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RECIP.NAME RECIPIENT AFFILIATION
Document Control Branch (Document Control Desk)

See Rpt.

SUBJECT: Forwards 60 day response to NRC 950207 RAI in support of INEL work under NRC contract to update Reg Guide 1.154, "Format & Content of Plant Specific PTS SARs for PWRs," including End Path Procedures (EPP) EPP-11 & EPP-16.

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CP&L

Carolina Power & Light Company
Robinson Nuclear Plant
3581 West Entrance Road
Hartsville SC 29550

Robinson File No.: 13510I
Serial: RNP-RA/95-0067

APR 13 1995

United States Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, DC 20555

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
DOCKET NO. 50-261/LICENSE NO. DPR-23
60 DAY RESPONSE TO REQUEST FOR INFORMATION DATED FEBRUARY 7, 1995

Gentlemen:

By letter dated February 7, 1995, the NRC requested certain information in support of Idaho National Engineering Laboratory's (INEL's) work under an NRC contract to update Regulatory Guide 1.154, "Format and Content of Plant Specific Pressurized Thermal Shock Safety Analysis Reports for Pressurized Water Reactors." The INEL work will involve thermal hydraulic calculations using the RELAP5/MOD3 code utilizing an existing input deck for H. B. Robinson Steam Electric Plant, Unit No. 2 which has not been updated since 1984. The requested information will be used to update the input deck.

The NRC requested that responses to items 7, 8, and 9 be provided within 30 days of receipt of the request, with responses to the remaining items within 60 days of receipt of the request. The 30 day response was provided by letter dated March 13, 1995, with a commitment to provide the response to item 7 in the 60 day response. The enclosure to this letter provides a complete response to items 1 through 4, and provides a status for the remaining items 5 through 7.

The Westinghouse Steam Generator Thermal and Hydraulic Report requested in item 5 is proprietary, and a non-proprietary summary has not been developed to support a request for withholding from public disclosure in accordance with 10 CFR 2.790. Because the development of a non-proprietary summary of the report involves significant effort by the Westinghouse Electric Corporation, the NRC should request the report from Westinghouse Electric Corporation directly.

A001

The response to items 6 and 7 requires that archived data be obtained from the Emergency Response Facility Information System. This information will not be available until May 1995, because of the significant effort required to retrieve the applicable archived data in a format usable to a third party.

Carolina Power & Light Company requests a copy of the RELAP5/MOD3 model input data listing on diskette when available.

Questions regarding this matter may be referred to Mr. K. R. Jury at (803) 857-1363.

Very truly yours,

Dan Stoddard for

R. M. Krich
Manager - Regulatory Affairs

Enclosure

c: Mr. S. D. Ebneter, Regional Administrator, USNRC, Region II
Ms. B. L. Mozafari, USNRC Project Manager, HBRSEP
Mr. W. T. Orders, USNRC Senior Resident Inspector, HBRSEP

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
60 DAY RESPONSE TO REQUEST FOR INFORMATION DATED FEBRUARY 7, 1995
ITEMS 1 THROUGH 7

Item 1

"Description of the operating procedures for tripping RCPs during a plant transient."

Response

Regarding criteria for manual Reactor Coolant Pump trip, plant procedures were being revised at the approximate time that discussions were taking place between Carolina Power & Light (CP&L) Company and Idaho National Engineering Laboratory (INEL). In combination with Safety Injection, the new criteria is less than 35 ^{0}F subcooling for normal containment conditions or less than 55 ^{0}F subcooling for adverse containment conditions (Reference 1). "Adverse containment conditions" are defined as pressure greater than 4 psig at the bottom of page 4 of the Safety Evaluation for NUREG-0737 Item II.K.3.5, "Auto Trip of RCPs" (Reference 2).

Item 2

"Information concerning the reactor coolant system (RCS) subcooling margin instrumentation, especially concerning the RTD bypass piping and how subcooling margin is calculated and displayed in the plant's control room."

Response

Pages 3 and 4 of Reference 2 explain that the extent of subcooling is determined by a microprocessor which receives inputs from four primary system pressure transmitters, six loop resistance temperature detectors (hot and cold leg for each loop), and 16 core exit thermocouples.

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
60 DAY RESPONSE TO REQUEST FOR INFORMATION DATED FEBRUARY 7, 1995
ITEMS 1 THROUGH 7

Item 3

"The pump curves for both HPI pumps and a single HPI pump. The curves should describe the mass flow rate from the pumps as a function of system pressure."

