ENCLOSURE

U.S. NUCLEAR REGULATORY COMMISSION
REGION IV

Docket No.:	50-397
License No.:	NPF-21
Report No.:	50-397/98-301
Licensee:	Washington Public Power Supply System
Facility:	Washington Nuclear Project-2
Location:	Richland, Washington
Dates:	November 2-6, 1998
Inspector(s):	T. O. McKernon, Chief Examiner, Operations Branch R. E. Lantz, Examiner, Operations Branch M. E. Murphy, Senior Examiner, Operations Branch T. R. Meadows, Senior Examiner, Operations Branch
Approved By:	John L. Pellet, Chief, Operations Branch Division of Reactor Safety

ATTACHMENTS:

Attachment 1:	Supplemental Information
Attachment 2:	Simulator Facility Report
Attachment 3:	Facility Initial License Written Examination Comments and Analysis
Attachment 4:	Final Written Examinations, Answer Keys, and Proctor's Comments

EXECUTIVE SUMMARY

Washington Nuclear Project-2
NRC Inspection Report 50-397/98-301

NRC examiners evaluated the competency of 4 reactor operator and 8 senior operator applicants for issuance of operating licenses at the Washington Nuclear Project-2 facility. The licensee developed the initial license examinations using NUREG-1021, "Operator Licensing Examination Standards for Power Reactors," Interim Revision 8. NRC examiners reviewed, approved, and administered the examinations. The initial written examinations were administered to all 12 applicants on October 30, 1998, by facility proctors in accordance with instructions provided by the chief examiner. The NRC examiners administered the operating tests on November 2-5, 1998.

Operations

- Good operator performance and communication practices were observed during the initial operator licensing examination (Section O4.2).
- All applicants for reactor operator and senior operator licenses displayed the requisite knowledge and skills to satisfy the requirements of 10 CFR Part 55 and were issued the appropriate licenses (Sections O4.1, O4.2).
- The licensee initially failed to submit an acceptable examination for administration to operator license applicants for the operating test portion of the examinations. The final as-given examination met the requirements of NUREG-1021 and was considered good quality (Section O5.1.2).

Report Details

Summary of Plant Status

The plant operated at essentially 100 percent for the duration of this inspection.

I. Operations

O4 Operator Knowledge and Performance

O4.1 Initial Written Examination

a. Inspection Scope

On October 30, 1998, the licensee proctored the administration of the written examination approved by the NRC to four individuals who had applied for initial reactor operator licenses and eight individuals who had applied for senior operator licenses. The licensee graded the written examinations and its staff reviewed the results. The licensee also performed a post-examination question analysis, which was reviewed by the examiners.

b. Observations and Findings

The minimum passing score was 80 percent. All applicants for a reactor operator and senior operator license passed the written examination. Scores for the reactor operator applicants ranged from 86.9 to 87.9 percent. All senior operator license applicants passed with scores ranging from 90.9 to 96.9 percent. The average score for reactor operator applicants was 87.6 percent and the average score for senior operator applicants was 93.4 percent.

The above grades reflected the results after examination changes recommended by the licensee as a result of post-examination question analysis were incorporated. The examiners reviewed and accepted these recommendations based on the technical merits of each recommendation. As a result of this analysis, two choices were accepted for one question (RO 30, SRO 30) because the wording in choice "b" was also a true statement for the conditions given in the question stem. Also, Questions RO 9 and SRO 11 were deleted because no correct choice existed. The submitted comments and analysis are included as part of Attachment 3 to this report.

c. Conclusions

All reactor operator and senior operator license applicants passed the written examination.

O4.2 Initial Operating Test

a. Inspection Scope

The examiners administered the various portions of the operating test to the 12 applicants on November 2-5, 1998. Each applicant participated in 2 or 3 dynamic simulator scenarios. Each applicant also received a walkthrough test, which consisted of either 10 or 5 system tasks, depending on application type, with 2 followup questions for each task. The applicants also received an operating test administrative portion consisting of tasks or questions related to 5 subjects in 4 administrative areas.

b. Observations and Findings

All applicants passed all sections of the operating test. The applicants generally performed well and used good communication practices and peer checks. The applicants exhibited good plant knowledge and ability to find components when asked. Some applicants exhibited slow control board awareness during the dynamic scenarios. For example, some applicants were slow to recognize that a control rod drive pump had tripped off after a loss of power to the applicable electrical bus. While the slowness in recognizing the loss of the component did not exacerbate plant conditions, it did exhibit weakness on the part of some board operators in control board awareness. The examiners observed good ownership, and application of principles for self verification and peer checks by the applicants throughout the examination.

c. Conclusions

All applicants passed all sections of the operating test. Overall, the license applicants demonstrated good performance and use of good communication practices and peer checks.

O5 Operator Training and Qualification

O5.1 Initial Licensing Examination Development

The licensee developed the initial licensing examination in accordance with guidance provided in NUREG-1021.

O5.1.1 Examination Outline

a. Inspection Scope

The licensee submitted the initial examination outline on July 1, 1998. The examiners reviewed the submittal against the requirements of NUREG-1021.

b. Observations

The chief examiner provided enhancement suggestions related to examination integrity and responsiveness to NUREG-1021 requirements, which were incorporated by the licensee into the job performance measure and dynamic scenarios outlines. The submitted scenario outlines contained malfunctions associated with instrument malfunctions which lacked significant operator response to correct. Additional enhancement suggestions were made to the scenarios in order to add balance to the malfunctions so that one particular applicant did not receive all the malfunctions.

c. Conclusions

The licensee submitted final revised examination outlines prior to the final version of the examination submittal. The revised outlines satisfied the requirements of NUREG-1021.

