FOR INFORMATION ONLY

R&B

CAROLINA POWER & LIGHT COMPANY

BRUNSWICK STEAM ELECTRIC PLANT

UNIT 0

PROCEDURE TYPE: REGULATORY COMPLIANCE INSTRUCTION (RCI)

NUMBER: 02.3

PROCEDURE TITLE: TECHNICAL SPECIFICATION INTERPRETATION REQUEST, PROCESSING, AND MAINTENANCE

REVISION 2

APPROVED BY: ______________________ DATE: 3/6/86
General Manager/
Director - Regulatory Compliance

8909280209 XA

LIST OF EFFECTIVE PAGES (LEP)

RCI-02.3

Pages	Revision
1-7	2

1.0 INTRODUCTION

1.1 Purpose

To provide the method by which formal interpretation of technical specification(s) shall be requested, processed, and maintained.

1.2 Scope

1.2.1 A technical specification interpretation is not intended to revise or substitute for the existing technical specifications, but to clarify the existing technical specifications. As such, a technical specification interpretation should be issued only if the existing technical specification:

1.2.1.1 Is not consistent with the physical or intended design of the plant.

1.2.1.2 Does not adequately demonstrate operability or is not sufficiently conservative.

1.2.1.3 Contains typographical or administrative errors that present operability or surveillance concerns.

1.2.1.4 Is determined necessary by the Plant Nuclear Safety Committee (PNSC).

Consequently, a technical specification interpretation is issued only if a technical specification change is required, unless determined otherwise by the Director - Regulatory Compliance.

1.3 References

1.3.1 RCI-02.1, Request for Operating License Changes

1.3.2 RCI-02.2, Operating License, Including Technical Specifications, Amendment Issuance, and Implementation

1.4 Responsibility

1.4.1 The Director - Regulatory Compliance shall ensure that this instruction is implemented to provide accurate, documented, and retrievable information.

1.4.2 Personnel shall follow this instruction to obtain formal interpretation of technical specification requirements(s).

2.0 INSTRUCTION

2.1 Inquiries requiring interpretation of technical specification purpose, intent, and/or meaning shall be made utilizing Attachment 1, Technical Specification Interpretation Request Form.

2.2 The originator shall complete Blocks 2 through 7 of the Technical Specification Interpretation Request Form and shall forward the interpretation request to his supervisor for concurrence.

2.3 The originator's supervisor and/or manager/director shall indicate concurrence with the request by signing Block 8, and shall forward the request to the Regulatory Compliance Unit for resolution.

2.4 Regulatory Compliance shall review the request for interpretation and shall:

2.4.1 Assign a technical specification interpretation serial number into Block 1 of Attachment 1, and log the interpretation.

2.4.2 Return the request to the originator, or his supervisor/manager/director, if a formal interpretation is not required (i.e., sufficient documentation exists to provide a response to the request), with a response citing the reason for interpretation rejection.

2.5 Regulatory Compliance shall investigate valid requests, provide an interpretation in Block 9 of Attachment 1, and sign Block 11. The need for a technical specification change will be indicated in Block 10.

2.6 The Director - Regulatory Compliance, or his designee, shall review the interpretation, and shall indicate his concurrence on Block 12 of Attachment 1.

2.7 Regulatory Compliance shall present the completed Technical Specification Interpretation Request Form to PNSC for approval on Block 13 of Attachment 1.

2.8 Following PNSC approval, the completed Attachment 1 shall be routed to Document Control for distribution in accordance with applicable Records Management Instructions (i.e., to holders of plant controlled copies of technical specifications, including the historical files).

2.9 Upon receipt of Attachment 1, holders of plant controlled copies of technical specifications shall insert the completed form into the "Interpretations" section of the technical specification binder, or a separate "Interpretations" binder, and shall annotate the affected technical specification page to indicate existence of the interpretation in their binder.

2.10 If a technical specification change is required by the interpretation, Regulatory Compliance shall initiate that change in accordance with RCI-02.1.

2.11 At least annually, Regulatory Compliance shall review the "Interpretations" section of the technical specifications, shall void any interpretations which are no longer applicable or have been remedied as a result of Technical Specification Amendment Issuance, and shall provide this information (in the form of a listing) to Document Control for issuance to holders of controlled copies of technical specifications, including the historical files.

