NOS. ADDED MOTOR OPERATORS TO 759 A&B			
2		DARKENED IN NORMALLY CLOSED VALVES				
2	IT	SO CPL 200 ECN-3814	REVISED PIPING TO CONFORM TO PHYSICAL LAYOUT.			
1	SO D	CPL-200 709878				

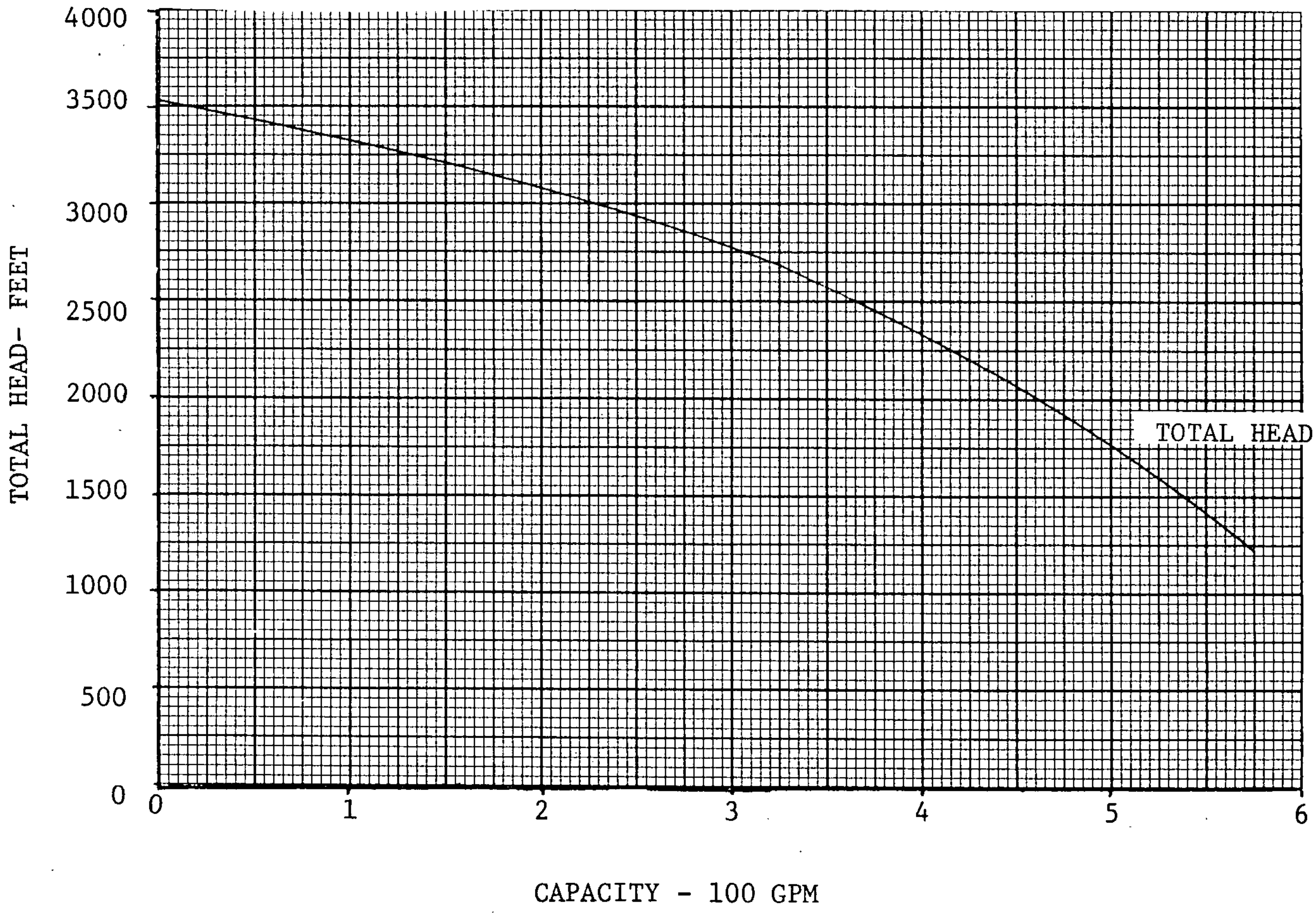
CP&L H.B. ROBINSON STATION
UNIT NUMBER 2
AUXILIARY COOLANT SYSTEM
RESIDUAL HEAT REMOVAL
ENGINEERING FLOW DIAGRAM

541F058 | CP-200-5379-1484

13 12 11 10 9 8 7 6 5 4 3 2 1

RESPONSE 3b

SAFETY INJECTION PUMP PERFORMANCE CHARACTERISTICS



Response 6b

RHR Systems

RESPONSE 6a

RESIDUAL HEAT REMOVAL SYSTEM
DESIGN, OPERATION AND TEST CONDITIONS

	Pumps	Heat Exchangers	Valves	Pipes and Fittings
Design Conditions				
Pressure, PSIG	600	600	665	700
Temperature, °F	400	400	400	400
Operating Conditions (Max)*				
Pressure, PSIG	160	160	160	160
Temperature, °F	180	180	180	180
Test Pressure, PSIG	1200	900	1100	900
Allowable Pressure at Operating Temp. PSIG	>600	>600	>690	>850