O5.1.2 Examination Package

a. Inspection Scope

The licensee submitted the initial examination package on September 1, 1998. Because of extensive NRC comments on the initial operating test submittal, the licensee submitted a revised operating test package on October 29, 1998, following an onsite review by the examiners during the week of October 19, 1998. The chief examiner reviewed the submittal against the requirements of NUREG-1021.

b. Observations and Findings

The licensee submitted 130 draft written examination questions, of which 70 were designated as common questions to both the reactor operator and senior operator examinations. The chief examiner provided comments or questions on 12 of the questions. Additionally, the licensee performed internal audits, which identified other questions for revision. The majority of changes made to the questions were changes to question stems and answers for clarification. Although failure to make the above changes would not have invalidated the written examinations, it would have degraded their discriminatory value. The written examinations were adequate for administration.

The licensee submitted two sets of the administrative test, one intended for the reactor operator applicants and one for the initial senior operator initial and upgrade senior operator applicants. The reactor operator and the senior operator administrative tests were similar but the senior operators test differed in the questions asked and in the emergency preparedness areas. The examiners' review indicated that the administrative portion of the examination was unacceptable for administration because a number of the questions did not discriminate at the appropriate level.

Similarly, the system walkthrough portion of the examination was inadequate for administration. The majority of the job performance measure followup questions could be answered by direct look up and did not discriminate appropriately for open reference questions.

The examiners reviewed the scenarios and found them of good quality: Some enhancement suggestions were made for balancing malfunctions between applicants.

As a result of the above comments, the licensee resubmitted the operating portion of the examination on October 29, 1998. The revised examination materials satisfied NUREG-1021 requirements and were of good quality.

c. Conclustions

Although the written examination was acceptable, the licensee initially failed to submit an acceptable examination for administration to the operator license applicants for the operating portion of the examination. The final as-given examination met the requirements of NUREG-1021 and was considered good quality.

O5.1.3 Licensing Conditions

a. Inspection Scope

The chief examiner reviewed the final applications as submitted by the facility for the license applicants against the requirements of NUREG-1021, Interim Revision 8.

b. Observations and Findings

The chief examiner verified that the facility licensee properly identified the required five significant reactivity manipulations on the applications. The chief examiner also verified that the facility had properly documented these manipulations and that they were significant in accordance with NRC Information Notice 97-67.

c. Conclusions

The facility's program was adequate to ensure that initial license applicants satisfied the requirements for performance of significant reactivity manipulations

O5.2 Simulation Facility Performance

a. Inspection Scope

The examiners observed simulator performance with regard to fidelity during examination validation and administration..

b. Observations and Findings

The simulation facility supported examination administration well. However, minor unexpected simulator malfunctions and annunciations occurred during the dynamic scenarios. Examples included: (1) Annunciation of a high reactor vessel level alert alarm; (2) failure of the rod worth minimizer screen to update and insert control rod withdrawal block alarm annunciations; and (3) failure of the Division 3 diesel to tie-in to its safety bus. The simulator errors did not create a modeling problem with respect to mass or energy transfer and did not adversely affect the planned scenarios. The simulator errors are further discussed in attachment 2.

c. Conclusions

The simulator supported examination administration well with few minor exceptions.

V. Management Meetings

X1 Exit Meeting Summary

The inspectors presented the inspection results to members of the licensee management at the conclusion of the inspection on November 5, 1998. The licensee acknowledged the findings presented.

The licensee did not identify as proprietary any information or materials examined during this inspection.

ATTACHMENT 1

PARTIAL LIST OF PERSONS CONTACTED

Licensee

P. Taylor, Operations Training Manager
R. Guthrie, Operations Training
W. Oxenford, Operations Manager
J. McDonald, Engineering, General Manager
D. Coleman, Regulatory Affairs Manager
G. Smith, Plant General Manager
C. Golightly, Simulator Group
M. Westergren, Operations Training

NRC

J. Arildsen, Operator Licensing Branch, Office of Nuclear Reactor Regulation
S. Boynton, Senior Resident Inspector

ATTACHMENT 2

SIMULATION FACILITY REPORT

Facility Licensee: Washington Nuclear Project-2

Facility Docket: 50-397

Operating Examinations Administered at: Richland, WA.

Operating Examinations Administered on: November 2-6, 1998

These observations do not constitute audit or inspection findings and are not, without further verification and review, indicative of noncompliance with 10 CFR 55.45(b). These observations do not affect NRC certification or approval of the simulation facility, other than to provide information, which may be used in future evaluations. No licensee action is required in response to these observations.

1. During the performance of Scenario 5, Event 5, in which an earthquake caused a loss of coolant accident with a loss of Electrical Bus TR-S, the Division III HPCS diesel failed to tie onto the SM-4 bus. This was an unplanned and unexplained simulator error.

2. During the same scenario described above, a RPV high level alert alarm unexpectedly annunciated.

3. In Scenario 1, during control rod withdrawal to critical the operators received an unexpected and repeated rod out block alarm and the rod worth minimizer display would not update data.