Response

Safety Injection delivery curves are attached as Reference 3. The curves are intended to be conservatively low estimates.

a. Two pump delivery is attached as Updated Final Safety Analysis Report (UFSAR), Figure 15.1.5-1. The figure is taken from Figure 2.1 in Siemens report XN-NF-84-74 Supplement 3, "Confirmatory Analysis of the Steamline Break Event," January 1985. It also is presented as Figure 2.5 in Siemens report EMF-89-081, "Plant Parameters for H.B. Robinson, Unit 2," Revision 4.

b. A single pump delivery curve is attached and titled Figure 3.4. This figure will be included in the next UFSAR revision and is derived from the draft Siemens report EMF-95-032 "Main Steam Line Break Analysis for H.B. Robinson Unit 2," and is based on 150% of Table 2.31 in EMF-89-081.

Item 4

"Operational procedures for controlling secondary level in unaffected steam generators and for a steam generator with a main steam line break."

Response

Uncontrolled copies of plant procedures Emergency Operating Procedures End Path Procedure (EPP)-11, "Faulted Steam Generator Isolation," and EPP-16, "Uncontrolled Depressurization of All Steam Generators," are attached as Reference 4. The 10% to 50% target range for steam generator level is required in Steps 11, 12, 37, and 64b.

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
60 DAY RESPONSE TO REQUEST FOR INFORMATION DATED FEBRUARY 7, 1995
ITEMS 1 THROUGH 7

Item 5

"The Westinghouse Steam Generator Thermal Hydraulic Report for HBR because this document can verify performance of the current steam generator model."

Response

The "Model 44F Steam Generator Thermal and Hydraulic Design Data Report For Carolina Power & Light Company H. B. Robinson No. 2 (CPL)," Revision 3, dated April 1, 1985, is a document which contains proprietary information owned by the Westinghouse Electric Corporation. A copy of the document should be obtained directly from Westinghouse Electric Corporation.

Item 6

"CP&L's Emergency Response Facility Information System data for available plant transients for validation of RELAP5/MOD3 response."

Response

H. B. Robinson Steam Electric Plant, Unit No. 2 experienced reactor trips on April 3, 1994, and on August 2, 1994. A request for the archived data has been submitted to the Emergency Response Facility Information System (ERFIS) custodian. The extracted data will be forwarded as soon as it is available, currently expected in early May 1995.

Item 7

"The range of accumulator temperature given in Table 6.3.2-2 of the Updated Final Safety Analysis Report (UFSAR) for HBR is 70 to 120° F. Can you verify that this is accurate? What is the seasonal mean water temperature for the accumulators?"

Response

There is no direct measurement of accumulator fluid temperature. Because the air temperature in the reactor containment building may give some indication of water temperature, a request for archived data has been submitted to the ERFIS custodian. The extracted data will be provided as soon as it is available, currently expected in early May 1995.

H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2
60 DAY RESPONSE TO REQUEST FOR INFORMATION DATED FEBRUARY 7, 1995
ITEMS 1 THROUGH 7

ATTACHED REFERENCES

1. Excerpt from plant procedure End Path Procedure (EPP), "EPP-Foldouts", Revision 15.

2. NRC letter dated July 7, 1987, "NUREG-0737 ITEM II.K.3.5 'Auto Trip of RCPs', TAC No. 49690."

3. a. Updated Final Safety Analysis Report, Figure 15.1.5-1

   b. Figure 3.4, "High Head Safety Injection Delivery Curve Based on One Pump with No Spills," from, "Main Steam Line Break Analysis for H. B. Robinson Unit 2," EMF-95-032(P), Draft.

4. a. End Path Procedure, EPP-11, "Faulted Steam Generator Isolation," Revision 4

   b. End Path Procedure, EPP-16, "Uncontrolled Depressurization Of All Steam Generators," Revision 10.

REFERENCE 1

CONTINUOUS USE

CAROLINA POWER AND LIGHT COMPANY

H. B. ROBINSON SEG PLANT

PLANT OPERATING MANUAL

VOLUME 3

PART 4

END PATH PROCEDURE

EPP-Foldouts

FOLDOUTS

REVISION 15

Effective Date 12/16/94

RECOMMENDED BY: ______________________ 12/16/94
Operations Procedure Coordinator Date

APPROVED BY: ______________________ for 12/16/94
Manager - Operations Date

UNCONTROLLED

1.0 PURPOSE

This procedure provides actions to respond to circumstances within the EOP Network which are time independent.