2.12 Interpretations established by this procedure may be canceled or deleted in concurrence with RCI-02.2 or at the direction of PNSC.

ATTACHMENT 1

TECHNICAL SPECIFICATION INTERPRETATION REQUEST FORM

1. Serial No. ____________________

2. Technical Specification Reference: ______________________________________

3. Technical Specification Page No.: ______________________________________

4. Subject: ____

5. Unit(s) Affected: BSEP-- ______________________________________

6. Description of Request (concise detailed description of requested interpretation or problem area): ______________________________

____

____

____

____

7. Originator: ______________________________ Date: ____________

8. Reviewed By: ______________________________ Date: ____________

9. Interpretation: ____

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10) Technical Specification Change Required: Yes______ No______

11) Prepared By: ______________________________ Date____________

12) Concurrence: ______________________________ Date____________
Director - Regulatory Compliance

13) Approved: ______________________________ Date____________
PNSC

ATTACHMENT 2

TECHNICAL SPECIFICATION INTERPRETATION SERIAL LOG

Serial No.	T/S Reference	TSC Number	Date Initiated	Date Approved

Appendix 4

Brunswick Technical
Specification Interpretations

[illegible]

[illegible]

1. Serial No. 84-01 Rev 5

2. Technical Specification Reference: 3.7.1.2

3. Technical Specification Page No.: 3/4 7-2

4. Subject: SERVICE WATER SYSTEM

5. Unit(s) Affected: 1 / 2 — BOTH AND 2

6. Description of Request (concise detailed description of requested interpretation or problem area): WHAT SERVICE WATER PUMPS MUST BE OPERABLE IN ORDER T SATISFY THE REQUIREMENTS OF THIS TECHNICAL SPECIFICATION?

7. Originator: [illegible] Date: 5/4/89

8. Reviewed By: [illegible] Date: 5/4/89

9. Interpretation: ATTACHMENT 1 PROVIDES THE REQUIREMENTS FOR SERVICE WATER PUMPS AND REQUIRED ACTIONS. ATTACHMENT 2 PROVIDES THE BASIS/ANALYSIS FOR THIS INTERPRETATION.

10) Technical Specification Change Required: Yes [illegible] No ____

11) Prepared By: [illegible] / [illegible] Date 5/4/89 / 5/4/89

12) Concurrence: [illegible] Date 5/8/89
Director - Regulatory Compliance

13) Approved: [illegible] Date 5/8/89
PNSC

RECEIVED
MAY 10 1989
BNP DOC. CONTROL

PLANT SYSTEMS

SERVICE WATER SYSTEM

LIMITING CONDITION FOR OPERATION

3.7.1.2 The unit 1 and 2 service water systems shall be OPERABLE with at least:

a. Four OPERABLE nuclear service water pumps for the site capable of supplying service water to the nuclear headers; and in addition,

b. Two OPERABLE Unit 1(2) conventional service water pumps, each powered from a different division and capable of supplying both the Unit 1(2) service water nuclear header and conventional header.

Reference Information: Division Assignments of Pumps

Division I (E1, E3)	Division II (E2,E4)
1A NSW	1B NSW
2A NSW	2B NSW
1B CSW	1A CSW
2A CSW	1C CSW
2C CSW	2B CSW

APPLICABILITY: CONDITIONS 1, 2, 3, 4, and 5.

ACTION:

SBFGK

In CONDITION 1, 2, ~~or~~ 3, 4 or 5

1. With only three OPERABLE nuclear service water pumps for the site, restore at least four nuclear service water pumps to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and COLD SHUTDOWN within the following 24 hours.

[illegible]

2. With a total of only two nuclear service water pumps for the site, operation may continue provided that the two pumps are in different divisions; restore three pumps to OPERABLE status within 72 hours and restore four pumps to OPERABLE status within 7 days from the time of the initial loss or be in HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

3. With only one OPERABLE Unit 1(2) conventional service water pump, operation may continue provided that two nuclear service water pumps are OPERABLE on the unit; restore at least two conventional service water pumps powered from different divisions to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 12 hours and COLD SHUTDOWN within the following 24 hours.