ATTACHMENT 3

Facility Initial Written Examination Comments and Analysis

Written Exam Performance Analysis

exam given at WNP-2 on Oct. 30, 1998

Our exam analysis revealed 11 questions for which greater than 25% of the students gave an incorrect response.

Of these 11, only 2 questions were determined to require modification that would alter the final grade. These were:
1) question - SRO 11/RO 9/ex98010 and
2) question - SRO 30/RO30/ex98036.

For the remaining 9 questions. No changes were made to these questions. However, training will be provided to those students that missed these questions to upgrade their knowledge in the appropriate areas.

Question No.	
SRO 11 RO 9 ex98010	**Problem identified:** All applicants missed this question. 9 of 12 answered A, 3 of 12 answered D. **Recommendation:** Recommend deleting this question. **Justification:** The answer to this question was based on a previous revision to PPM 5.0.10. The current revision (rev 5), no longer defines "MSIV operation" as the basis. **Reference:** PPM 5.0.10 rev 5, pg 76, 77, 93. Exceeding 241°F WW at 600 psig in the RPV exceeds HCTL, which exceeds (potentially) PCPL.

The plant is in an ATWS. Suppression Pool level is normal, Suppression pool temperature is 241°F, and reactor pressure is 600 psig.

Which ONE of the following is correct concerning these conditions?

A. A LOCA may cause a Wetwell/Drywell interface failure.

B. A reactor depressurization may cause the MSIVs to become inoperable.

C. Spraying the Wetwell may cause a containment failure due to low internal pressure.

D. An emergency depressurization may cause SRV Tailpipe damage.

ANSWER: B

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295013AK3.02 3.6/3.8

REFERENCE: PPM 5.0.10, rev 4, page 101

SOURCE: NEW QUESTION SRO T1, G1, #10 RO T1, G2, #9

LO: 8303

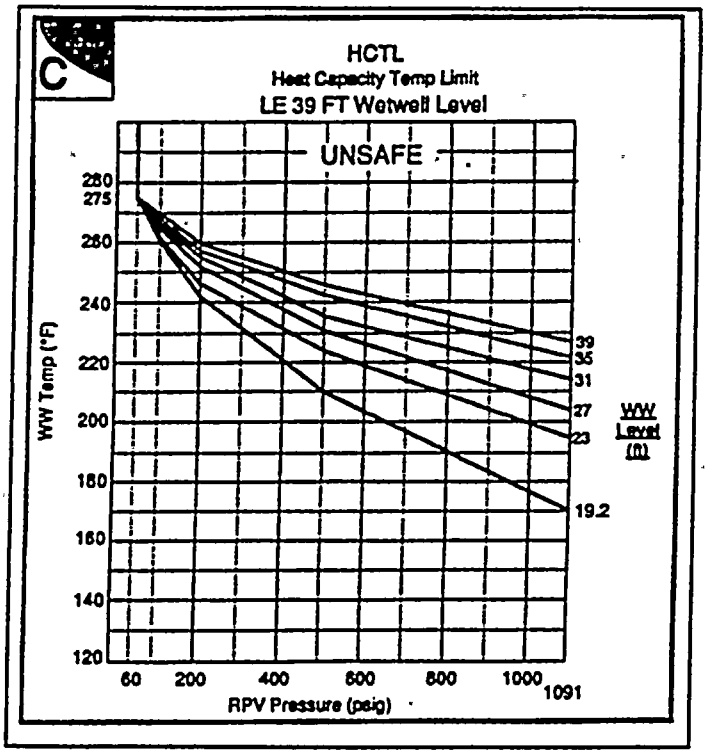
RATING: L2

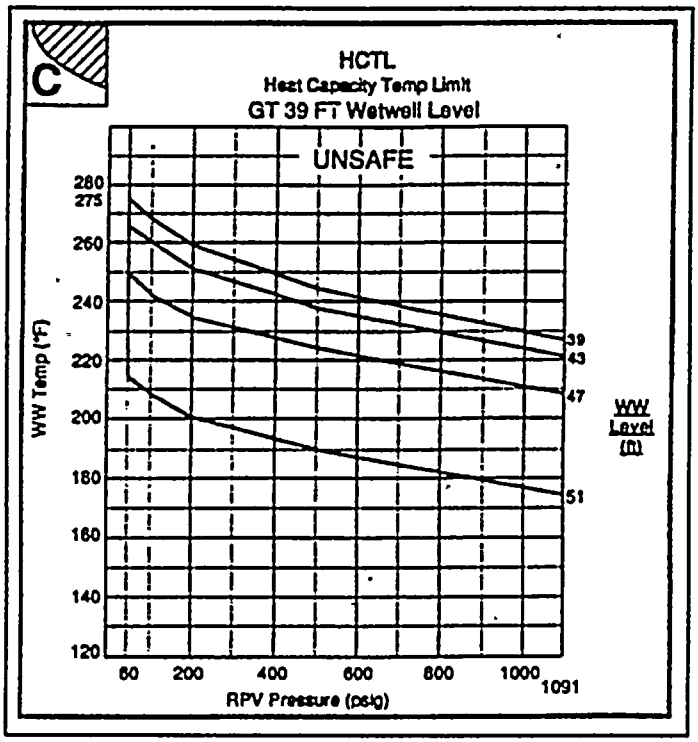
ATTACHMENT: PPM 5.2.1

JUSTIFICATION: A is incorrect because a LOCA would have no effect on the drywell floor/seal. C is incorrect because spraying the wetwell airspace would have no effect under these conditions. D is incorrect because there would be no damage with wetwell level in the normal range during an ED.