2.0 ENTRY CONDITIONS

NOTE
The Foldouts do not apply during performance of the FRPs.

When directed by the EOP Network. Only one Foldout is applicable at a time.

CONTINUOUS USE

FOLDOUT A

(Page 1 of 4)

1. RCP TRIP CRITERIA

IF BOTH conditions below are met, THEN stop all RCPs:

- SI Pumps - AT LEAST ONE RUNNING
- RCS Subcooling - LESS THAN 35°F [55°F]

2. SI ACTUATION CRITERIA

IF EITHER condition below occurs, THEN Actuate SI and Go To PATH-1, Entry Point A:

- RCS Subcooling - LESS THAN 35°F [55°F]
- PZR Level - CAN NOT BE MAINTAINED GREATER THAN 10% [26%]

3. AFW SUPPLY SWITCHOVER CRITERIA

IF CST level decreases to less than 10%, THEN switch to backup water supply using OP-402, Auxiliary Feedwater System.

4. EMERGENCY COOLING WATER SWITCHOVER CRITERIA

IF normal cooling is lost to any of the following components, THEN establish emergency cooling water using the referenced procedure:

- Charging Pump Oil Coolers - Use Attachment 1 of AOP-014, Component Cooling Water System Malfunction.
- SDAFW Pump Oil Coolers - Use Attachment 1 of AOP-022, Loss of Service Water.
- SI Pump Thrust Bearing - Use Attachment 2 of AOP-022, Loss of Service Water.
- MDAFW Pumps - Use Attachment 3 of AOP-022, Loss of Service Water.

REFERENCE 2

RECEIVED JUL 0 9 1987

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

JUL 07 1987

7-9-87
NLS-87-382

Docket No. 50-261

Mr. E. E. Utley
Senior Executive Vice President
Power Supply and Engineering & Construction
Carolina Power & Light Company
Post Office Box 1551
Raleigh, North Carolina 27602

Dear Mr. Utley:

SUBJECT: NUREG-0737 ITEM II.K.3.5 "AUTO TRIP OF RCPs", TAC NO. 49690

We have completed our review of your submittals dated September 30, 1985, December 30, 1986 and February 19, 1987 on this subject and find your responses acceptable. Consequently, we consider this item closed for H. B. Robinson, Unit No. 2.

The enclosed Safety Evaluation documents our findings.

Sincerely,

Kenneth T. Eccleston

Kenneth T. Eccleston, Senior Project Manager
Project Directorate II-1
Division of Reactor Projects I/II
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc w/enclosure:
See next page

8707130087 2pp

Mr. E. E. Utley
Carolina Power & Light Company

H. B. Robinson 2

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

ENCLOSURE 1

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION RELATING TO IMPLEMENTATION OF TMI ACTION ITEM II.K.3.5 "AUTOMATIC TRIP OF REACTOR COOLANT PUMPS" (RESPONSE TO GENERIC LETTER NO. 85-12) CAROLINA POWER & LIGHT COMPANY H. B. ROBINSON STEAM ELECTRIC PLANT, UNIT NO. 2 DOCKET NO. 50-261

1.0 SUMMARY

We reported in Reference 1 that the information provided by the Westinghouse Owners Group (WOG) in support of alternative Reactor Coolant Pump (RCP) trip was acceptable on a generic basis. The review noted that a number of considerations were assigned plant specific status. Accordingly, we requested that operating reactor licensees select and implement an appropriate RCP trip criterion based upon the WOG methodology. This Safety Evaluation Report (SER) contains the staff's findings concerning this issue for Carolina Power & Light Company's H. B. Robinson Steam Electric Plant Unit No. 2.

Reference 1 required owners of Westinghouse Nuclear Steam Generating Systems to evaluate their plants with respect to RCP trip. The objective was to demonstrate that their proposed RCP trip setpoints assure pump trip for small break LOCAs, and in addition to provide reasonable assurance that RCPs are not tripped unnecessarily during non-LOCA events. A number of plant specific items were identified which were to be considered by applicants and licensees, including the selected RCP trip parameter, instrumentation quality and redundancy, instrumentation uncertainty, possible adverse environments, calculational uncertainty, potential RCP and RCP associated problems, operator training, and operating procedures.

The licensee has addressed the Reference 1 criteria and we have evaluated this information. We find the material submitted by the licensee to be acceptable

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and find that the licensee has satisfied the requirements in regard to TMI Action Item II.K.3.5.