4. With only two OPERABLE Unit 1(2) conventional service water pumps powered from the same division, operation may continue provided that two nuclear service water pumps are OPERABLE on the unit; restore at least two conventional service water pumps powered from different divisions to OPERABLE status within 7 days or be in HOT SHUTDOWN within the next 12 hours and COLD SHUTDOWN within the following 24 hours.

ATTACHMENT 2

TITLE: FAILURE ANALYSIS FOR TSI

OBJECTIVE: TO EVALUATE THE PLANT SERVICE WATER OPERATING CONDITIONS ALLOWED BY TSI AGAINST A FAILURE ANALYSIS. THIS EXERCISE IS BEING PERFORMED TO ENSURE ACCEPTABLE OPERATING LIMITATIONS ARE REQUIRE BY TSI AND THAT THE ALLOWED OPERATIONAL CONDITIONS RESULT IN ACCEPTABLE HYDRAULIC PERFORMANCE. ADDITIONALLY, THE OPERATIONAL CONDITIONS ARE EVALUATED TO ENSURE THE SYSTEM SAFETY FUNCTIONS AS DEFINED IN CHAPTER 15 OF THE UFSAR.

METHODOLOGY:

PSW CONFIGURATIONS ALLOWED BY TSI THAT REQUIRE A SINGLE FAILURE ANALYSIS WILL EVALUATED AGAINST INDEPENDENT FAILURES OF E1, E2, E3, & E4. CONDITIONS THAT DO NOT REQUIRE ANALYSIS OF ADDITIONAL FAILURES WILL BE

REV NO	DATE	DESCRIPTION	DRWN BY	CHK'D	RV'WD	APPR'V	CAROLINA POWER & LIGHT CO. BRUNSWICK STEAM ELECTRIC PLANT
							REF DWG
							SKETCH NO

EVALUATED TO ENSURE ADEQUATE FLOW IS AVAILABLE TO PERFORM ESSENTIAL SAFETY FUNCTIONS.

ASSUMPTIONS

1. PSW PUMPS OPERATE AT RATED FLOW OF 8000 GPM.
2. ANALYSIS IS PERFORMED FOR LOCA/LOOP ON THE AFFECTED UNIT.
3. TEST RESULTS SHOW THAT:
   INITIALLY RBCCW HX FLOW RATE WAS 4100 GPM. IT DECREASED TO 3250 GPM FOR TWO PUMP OPERATION AT MIN. SW HEADER PRESSURE. IF NORMAL RBCCW HX FLOW IS 4500 GPM, THEN THE FLOW AT MINIMUM HEADER PRESSURE WOULD BE APPROX. 3650 GPM. 3650 GPM IS USED FOR ANALYSIS.
4. RHRSW HX FLOW MAY NEED TO BE LIMITED TO 7200 GPM FOR SOME CASES. FLOWS OF 4000 TO 8000 GPM ARE ACCEPTABLE PER THE FSAR.
5. NO CREDIT IS TAKEN FOR OPERATOR ACTION BEFORE 10 MIN.

						CAROLINA POWER & LIGHT CO. BRUNSWICK STEAM ELECTRIC PLANT
REV NO	DATE	DESCRIPTION	DRWN / CHKD	RV'WD	APPRV	REF DWG. SKETCH NO.

Power Sources for Key Equipment

E3	E4	E1	E2
—	—	NSW 1A	NSW 1B
—	CSW 1A	CSW 1B	CSW 1C
RHRSW1A	RHRSW1B	RHRSW 1C	RHR-SW 1D
—	—	F002A & F068A	F002B & F068B
—	—	V106	—
—	—	V118 (N.O. xtie)	—
—	—	V111 (N.C. conv)	V117 (N.C. nuc)
—	—	V101 (N.C. conv)	V105 (N.C. nuc)
—	—	—	V102 (N.C. xtie)

NSW 2A	NSW 2B	—	—
CSW 2A	CSW 2B	CSW 2C	—
RHRSW 2A	RHRSW 2B	RHRSW 2C	RHRSW 2D
F002A & F068A	F002B & F068B	—	—
V106	—	—	—
V118 (N.O. xtie)	—	—	—
V111 (N.C. conv)	V117 (N.C. nuc)	—	—
V101 (N.C. conv)	V105 (N.C. nuc)	—	—
—	V102 (N.C. nuc/xtie)	—	—

REV. NO.	DATE	DESCRIPTION	ENG'D / ACAD	CHK'D	RV'WD	APPR'V

CAROLINA POWER & LIGHT CO.
BRUNSWICK STEAM ELECTRIC PLANT

REF. DWG.