COMMENTS:

7.3 Heat Capacity Temperature Limit





The Heat Capacity Temperature Limit (HCTL) is the highest wetwell temperature at which initiation of RPV depressurization will not result in exceeding the PCPL before the rate of energy transfer to the containment is within the capacity of the containment vent. (Refer to discussion of the Primary Containment Pressure Limit.)

The HCTL is used to preclude failure of the containment or equipment necessary for safe shutdown by assuring that RPV blowdown does not cause containment pressure to exceed the PCPL.

For RPV pressures below 60 psig, the rate of energy transfer to the containment with 5 SRVs open is within the capacity of the containment vent. Therefore when the energy discharged from the RPV to the wetwell is equal to or less than the energy discharged outside the primary containment through the open vent, primary containment pressure will not exceed the PCPL. Five open SRVs is the Minimum Number of SRVs Required for Emergency Depressurization. (Refer to the discussion of the Minimum Number of SRVs Required for Emergency Depressurization in section 7.8) 60 psig is the Minimum RPV Flooding Pressure. (Refer to the discussion of the Minimum RPV Flooding Pressure in Section 7.9)

For RPV pressures above 60 psig, wetwell heatup during RPV depressurization is proportional to RPV pressure. Thus this segment of the HCTL decreases with increasing RPV pressure. If depressurization is performed with more than 5 open SRVs, the integrated energy addition to the wetwell is less than that resulting from fewer SRVs open because the rate of depressurization of the RPV is faster.

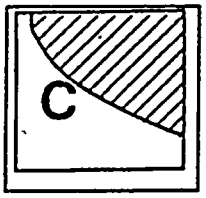
The HCTL also assumes all heat removal capability from the containment is lost, and the airspace and water in the wetwell are in thermal equilibrium.

Operation at pressure above the minimum pressure in the RPV at which an SRV is set to lift (1091 psig) is not expected to occur, therefore, the HCTL is not defined above this pressure.

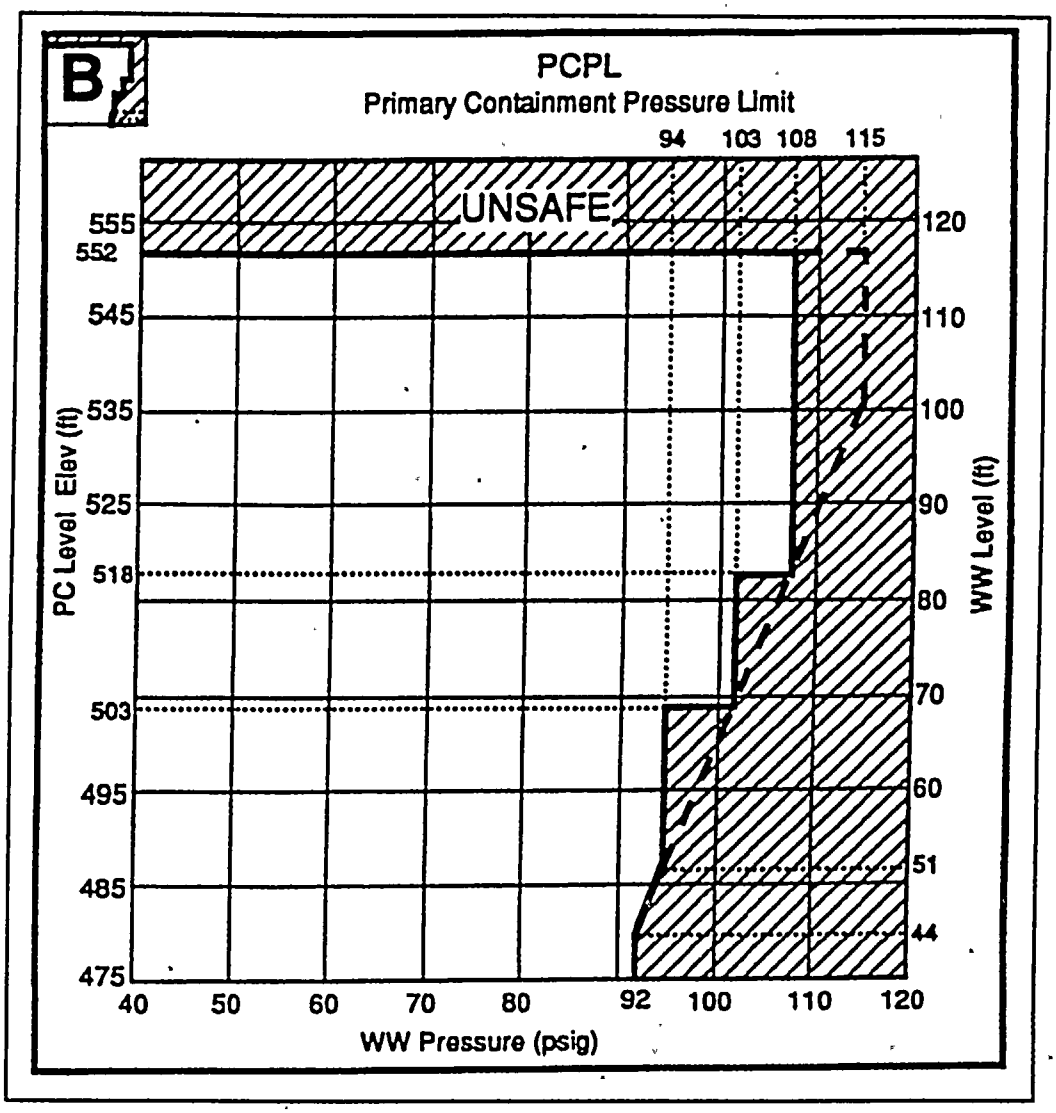
An override in the RPV pressure control flowpath of PPM 5.1.1 and PPM 5.1.2 direct the operator to control RPV pressure below HCTL. The wetwell temperature control flowpath of the PPM 5.2.1 requires emergency RPV depressurization when RPV pressure and wetwell temperature can not be maintained below HCTL.