Further information regarding the acceptance criteria and our evaluation is provided in the remainder of this SER.

2.0 BACKGROUND

TMI Action Plan Item II.K.3.5 of NUREG-0737 (Ref. 2) required all licensees to consider solutions pertinent to tripping RCPs under transient and Loss Of Coolant Accident (LOCA) conditions. A summary of the industry and NRC programs concerning RCP trip was provided in SECY-82-475 (Ref. 3). Reference 3 also provided NRC guidance and criteria for resolution of II.K.3.5, and enclosed Generic Letters 83-10 (Ref. 4), which outlined requirements pertinent to RCP trip.

The Westinghouse Owners Group responded to Reference 4 by developing a set of three alternative RCP trip criteria and provided information which individual utilities could use to select the criterion best suited to their plants (Refs. 5 - 7). The staff then issued Generic Letter 85-12, and directed each applicant and licensee to provide RCP trip criteria and substantiating information.

The licensee addressed this issue in Reference 8, which was reviewed by the staff and was discussed with the licensee in a telephone conference call (Ref. 9). This resulted in an additional licensee submittal (Ref. 10). The staff has conducted a detailed review of this information, as documented in Reference 9 and in the remainder of this SER.

3.0 EVALUATION

The staff finds that Carolina Power & Light Company has complied with the requirements of Generic Letter 85-12 and that they have therefore met the requirements in regard to implementation of TMI Action Item II.K.3.5.

Organization of the remainder of this SER Section is essentially identical to that of the Generic Letter (Ref. 1) to which the licensee responded. Each criterion is listed, followed by a staff summary of the licensee position and a staff evaluation.

A. Determination of RCP Trip Criteria

The licensee has selected the subcooling criterion as the basis for RCP trip. This is one of the three criteria which the staff previously found acceptable on a generic basis.

The quantitative value associated with subcooling that is to be used as the trip criterion is identified as 25° F for normal containment conditions and 35° F for adverse containment conditions. The selection was based on the Westinghouse Owners Group (WOG) Emergency Response Guidelines and recommendations. Plant specific calculations eliminated use of RCS pressure as a criterion due to insufficient separation of LOCA and non-LOCA events. A pressure of less than 1230 psi was needed, and 1300 psi was obtained. Selection of RCS to steam generator secondary pressure differential was rejected because this requires an operator calculation, which was judged to be an unnecessary complication since RCS subcooling met the criteria.

A1. Instrumentation Identification, Including Redundancy and Quality Level

CP&L has elected to use subcooling as determined by a microprocessor which receives inputs from four primary system pressure transmitters, six loop RTDs, and 16 core exit thermocouples, and which outputs a margin to saturation. The subcooling monitor is stated to possess redundancy in that it is comprised of two channels which operate completely independently. Each channel is stated to be powered from a critical instrument bus which receives power from off site or emergency diesels. Testing capability is provided via front panel test switches which test the warning lights, alarms, meter movements, and associated electronics.

Redundant control grade temperature inputs are used from each hot and cold leg RTD. These are backed up by multiple core exit thermocouples. Each channel is provided with three pressure signals, one narrow range safety grade pressure and two wide range control grade pressures. All safety grade sensors are isolated from the subcooling monitor by isolation amplifiers. References are provided to further information concerning the reactor coolant subcooling monitor.

Control grade instrumentation was permitted for the existing subcooling monitoring system since much of the applicable work was performed prior to issue of Regulatory Guide 1.97. All instrumentation and transmitters associated with the subcooling calculations are located within containment.

There are two subcooling monitors. Operating procedures are in place which cover response if these are inoperative. These include calculation methods which are based upon in core thermocouples or loop resistance temperature detectors (RTDs) and pressurizer pressure. (Narrow range pressurizer pressure involves safety related instrumentation, whereas wide range pressure does not.) Narrow range pressure would be used in the range of applicability, which ranges from 1700 to 2500 psi. The thermocouples will be upgraded to comply with the HBR2 Reg. Guide 1.97 commitments during the 1987 refueling outage.

These items meet the staff criteria and are acceptable.

A2. Instrumentation Uncertainties for Normal and Adverse Environments

Instrumentation uncertainty for normal and adverse containment environmental conditions is stated to be 25° F and 35° F, respectively. (References are provided for information pertinent to the determination.) An adverse containment condition is assumed to exist if containment pressure is greater than 4 psig.