SKETCH NO.

THIS LCO CONDITION IS [illegible].

1st REQUIREMENT

THE UNIT 1 AND 2 SERVICE WATER SYSTEMS SHALL BE OPERABLE WITH AT LEAST:

a. FOUR OPERABLE NSW PUMPS FOR THE SITE CAPABLE OF SUPPLYING SERVICE WATER TO THE NUCLEAR HEADERS; AND IN ADDITION,

b. TWO OPERABLE UNIT 1(2) CSW PUMPS, EACH POWERED FROM A DIFFERENT DIVISION AND CAPABLE OF SUPPLYING BOTH THE UNIT 1(2) SERVICE WATER NUCLEAR HEADER AND CONVENTIONAL HEADER.

DISCUSSION

° CONSIDER LOSS OF E1:

AVAILABLE EQUIPMENT:	UNIT 1	UNIT 2
	NSW 1B	NSW 2A & 2B
	CSW 1C	CSW 2A & 2B
	RHRSW LOOP B	RHRSW 2A, 2B & 2D
	NUC HDR → VITAL HDR	EITHER HDR → VITAL HDR
	NUC HDR → RHRSW LOOP B	EITHER HDR → RHRSW HDR
	V106 FAILS OPEN	

LOAD	UNIT 1 <10 MIN NUC	UNIT 1 >10 MIN NUC/CONV	UNIT 2 <10 MIN NUC	UNIT 2 >10 MIN NUC/CONV
D/G COOLING	2000	2000/—	2000	2000/—
VITAL HEADER	800	800/—	800	800/—
RBCCW HX	3650	3650/—	—	—/—
RHR-SW	—	8000/—	—	—/8000
LUBE WATER	200	200/—	200	200/—
TOTAL DEMAND	6650	14650/—	3000	3000/8000
# PUMPS REQ	1	2/—	1	1/1
# PUMPS AVAIL	1	2/—	2	2/2 PUMPS

REV NO	DATE	DESCRIPTION	DRWN BY / CHK'D	RV'WD	APPRV	CAROLINA POWER & LIGHT CO. BRUNSWICK STEAM ELECTRIC PLANT
						REF. DWG
						SKETCH NO

• CONSIDER LOSS OF E2:

AVAILABLE EQUIPMENT:

Unit 1	Unit 2
NSW 1B	NSW 2A & 2B
CSW 1B	CSW 2A & 2B
RHR SW - LOOP A	RHRSW 2A, 2B &
CONV. HDR → VITAL HDR.	EITHER HDR → VITAL HDR
CONV HDR → RHRSW LOOP A	EITHER HDR → RHRSW

LOAD	Unit 1 <10 min NUC	Unit 1 >10 min NUC/CONV	Unit 2 <10 min NUC	Unit 2 >10 min NUC/CONV
D/G COOLING	2000	2000/—	2000	2000/—
VITAL HEADER	—	—/800	800	800/—
RBCCW Hx	—	—/—	—	—/—
RHRSW	—	—/8000	—	—/8000
LUBE WATER	200	200/—	200	200/—
TOTAL DEMAND	2200	2200/8800 (NOTE 1)	3000	3000/8000
# PUMPS RQD.	1	1/1	1	1/1
# PUMPS AVAIL.	1	1/1	2	2/2

CAROLINA POWER & LIGHT CO.
BRUNSWICK STEAM ELECTRIC PLANT

REV NO	DATE	DESCRIPTION	CHK'D	PVWD	APPRV

REF DH3
SKETCH NO.