Two graphs are given, LE 39 ft. and GT 39 ft.. The unsafe area of the HCTL curve is above and to the right of the level curve.

The HCTL is referenced in PPM 5.1.1, PPM 5.1.2, and PPM 5.2.1 with the following identifier:



7.13 Primary Containment Pressure Limit



The Primary Containment Pressure Limit (PCPL) is defined as limiting component in the primary containment. For conditions in the primary containment where containment level is LT 535 ft. the limit is based on SRV operation. Above 535 ft. the limit is base on the girder joint of the primary containment.The Primary Containment Pressure Limit (PCPL) is used to preclude failure of the containment and resultant loss of systems required to maintain adequate core cooling.

For wetwell water levels below 44 ft (479 ft elevation), the wetwell pressure instrument tap is not submerged and directly senses the airspace pressure. Therefore, the PCPL is a constant value (92 psig) defined by a vertical line between 0 ft and 44 ft.

Question No.	
SRO 30 RO30 ex98036	**Problem identified:** 9 of 12 applicants missed this question; 6 SROs, 3 ROs. All 9 answered B. **Recommendation:** Accept either answer B or D. **Justification:** Answer B would be correct if temperature got high enough to receive the "HI-HI" alarms. However, the students are not required to memorize the input logic to these alarms, and would expect an immediate isolation of RWCU-V-1 and 4 **Reference:** PPM 4.601.A3 pg 27, 28 PPM 4.601.A2 pg 33, 34 PPM 4.601.A12 pg 28 PPM 4.601.A2 pg 15 PPM 4.601.A3 pg 20

The plant is operating at 99% power, when the following alarms are received:

LEAK DET RWCU CH A DIFF FLOW HIGH/FLOW HIGH (59.5 GPM)
LEAK DET RWCU/RCIC PIPE AREA TEMP HIGH

Which ONE of the following describes the expected plant response?

In addition to the operating RWCU pump trip, ...

A. RWCU-V-4 closes immediately.

B. RWCU-V-4 and V-1 close immediately.

C. after a 45 second time delay, RWCU-V-4 only closes.

D. after a 45 second time delay, RWCU-V-4 and V-1 close.

ANSWER: D

QUESTION TYPE: SRO/RO

KA # & KA VALUE: 295032EA1.05 3.7/3.9

REFERENCE: PPM 4.601.A3 drop 3-4, rev 11 PPM 4.601.A12 drop 6-2, rev 16

SOURCE: NEW QUESTION SRO T1, G2, #10 RO T1, G3, #3

LO: 5035

RATING: H3

ATTACHMENT: NONE

JUSTIFICATION: D is correct because both RWCU-V-1 and 4 close after a 45 second time delay.

COMMENTS:

3-4 LEAK DET RWCU DIFFERENTIAL FLOW HI/CH A

3-4 WINDOW	SOURCE		AUTOMATIC ACTIONS
LEAK DET RWCU CH A DIFF FLOW HI/ FLOW HI	LD-FS-605A	(58.5 GPM Δflow and 45 second time delay) (LD-RLY-K7A)	• RWCU-V-4 Closes
	LD-FS-15	(253.5 gpm blowdown flow and 1.6 second time delay) (MS-RLY-K27)	

NOTE: Delta flow indication is from LD-FI-620 on H13-P602. Total blowdown flow is indicated on RWCU-FI-602 at H13-P602.

1. If differential flow is GE 58.5 gpm and automatic isolation is imminent, perform all of the following: {P-60468}

   a. Consider throttling open RWCU-V-104, Cleanup System Bypass, to preclude lifting RWCU HXR relief valves.

   b. Stop RWCU-P-1A(1B).

   c. Close RWCU-V-1.

   d. Close RWCU-V-4.

   e. If necessary, close RWCU-V-40 and secure RWCU CRD purge.

   f. Ensure RWCU-FCV-33 is closed.

2. If isolation has occurred:

   a. Ensure RWCU-V-4 closed.

   b. Consider throttling open RWCU-V-104, Cleanup System Bypass, to preclude lifting RWCU HXR relief valves.

   c. Secure RWCU pump CRD purge if RWCU-V-1 or V-4 and RWCU-V-40 are closed.

   d. Ensure RWCU-FCV-33 closed, locally if possible (RB 501, SW).

   e. Ensure RWCU Pump tripped.