Following the 1987 refueling outage, the instrumentation associated with the subcooling calculation will meet Regulatory Guide 1.97 separation criteria. The monitors are separated consistent with the requirements for safety related equipment. The two wide range pressure transmitters are widely separated, and their relative locations are similar to Engineered Safety Features instrumentation.

The uncertainty associated with various conditions has been addressed. The staff finds this treatment acceptable.

A3. Analysis Uncertainties

Calculations of instrument uncertainties are summarized, and comparisons are discussed between plant data and calculations in the Westinghouse Owners Group (WOG) information. CP&L states that the calculated overall uncertainty for H. B. Robinson is from +1° F to +5° F for the subcooling trip point.

The Staff felt the uncertainty was smaller than normally anticipated for generic calculations and requested confirmation of the values. In response, the Licensee indicated that the uncertainty range is a Westinghouse provided value which covers the variation between the generic calculations and the Robinson Unit 2 plant. The WOG performed bounding calculations and plant information were reviewed by the Licensee, and determined to be directly applicable to the Robinson Unit 2 plant with no need for changes or modifications to the calculations.

The licensed Westinghouse LOFTRAN computer code is referenced for performance of the non-LOCA analyses. The computer program result uncertainties evaluation is based on the assumption of no changes in initial plant conditions (such as full power, pressurizer level, all Safety Injection (SI) pumps running, and all Auxiliary Feed Water (AFW) pumps running). The major contributors to uncertainty are stated to be break flow rate, SI flow rate, decay heat generation rate, and AFW flow rate. Parametric studies are summarized in which the major uncertainties are stated to be due to the break flow model and SI flow inputs.

The licensee has not directly addressed such topics as the accuracy of the numerical solution scheme or of nodalization. Further, there is no determination of the influence of equipment or operational failures. Information pertinent to the former result from comparisons of the LOFTRAN code to operational and experimental data, and as a result should have been factored into the calculational basis and included in the uncertainty value. Determination of equipment or operational failures is not a necessity as long as the expected configuration of the plant is addressed since the objective of RCP trip is to provide reasonable assurance of not tripping for transients for which a trip is undesirable. It is not necessary to establish that one will never trip unnecessarily since the plant is capable of being safely controlled if an unnecessary trip does occur.

The staff finds the information presented under Item A3 to be acceptable.

B. Potential Reactor Coolant Pump Problems

B1. Containment Isolation Impact Upon RCP Operation

CP&L has stated that it is their policy to trip RCPs if essential services are lost unless RCP operation is required to prevent core damage. No actions occur which affect RCP operation in response to a containment phase A isolation. A phase B containment isolation signal causes rerouting of the seal water return to the pressurizer relief tank as well as isolation of Component Cooling Water (CCW). Seal injection is maintained unless a loss of offsite power has occurred. If this happens, the RCPs stop since they are powered from offsite power. Two of the three charging pumps are powered from emergency buses, but do not restart automatically following loss of offsite power.

RCPs are tripped in response to a complete loss of CCW to the RCPs since this causes loss of cooling to both the thermal barrier heat exchangers and to other pump components. RCP operation is allowed if seal injection is lost, but CCW continues to be available. Conversely, if CCW is lost to the RCP thermal barriers but is available to other RCP components, and seal injection remains available, RCP operation is allowed.

Restart of CCW and/or seal injection after its termination is covered by plant procedures. The major concern is avoidance of thermal shock, which could damage pump components as well as lead to seal leakage.

The staff discussion of RCP operation and operator actions with the licensee clearly established a depth of knowledge on the part of plant personnel with respect to this topic. The licensee has stated that no essential services to the RCPs are lost in a Phase A containment isolation, but that most essential services are lost as a result of a Phase B containment isolation. Instructions for tripping the RCPs under these conditions are provided, as are RCP restart procedures. The staff finds that the issues pertinent to Item B1 are adequately addressed.

B2. Components Required for RCP Trip

The major components associated with RCP trip are identified, as is their location. Separation and interaction with high energy lines is stated by the licensee to be consistent with the prior discussion on these topics.

The RCP trip breakers are located near the control room, and can be readily tripped. Two of the RCPs could additionally be tripped by deenergizing buses. The third RCP could be tripped by deenergizing its respective bus, but critical equipment is also supplied from that bus. Loss of bus voltage on that bus would result in the start of a diesel generator, which in turn would reenergize the affected emergency bus so that critical equipment would be supplied with electrical power. This last option would not normally be used to trip RCPs.