...SIDER LOSS OF E3:

...LABLE
...IPMENT:

Unit 1	Unit 2
NSW 1A & 1B	NSW 2B
CSW 1B & 1C	CSW 2B & 2C
RHRSW 1B, 1C, & 1D	RHRSW Loop B
Either Hdr → Vital Hdr	Nuc Hdr → Vital Hdr
Either Hdr - RHRSW	Nuc Hdr → RHRSW-B
	V106 Fails Open

	Unit 1		Unit 2	
...AND	<10 min Nuc.	>10 min Nuc/Conv	<10 min Nuc	>10 min Nuc/Conv
...LING	2000	2000/—	2000	2000/—
...EADER	800	800/—	800	800/—
...D Hx	—	—/—	3650	3650/—
...N	—	—/8000	—	8000/—
Water	200	200/—	200	200/—
Demand	3000	3000/8000	6650	14650/—
Pumps Rqd	1	1/1	1	2/—
...umps Avail	2	2/2	1	3/—

Page 6 of 13

				CAROLINA POWER & LIGHT CO. BRUNSWICK STEAM ELECTRIC PLANT
DESCRIPTION	INITIATED BY / CHK'D	RV'WD	APP'V	REF DWG SKETCH NO

- Consider Loss of E3:

Available Equipment:

Unit 1	Unit 2
NSW 1A & 1B	NSW 2B
CSW 1B & 1C	CSW 2B & 2C
RHRSW 1B, 1C, & 1D	RHRSW Loop B
Either Hdr → Vital Hdr	Nuc Hdr → Vital Hdr
Either Hdr - RHRSW	Nuc Hdr → RHRSW - V106 Fails Open

	Unit 1		Unit 2	
Load	<10 min Nuc.	>10 min Nuc/Conv	<10 min Nuc	>10 min Nuc/Conv
D/G Cooling	2000	2000/—	2000	2000/—
Vital Header	800	800/—	800	800/—
RBCCW Hx	—	—/—	3650	3650/—
RHRSW	—	—/8000	—	8000/—
Lube Water	200	200/—	200	200/—
Total Demand	3000	3000/8000	6650	14650/—
# Pumps Req'd	1	1/1	1	2/—
# Pumps Avail	2	2/2	1	3/—

DATE	DESCRIPTION	DRWN BY	CHK'D	RV'WD	APPRV	CAROLINA POWER & LIGHT CO. BRUNSWICK STEAM ELECTRIC PLANT
						REF DWG
						SKETCH NO

FORM NO. 2103 3/84

o CONSIDER LOSS OF E4:

AVAILABLE EQUIPMENT:	UNIT 1	UNIT 2
	NSW 1A & 1B	NSW 2A
	CSW 1B & 1C	CSW 2A *
	RHRSW 1A, 1C, & 1D	RHRSW - A LOOP
	EITHER HDR → VITAL HDR	CONV HDR → VITAL HDR
	EITHER HDR → RHRSW	CONV HDR → RHRSW A

* CSW 2C IS ALSO AVAILABLE BUT CSW 2A & 2C ARE BOTH DIV I

LOAD	UNIT 1 <10 min Nuc.	UNIT 1 >10 min Nuc/Conv	UNIT 2 <10 min Nuc	UNIT 2 >10 min Nuc/Conv
D/G COOLING	2000	2000/—	2000	2000/—
VITAL HEADER	800	800/—	—	—/800
RBCCW Hx	—	—/—	—	—/—
RHRSW	—	—/8000	—	—/8000
LUBE WATER	200	200/—	200	200/—
TOTAL DEMAND	3000	3000/8000	2200	2200/8800 (NOTE 1)
# PUMPS REQ	1	1/1	1	1/1
# PUMPS AVAIL.	2	2/2	1	1/1

CONCLUSION: WITH THE PLANTS IN THE PSW CONFIGURATION REQUIRED BY THE FIRST REQUIREMENT, IT HAS BEEN SHOWN THAT THE PUMP REQUIREMENTS ARE ACCEPTABLE FOR PSW TO COMPLETE ITS SAFETY FUNCTION WITH A SINGLE FAILURE OF E1, E2, E3, OR E4. [illegible]

REV NO.	DATE	DESCRIPTION	DRWN BY / CHK'D	RV'WD	APPRV	CAROLINA POWER & LIGHT CO. BRUNSWICK STEAM ELECTRIC PLANT
						REF DWG.
						SKETCH NO.