(CONTINUED ON NEXT PAGE)

3. If system restart is not anticipated in the near future, consider closing RWCU-V-34 or RWCU-V-35 to stop any leakage through RWCU-FCV-33.

4. Monitor RPV level, Reactor Building sumps, RWCU piping and equipment area temperatures, and Reactor Building ARMs for indication of RWCU leak.

5. Monitor Reactor water conductivity at RWCU-CR-601 per PPM 1.13.1, Chemical Process Management and Control.

6. If instrument operability is in doubt, refer to Technical Specification 3.3.6.1 in Modes 1, 2, and 3.

7. If applicable, refer to PPM 4.12.4.1A, High Energy Line Break.

8. If applicable, refer to PPM 4.11.2.1, Liquid Radioactive Spills.

REFERENCES: CVI 02E31-05,7,9 (GE 807E171TC) (Leak Det.)
EWD-4E-0017, EWD-4E-0023
EWD-19E-0006, EWD-19E-0028, EWD-19E-0038
LER 92-039-00-05

{P-60468}

4-2 LEAK DET RWCU DIFF FLOW HI/CH B

4-2 WINDOW	SOURCE		AUTOMATIC ACTIONS
LEAK DET RWCU CH B DIFF FLOW HI/ FLOW HI	LD-FS-605B	(58.5 gpm Δflow and 45 second time delay) (LD-RLY-K7B)	RWCU-V-1 Closes
	LD-FS-16	(253.5 gpm blowdown flow and 1.6 second time delay) (MS-RLY-K26)	

NOTE: Delta flow indication is from LD-FI-620 on H13-P602. Total blowdown flow is indicated on RWCU-FI-16 at H13-P613.

1. If differential flow is GE 58.5 gpm and automatic isolation is imminent, perform all of the following: {P-60468}

   a. Consider Throttling open RWCU-V-104, Cleanup System Bypass, to preclude lifting RWCU HXR relief valves.

   b. Stop RWCU-P-1A(1B).

   c. Close RWCU-V-1.

   d. Close RWCU-V-4.

   e. If necessary, close RWCU-V-40 and secure RWCU CRD purge.

   f. Ensure RWCU-FCV-33 is closed.

2. If isolation has occurred:

   a. Ensure RWCU-V-1 closed.

   b. Consider throttling open RWCU-V-104, Cleanup System Bypass, to preclude lifting RWCU HXR relief valves.

   c. Secure RWCU pump CRD purge if RWCU-V-1 or 4 and RWCU-V-40 is closed.

   d. Ensure RWCU-FCV-33 closed, locally if possible (RB 501, SW).

   e. Ensure RWCU pump tripped.

(CONTINUED ON NEXT PAGE)

3. If system restart is not anticipated in the near future, consider closing RWCU-V-34 or RWCU-V-35 to stop any leakage through RWCU-FCV-33.

4. Monitor RPV level, Reactor Building sumps, RWCU piping and equipment area temperatures, and Reactor Building ARMs for indication of RWCU leak.

5. Monitor Reactor water conductivity at RWCU-CR-601 per PPM 1.13.1, Chemical Process Management and Control.

6. Refer to Technical Specification 3.3.6.1 and 3.3.6.2 in Modes 1, 2, and 3.

7. If applicable, refer to PPM 4.12.4.1A, High Energy Line Break.

8. If applicable, refer to PPM 4.11.2.1, Liquid Radioactive Spills.

REFERENCES:
EWD-1E-0060
EWD-4E-0016
EWD-19E-0029
EWD-19E-0041
CVI 02E31-05, 7, 11 (Leak Det.)
LER 92-039-00-05

{P-60468}

6-2 LEAK DET RWCU/RCIC PIPE AREA TEMP HIGH

6-2 WINDOW	SOURCE	AUTOMATIC ACTIONS
LEAK DET RWCU/RCIC PIPE AREA TEMP HIGH	LD-TRS-624 • point 2 - 118° • point 3 - 130° • point 4 - 128° • point 5 - 125° • point 6 - 140°	None

1. Identify the source of the high temperature.

2. Compare recorder reading with temperature points on H13-P632 and H13-P642:

LD-TRS-624	LD-MON-1A	LD-MON-1B
Point 2	A3-3 (LD-TE-24A)	A3-3 (LD-TE-24B)
Point 3	A1-4 (LD-TE-24C)	A1-4 (LD-TE-24D)
Point 4	A2-4 (LD-TE-24E)	A2-4 (LD-TE-24F)
Point 5	A3-4 (LD-TE-24G)	A3-4 (LD-TE-24H)
Point 6	A1-5 (LD-TE-24J)	A1-5 (LD-TE-24K)

3. Ensure adequate ventilation to area.

4. Check the RWCU/RCIC PIPE AREA for leakage and isolate if possible.

5. Refer to PPM 5.3.1, Secondary Containment Control.

6. If a high energy release is confirmed:

   a. Refer to PPM 4.12.4.1A, High Energy Line Break.

   b. Refer to PPM 13.1.1, Classifying the Emergency.