RCP trip criteria are designed to avoid RCP operation under voided conditions. (Note there is an exception under response to inadequate core cooling conditions, where RCPs could be operated regardless of the RCS inventory condition.)

The staff concludes that the licensee has adequately addressed the issues pertaining to equipment required to trip the RCPs.

C. Operator Training and Procedures (RCP Trip)

C1. Operator Training

The formal submittal regarding operator training is brief. However, discussion with the licensee clearly established an understanding of plant behavior and of RCP operation on the part of the Licensee. The Licensee nas stated that the background information pertinent to the need for RCP trip as contrasted to RCP operation is covered in both training and simulator operation.

If RCP trip were required in response to an event, and the operators were to miss the trip, then the RCPs would ordinarily be tripped when the mistake was identified. Plant response and necessary operator actions from that point are covered in the plant emergency operating procedures.

An additional qualifier on RCP trip is that safety injection must be available. RCPs are left running if there is no safety injection capability.

The licensee has demonstrated an understanding of background concerns and the need for RCP trip as contrasted to running RCPs. In regard to operation outside the design basis, the operator is instructed not to trip RCPs if there is no safety injection available. If an operator were to fail to trip RCPs, they would apparently be tripped at the time the mistake was discovered. It would be better to check on plant status prior to taking this action. For example, if the reactor vessel liquid level instrumentation system indicated that inventory was insufficient to exclude an Inadequate Core Cooling (ICC) condition if RCPs were tripped, then clearly, a trip should not be performed.

The instructions appear consistent with the Westinghouse Owners Group (WOG) recommendations for a condition where no ECCS is available. Further, under such a condition, the operators may follow the WOG recommendations, and may run RCPs as long as is reasonable to extend core cooling beyond what would be obtained if the RCPs were tripped under these unique and unexpected conditions. Similar instructions are provided to operators at many plants where the procedures are based on WOG recommendations. Such actions are taken under conditions which are unlikely, which are beyond the design basis, and which are beyond conditions reported in the FSAR.

The staff notes that regulations require acceptable clad temperatures throughout all operation modes, and that these temperatures must be calculated according to approved models. Operating procedures which are dealing with conditions within the design basis should not result in a violation of the required temperatures. Therefore, timing considerations associated with RCP trip should extend sufficiently far into a transient that the window of small break LOCA vulnerability is covered. This is applicable to the situation where an RCP trip is missed during an accident, with later discovery of the operator error.

The staff has previously stated that, given the above described operator error, the RCPs should be left running. However, the staff has not conducted an indepth review of such situations. There are logical arguments to support both positions. Since many Westinghouse plant licensees have incorporated this WOG provided guidance into their emergency procedures, the present staff thinking is to accept these approaches unless there is a clear need for rejection. No such need is presently apparent. (In general, the staff supports providing operators with the latitude to apply their training and understanding in the mitigation of plant adverse conditions provided such actions are consistent with procedures.) This item will be addressed generically.

C2. Procedures Related to RCP Operation

The licensee has listed several applicable procedures and has provided references from which the Staff can obtain additional information, including all applicable procedures which are available to the Staff at the plant site. This is sufficient information for staff purposes since it is beyond the scope of the present review to perform an indepth evaluation of procedures.

4.0 CONCLUSION

Each of the points identified in Reference 1 has been satisfactorily addressed by the licensee. Further, the licensee has considered items pertinent to RCP trip and operation which are in addition to the Reference 1 requirements. The staff finds the licensee treatment of RCP trip to be acceptable and the Licensee has satisfied the requirements of TMI Action Item II.K.3.5.

5.0 REFERENCES

1. Thompson, Hugh L. Jr., "Implementation of TMI Action Item II.K.3.5, 'Automatic Trip of Reactor Coolant Pumps' (Generic Letter No. 85-12)," Letter from Director, Division of Licensing, NRC, to all applicants and licensees with Westinghouse (W) designed nuclear steam supply systems (NSSSs), June 28, 1985.

2. "Clarification of TMI Action Plan Requirements," NUREG-0737, US NRC, November 1980.

3. Dircks, William J., "Staff Resolution of the Reactor Coolant Pump Trip Issue," Policy Issue for the Commissioners from the Executive Director for Operations, NRC, SECY-82-475, November 30, 1982.