*During post loss-of-coolant recirculation

RESPONSE 6e

RELIEF VALVE SET POINTS

VALVE NO.	LOCATION	SETPOINT
*706 Safety Valve	Residual Heat Removal Loop	600 psig ± 18 psig
707	Component Cooling Surge Tank	100 psig ± 3 psig
715	Excess Letdown Heat Exchanger CCW Outlet	125 psig ± 3.5 psig
722A, B, & C	RCP Thermal Barrier CCW Outlet	2485 psig ± 24 psig
729	RCP Brng. Oil Cooler CCW Outlet	125 psig ± 3.5 psig
747A & B	RHRL Heat Exchanger CCW Outlet	150 psig ± 4.5 psig
791A	BA Evaporator A Condensate Cooler CCW Outlet	150 psig ± 4.5 psig
791B	BA Evaporator A Concentrator CCW Outlet	150 psig ± 4.5 psig
791C	Non-Regenerative Heat Exchanger CCW Outlet	150 psig ± 4.5 psig
791D	Waste Gas Compresser A CCW Outlet	150 psig ± 4.5 psig
791E	Waste Gas Compressor B CCW Outlet	150 psig ± 4.5 psig
791F	Waste Evaporator CCW Outlet	150 psig ± 4.5 psig
791G	Sampling System Heat Exchangers CCW Outlet	150 psig ± 4.5 psig
791H	Seal Water Heat Exchanger CCW Outlet	150 psig ± 4.5 psig
791J	BA Evaporator B Condensate Cooler CCW Outlet	150 psig ± 4.5 psig
791K	BA Evaporator B Concentrator CCW Outlet	150 psig ± 4.5 psig

*RHR safety valve capacity is 514 gpm of water at 10% overpressure.

RESPONSE 6f

PRESSURE ALARMS, SET POINTS & ASSOCIATED ANNUNCIATION

ALARM

RHR Pump A Cooling Water LOW Flow.

AUTO ACTION

None.

CAUSE

1. Loss of component cooling to pump.

OBSERVATIONS (local)

1. Component cooling flow to pumps. (FI 638)
2. Temperature of pump casing.

ACTION

1. Shift RHR Pumps.
2. If component cooling is lost the emergency cooling connections are available.

DEVICE/SET POINTS

1. FC-638/7 gpm

POSSIBLE PLANT EFFECTS

1. Loss of RHR system.

ALARM

RHR Pump B Cooling Water LOW Flow.

AUTO ACTION

None.

CAUSE

1. Loss of component cooling to pump.

OBSERVATIONS

1. Component cooling to pump (FI 637).
2. Temperature of pump casing.

ACTION

1. Shift RHR pumps.
2. If component cooling is lost the emergency cooling connections are available.

DEVICE/SET POINTS

1. FC-637/7 gpm.

POSSIBLE PLANT EFFECTS

1. Loss of RHR system.

ALARM

RHR Pump A Hi Pressure.

AUTO ACTION

None.

CAUSE

1. Increase of Reactor Coolant Pressure.
2. Malfunction of low pressure letdown valve if on solid pressure control.
3. Thermal expansion if isolated.
4. Improper valve line up.

OBSERVATIONS

1. Reactor coolant pressure (PI 403)
2. RHR loop pressure (Local - PI 600)
3. Valve status lights on RTGB

ACTION

1. Reduce charging speed to reduce charging flow if solid
2. Spray down pressurizer if bubble is drawn and reactor coolant pump running
3. Use HCV-142 to reduce system pressure

DEVICE/SET POINTS

1. PC 600/560 psig

POSSIBLE PLANT EFFECTS

1. Loss of use of RHR loop for decay heat removal

ALARM

RHR Pump B Hi Pressure

AUTO ACTION

None

CAUSE

1. Increase of Reactor Coolant Pressure
2. Malfunction of low pressure letdown valve if on solid pressure control
3. Thermal expansion if isolated
4. Improper valve line up

OBSERVATIONS

1. Reactor coolant pressure (PI 403)
2. RHR loop pressure (local PI 601)
3. Valve status lights on RTGB

ACTION

1. Reduce charging speed to reduce charging flow if solid
2. Spray down pressurizer if bubble is drawn and Reactor Coolant pump running
3. Use HCV-142 to decrease system pressure

DEVICE/SET POINTS

1. PC-601/560 psig

POSSIBLE PLANT EFFECTS

1. Loss of use of RHR loop for decay heat removal

ALARM

RHR Pumps Motor Overload/Trip.

AUTO ACTION

1. RHR Pump stops.

CAUSE

1. RHR pump breaker has opened on electrical fault.

OBSERVATIONS

1. RHR loop flow (FI 605)
2. Pump status lights on RTGB

ACTION

1. Start idle pump.

DEVICE/SET POINTS

1. NA

POSSIBLE PLANT EFFECTS

1. Loss of use of RHR loop for decay heat removal.