REFERENCES: EWD-19E-0008, 0026, 0030

2-2 LEAK DET RWCU ROOMS TEMP HI-HI

2-2 WINDOW	SOURCE	AUTOMATIC ACTIONS
LEAK DET RWCU ROOMS TEMP HI-HI	Any of the following on LD-MON-1B, after a 1.0 second time delay: A1-2 (160°F) A2-2 (160°F) A3-2 (150°F) A3-3 (160°F) A1-4 (160°F) A2-4 (160°F) A3-4 (160°F) (LD-MON-1B/K6)	• RWCU-V-1 Closes
	A1-5 (160°F) (LD-MON-1B/K8)	• RWCU-V-1 Closes • RCIC-V-63 Closes • RCIC-V-76 Closes

1. Identify the alarmed temperature point(s) on LD-MON-1B (H13-P642) and compare with reading of adjacent sensors on LD-MON-1A as indicated below:

LD-MON-1B (H13-P642)	LOCATION	LD-MON-1A (H13-P632)	LD-TRS-608
A1-2 (LD-TE-3B)	RWCU-P-1A Rm	A1-2 (LD-TE-3A)	Point 10
A2-2 (LD-TE-3D)	RWCU-P-1B Rm	A2-2 (LD-TE-3C)	Point 11
A3-2 (LD-TE-3F)	RWCU-HX-Rm	A3-2 (LD-TE-3E)	Point 12

LD-MON-1B (H13-P642)	LOCATION	LD-MON-1A (H13-P632)	LD-TRS-624
A3-3 (LD-TE-24B)	548' NORTH PIPE CHASE	A3-3 (LD-TE-24A)	Point 2
A1-4 (LD-TE-24D)	548' SOUTH PIPE CHASE	A1-4 (LD-TE-24C)	Point 3
A2-4 (LD-TE-24F)	522' NORTH RHR A/C VLV RM	A2-4 (LD-TE-24E)	Point 4
A3-4 (LD-TE-24H)	522' RWCU PUMP RM MEZZANINE	A3-4 (LD-TE-24G)	Point 5
A1-5 (LD-TE-24K)	501' TIP RM MEZZANINE	A1-5 (LD-TE-24J)	Point 6

(CONTINUED ON NEXT PAGE)

2-5 LEAK DET RWCU ROOMS TEMP HI-HI

2-5 WINDOW	SOURCE	AUTOMATIC ACTIONS
LEAK DET RWCU ROOMS TEMP HI-HI	Any of the following on LD-MON-1A, after a 1.0 second time delay: A1-2 (160°F) A2-2 (160°F) A3-2 (150°F) A3-3 (160°F) A1-4 (160°F) A2-4 (160°F) A3-4 (160°F) (LD-MON-1A/K6) or A1-5 (160°F) (LD-MON-1A/K8)	• RWCU-V-4 Closes • RWCU-V-4 Closes • RCIC-V-8 Closes

NOTE: The high alarm setpoint is set 20°F (minimum) LT the HI-HI (Isolation) setpoint.

1. Identify the alarmed temperature point(s) on LD-MON-1A (H13-P632) and compare with reading of adjacent sensors as indicated below:

LD-MON-1A (H13-P632)	LD-TRS-608	LOCATION	LD-MON-1B (H13-P642)
A1-2 (LD-TE-3A)	Point 10	RWCU-P-1A Rm	A1-2 (LD-TE-3B)
A2-2 (LD-TE-3C)	Point 11	RWCU-P-1B Rm	A2-2 (LD-TE-3D)
A3-2 (LD-TE-3E)	Point 12	RWCU-HX-Rm	A3-2 (LD-TE-3F)

LD-MON-1A (H13-P632)	LD-TRS-624	LOCATION	LD-MON-1B (H13-P642)
A3-3 (LD-TE-24A)	Point 2	548' NORTH PIPE CHASE	A3-3 (LD-TE-24B)
A1-4 (LD-TE-24C)	Point 3	548' SOUTH PIPE CHASE	A1-4 (LD-TE-24D)
A2-4 (LD-TE-24E)	Point 4	522' RWCU PIPE CHASE	A2-4 (LD-TE-24F)
A3-4 (LD-TE-24G)	Point 5	522' RWCU PUMP RM MEZZANINE	A3-4 (LD-TE-24H)
A1-5 (LD-TE-24J)	Point 6	501' RWCU/RCIC MEZZANINE	A1-5 (LD-TE-24K)

(CONTINUED ON NEXT PAGE)