4. Eisenhut, Darrell G., "Resolution of TMI Action Item II.K.3.5, 'Automatic Trip of Reactor Coolant Pumps,' (GENERIC LETTER No. 83-10)," Letter to all applicants with (PWR vendor) designed nuclear steam supply systems from Director, Division of Licensing, NRC.

5. Westinghouse Owners Group, Letter OG-110, "Evaluation of Alternate RCP Trip Criteria," December 1, 1983.

6. Westinghouse Owners Group, Letter OG-117, "Justification of Manual RCP Trip for SBLOCA Events," March 9, 1984.

7. Westinghouse Owners Group, Letter OG-137, "Response to NRC Questions on RCP Trip," October 25, 1984.

8. Zimmerman, S. R., "H. B. Robinson Steam Electric Plant, Unit No. 2, Docket No. 50-261/License No. DPR-23, Response to TMI Action Item II.K.3.5, Generic Letter 85-12," Letter to Hugh L. Thompson, Jr., NRC, from Carolina Power & Light Company, Serial: NLS-85-335, NRC TAC # 49690, Sep 30, 1985.

9. Requa, Glode, "Reactor Coolant Pump Trip (TMI Action Item II.K.3.5, Generic Letter 85-12; Conference Call, H. B. Robinson Steam Electric Plant Unit 2", Letter from NRC to E. E. Utley, Carolina Power & Light Company, Dec. 30, 1986.

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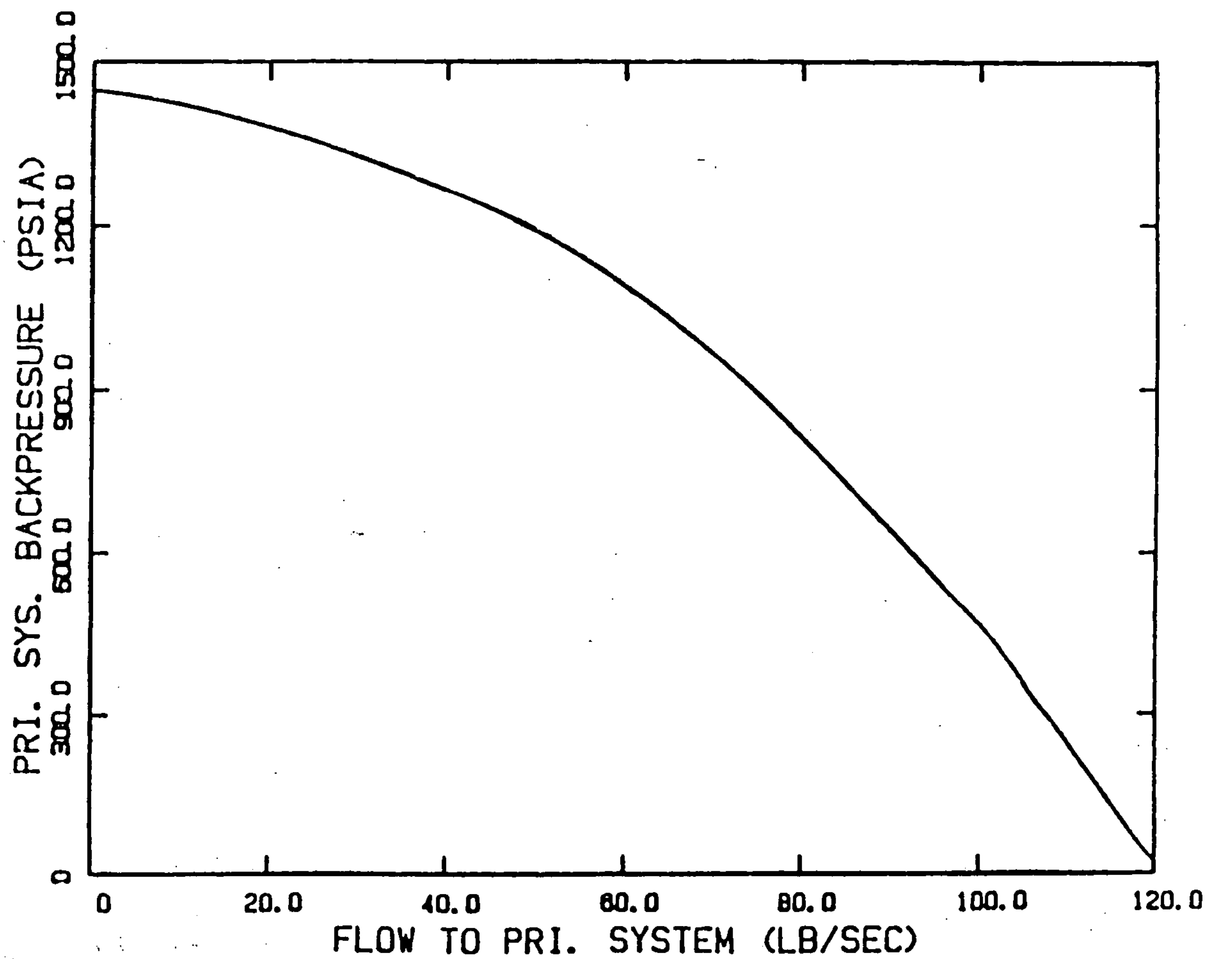
Principal Contributor: W. Lyon