2ND REQUIREMENT

IN CONDITION 1, 2, OR 3:

WITH ONLY THREE OPERABLE NUCLEAR SERVICE WATER PUMPS FOR THE SITE, RESTORE AT LEAST FOUR NUCLEAR SERVICE WATER PUMPS TO OPERABLE STATUS WITHIN 7 DAYS OR BE IN AT LEAST HOT SHUTDOWN WITHIN THE NEXT 12 HOURS AND COLD SHUTDOWN WITHIN THE FOLLOWING 24 HOURS.

DISCUSSION

THE ANALYSIS FOR THIS SCENARIO IS THE SAME AS ANALYZED FOR THE FIRST REQUIREMENT. IN THE CALCS FOR LOSS OF EACH OF E1, E2, E3, AND E4, THERE WERE THREE OPERABLE NSW PUMPS. AN ADDITIONAL SINGLE FAILURE IS NOT CONSIDERED.

CONCLUSION

WITH THE PLANT PLACED IN THE PSW CONFIGURATION OF THE 2ND REQUIREMENT IT IS SHOWN THAT THE PUMP REQUIREMENTS ARE ACCEPTABLE FOR THE SYSTEM TO PERFORM ITS SAFETY FUNCTION.

3RD REQUIREMENT

IN CONDITION 1, 2, OR 3:

WITH A TOTAL OF ONLY TWO NUCLEAR SERVICE WATER

						CAROLINA POWER & LIGHT CO. BRUNSWICK STEAM ELECTRIC PLANT
REV NO	DATE	DESCRIPTION	DRWN BY / CHK'D BY	RV'WD	APP'RV	REF DWG SKETCH NO.

PUMPS FOR THE SITE, OPERATION MAY CONTINUE PROVIDED THAT THE TWO PUMPS ARE IN DIFFERENT DIVISIONS. RESTORE THREE PUMPS TO OPERABLE STATUS WITHIN 72 HOURS AND RESTORE FOUR PUMPS TO OPERABLE STATUS WITHIN 7 DAYS FROM THE TIME OF THE INITIAL LOSS OR BE IN HOT SHUTDOWN WITHIN 12 HOURS AND IN COLD SHUTDOWN WITHIN THE FOLLOWING 24 HOURS.

DISCUSSION

THE WORST CASE SITUATION FOR THIS REQUIREMENT IS THAT ONE UNIT HAS NO OPERABLE NSW PUMPS AND THE OTHER UNIT HAS TWO. IT IS EXPECTED THAT THE UNIT WITH NO NSW PUMPS WILL HAVE ONE CSW PUMP ALIGNED TO THE NUCLEAR HEADER. THE FOLLOWING ANALYSIS ASSUMES UNIT 1 HAS THE TWO INOPERABLE NSW PUMPS BUT THE RESULTS ARE THE SAME IF UNIT TWO HAD THE INOPERABLE NSW PUMPS INSTEAD.

LOAD	UNIT 1		UNIT 2	
	<10 min NUC	>10 min. NUC/CONV	<10 min NUC	>10 min NUC/CONV
D/G COOLING	2000	2000/—	2000	2000/—
VITAL HEADER	800	800/—	800	800/—
RBCCW	—	—/—	—	—/—
RHRSW	—	—/8000	—	—/8000
LUBE WATER	200	200/—	200	200/—
TOTAL DEMAND	3000	3000/8000	3000	3000/8000
# PUMPS REQ.	1	1/1	1	1/1
# PUMPS AVAIL.	1	1/2	2	2/3

REV NO	DATE	DESCRIPTION	DRWN BY / CHK'D	RV'WD	APPRV	CAROLINA POWER & LIGHT CO. BRUNSWICK STEAM ELECTRIC PLANT
						REF. DWG.
						SKETCH NO.

* UTILIZES A LOOP OF RHR SW

CONCLUSION

IT HAS BEEN SHOWN BY WORST CASE ANALYSIS THAT IN THIS LIMITED CONFIGURATION, PSW CAN PERFORM ITS SAFETY FUNCTION.

4th REQUIREMENT

IN CONDITION 1, 2, OR 3.