1998 Group 1
NRC Exam Analysis

Exam Bank #	RO Exam #	SRO Exam #	A	B	C	D	Miss Rate	Comment
98001	24	10	X	2			16.7%	2 RO
98002		1		X			0.0%	
98003	4	2		2	X		16.7%	2 SRO
98004	27	5			2	X	16.7%	2 RO
98005	5	15	X		2	1	25.0%	1 RO, 2 SRO
98006	8	6		X			0.0%	
98007	94	7			X		0.0%	
98008		8				X	0.0%	
98009	95	9	X				0.0%	
98010	9	11	9	X		3	100.0%	
98011	12	94			X		0.0%	
98012	89	14				X	0.0%	
98013	13	16	X				0.0%	
98014	16	83		X		1	8.3%	1 RO
98015	18	84			X		0.0%	
98016	86	17	3			X	25.0%	1 RO, 2 SRO
98017	19	20	X				0.0%	
98018		25		X			0.0%	
98019	75	21	1		X		8.3%	1 SRO
98020	20	24	1		1	X	16.7%	1 RO, 1SRO
98021		42	X				0.0%	
98022	21	76	2	X			16.7%	2 RO
98023		57			X		0.0%	
98024		43				X	0.0%	
98025		44				X	0.0%	
98026	17	47	X				0.0%	
98027	22	48		1		X	8.3%	1 SRO
98028	23	45	X	1	1	4	50.0%	2 RO, 4 SRO
98029	25	46			1	X	8.3%	1 SRO
98030	2	26	1	1		X	16.7%	1 RO, 1SRO
98031	3	27		X		1	8.3%	1 RO
98032	26	33	4		X		33.3%	1 RO, 3 SRO
98033	28	34	X				0.0%	
98034	35	28				X	0.0%	
98035	29	29	X			1	8.3%	1 SRO
98036	30	30		9		X	75.0%	3 RO, 6 SRO
98037							0.0%	Not Used
98038	32	32	X		1		8.3%	1 RO
98039		35	1	X			12.5%	1 SRO
98040		36				X	0.0%	
98041		18			X	1	12.5%	1 SRO
98042	33	19	1	X			8.3%	1 RO
98043	34	37		X			0.0%	
98044	36	38	X	1			8.3%	1 RO
98045	76	39				X	0.0%	
98046	37	40		3		X	25.0%	2 RO, 1 SRO
98047	38	41	X				0.0%	
98048	39	49	1		X		8.3%	1 RO
98049		50				X	0.0%	
98050	40	51		X			0.0%	

1998 Group 1
NRC Exam Analysis

Exam Bank #	RO Exam #	SRO Exam #	A	B	C	D	Miss Rate	Comment
98051	41	52	X				0.0%	
98052	42	53				X	0.0%	
98053	43	54			X		0.0%	
98054	44	55	X	2			16.7%	2 SRO
98055	46	56			X		0.0%	
98056	51	75				X	0.0%	
98057	47	58		X		2	16.7%	2 RO
98058	48	22	X				0.0%	
98059	85					X	0.0%	
98060	50	59		X		2	16.7%	1 RO, 1 SRO
98061	52	60	X				0.0%	
98062	56	61			X		0.0%	
98063	53	62			X		0.0%	
98064	62	63	X				0.0%	
98065	54	64				X	0.0%	
98066		67		X			0.0%	
98067	55	68	1		1	X	16.7%	1 RO, 1 SRO
98068	57	71	X				0.0%	
98069	58	72	X			1	8.3%	1 SRO
98070	59	69			X		0.0%	
98071	60	70			X	5	41.7%	1 RO, 4 SRO
98072	61	73		X			0.0%	
98073	63	74	X	1	1		16.7%	2 SRO
98074	70	77				X	0.0%	
98075	81	78	1			X	8.3%	1 RO
98076	64	81			X		0.0%	
98077	66	82				X	0.0%	
98078	67	85	X				0.0%	
98079		86			X		0.0%	
98080	68	12	X				0.0%	
98081	1	13			X		0.0%	
98082	69	90				X	0.0%	
98083	71	65		X			0.0%	
98084		66	1		X	1	25.0%	2 SRO
98085		91	1	1	X		25.0%	2 SRO
98086		92	2		X		25.0%	2 SRO
98087		93				X	0.0%	
98088		79	X				0.0%	
98089		80	1			X	12.5%	1 SRO
98090		95	X	1			12.5%	1 SRO
98091		3	2		X		25.0%	2 SRO
98092	72	4			X	1	8.3%	1 RO
98093		96	X			1	12.5%	1 SRO
98094		87				X	0.0%	
98095		88	1	2		X	37.5%	3 SRO
98096		89		X			0.0%	
98097		97	X	1		2	37.5%	3 SRO
98098		98		X		1	12.5%	1 SRO
98099		99		X			0.0%	
98100		100				X	0.0%	

1998 Group 1
NRC Exam Analysis

Exam Bank #	RO Exam #	SRO Exam #	A	B	C	D	Miss Rate	Comment
98101	73			X	1		25.0%	1 RO
98102	74					X	0.0%	
98103	31	31	X	2			16.7%	1 RO, 1 SRO
98104	77		X				0.0%	
98105	45				X		0.0%	
98106	78				X		0.0%	
98107	65				X		0.0%	
98108	79			X	2		50.0%	2 RO
98109	80				2	X	50.0%	2 RO
98110	82				X		0.0%	
98111	83					X	0.0%	
98112	100					X	0.0%	
98113	84			X		2	50.0%	2 RO
98114	14		X				0.0%	
98115	15				X		0.0%	
98116	49	23		X		1	8.3%	1 RO
98117	87		X			1	25.0%	1 RO
98118	88		2	2	X		100.0%	4 RO
98119	90					X	0.0%	
98120	6		X				0.0%	
98121	7			X			0.0%	
98122	91			1	X		25.0%	1 RO
98123	92				1	X	25.0%	1 RO
98124	93				X		0.0%	
98125	96			X			0.0%	
98126	97		X				0.0%	
98127	10		1			X	25.0%	
98128	11		1		X		25.0%	1 RO
98129	98			X			0.0%	1 RO
98130	99		X				0.0%	