Dated: July 6, 1987

REFERENCE 3



AMENDMENT 3

H. B. ROBINSON UNIT 2 Carolina Power & Light Company UPDATED FINAL SAFETY ANALYSIS REPORT	TWO PUMP SAFETY INJECTION SYSTEM DELIVERY CURVE	FIGURE 15.1.5-1

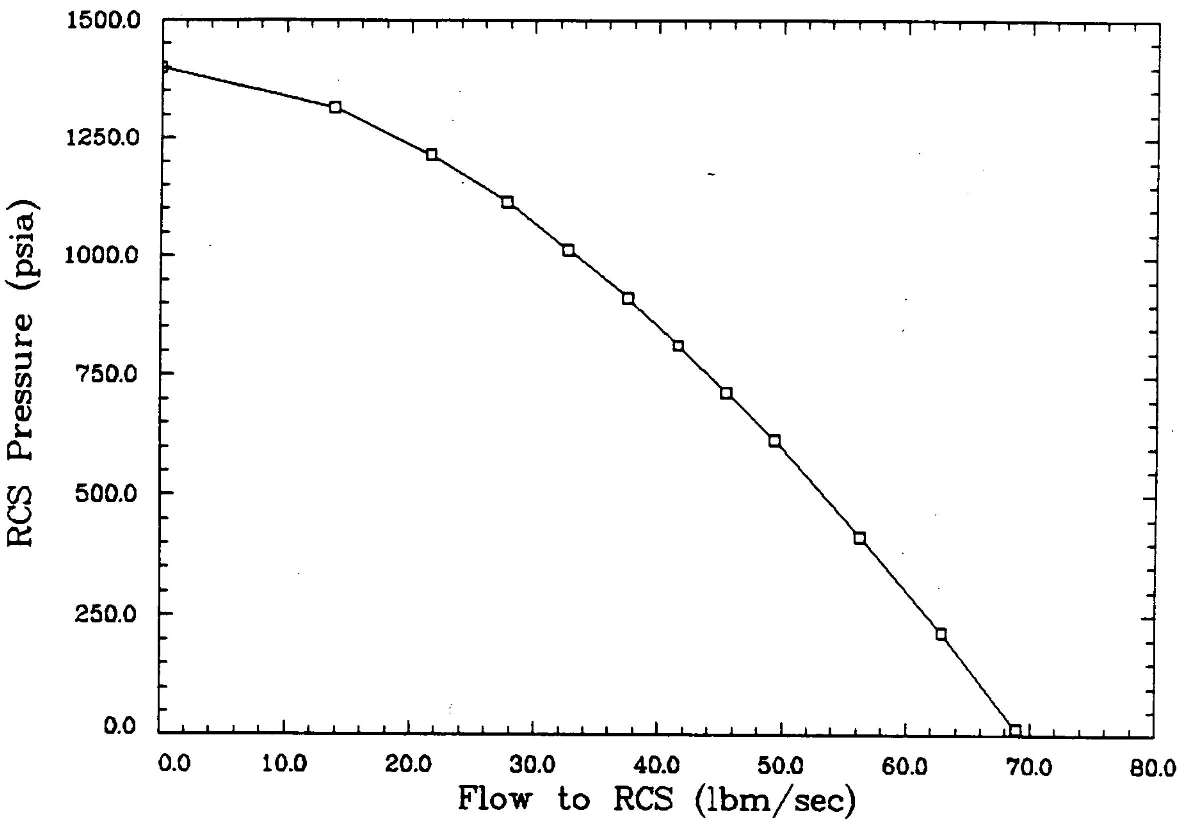


FIGURE 3.4

High Head Safety Injection Delivery Curve
Based on One Pump with No Spills

REFERENCE 4