WITH ONLY ONE OPERABLE UNIT 1(2) CONVENTIONAL SERVICE WATER PUMP, OPERATION MAY CONTINUE PROVIDED THAT TWO NUCLEAR SERVICE WATER PUMPS ARE OPERABLE ON THE UNIT; RESTORE AT LEAST TWO CONVENTIONAL SERVICE WATER PUMPS POWERED FROM DIFFERENT DIVISIONS TO OPERABLE STATUS WITHIN 7 DAYS OR BE IN HOT SHUTDOWN WITHIN THE NEXT 12 HOURS AND COLD SHUTDOWN WITHIN THE FOLLOWING 24 HOURS.

DISCUSSION

THE FOLLOWING ANALYSIS ASSUMES UNIT 1 HAS THE INOPERABLE CSW PUMPS BUT THE RESULTS ARE THE SAME IF UNIT 2 HAD THE INOPERABE CSW PUMPS INSTEAD.

REV NO	DATE	DESCRIPTION	DRWN BY / CHK'D	RV'WD	APPRV	CAROLINA POWER & LIGHT CO. BRUNSWICK STEAM ELECTRIC PLANT
						REF. DWG.
						SKETCH NO.

LOAD	UNIT 1 <10 MIN NUC	UNIT 1 >10 MIN NUC/CONV	UNIT 2 <10 MIN NUC	UNIT 2 >10 MIN NUC/CONV
D/G COOLING	2000	2000/—	2000	2000/—
VITAL HEADER	800	800/—	800	800/—
RBCCW	—	—/—	—	—/—
RHRSW	—	8000/—	—	8000/—
LUBE WATER	200	200/—	200	200/—
TOTAL DEMAND	3000	*11,000/—	3000	*11,000/—
# PUMPS REQ.	1	2/0	1	2/0
# PUMPS AVAIL	2	2/1	2	2/3

* UTILIZES B LOOP OF RHRSW

CONCLUSION

IT HAS BEEN SHOWN BY ANALYSIS THAT IN THIS LIMITED CONFIGURATION, PSW CAN PERFORM ITS SAFETY FUNCTION.

5TH REQUIREMENT

IN CONDITION 1, 2, OR 3 ...

WITH ONLY TWO OPERABLE UNIT 1(2) CONVENTIONAL SERVICE WATER PUMPS POWERED FROM THE SAME DIVISION, OPERATION MAY CONTINUE PROVIDED THAT TWO NUCLEAR SERVICE WATER PUMPS ARE OPERABLE ON THE UNIT; RESTORE AT LEAST TWO CONVENTIONAL SERVICE WATER PUMPS

REV NO	DATE	DESCRIPTION	DRWN BY / CHK'D	RV'WD	APPRV	CAROLINA POWER & LIGHT CO. BRUNSWICK STEAM ELECTRIC PLANT
						REF. DWG.
						SKETCH NO.

POWERED FROM DIFFERENT DIVISIONS TO OPERABLE STATUS WITHIN 7 DAYS OR BE IN HOT SHUTDOWN WITHIN THE NEXT 12 HOURS AND COLD SHUTDOWN WITHIN THE FOLLOWING 24 HOURS

DISCUSSION

THE ANALYSIS FOR THIS REQUIREMENT IS ENVELOPED BY THE ANALYSIS FOR REQUIREMENT #4 DUE TO THE FACT THAT THERE ARE MORE CSW PUMPS AVAILABLE, THE SAME NUMBER OF NSW PUMPS AVAILABLE AND NO ADDITIONAL FAILURES ARE ASSUMED.

CONCLUSION

IT IS CONCLUDED THAT IN THIS CONFIGURATION, PSW IS CAPABLE OF PERFORMING ITS SAFETY FUNCTION

FORM NO. 2103 3/84

NOTES

1. CROSS-TIE VALVE LEAKAGE HAS BEEN EVALUATED AT APPROXIMATELY 1900 GPM. THE CROSS-TIE LEAKAGE IS FROM THE NUCLEAR TO THE CONVENTIONAL HEADER THEREFOR A DEMAND OF 8800 GPM FOR ONE AVAILABLE PUMP IS NOT A CONCERN.

REV NO	DATE	DESCRIPTION	DRWN BY / CHK'D	RV'WD	APPRV	CAROLINA POWER & LIGHT CO. BRUNSWICK STEAM ELECTRIC PLANT
						REF. DWG.
						SKETCH NO.

ATTACHMENT 1

TECHNICAL SPECIFICATION INTERPRETATION REQUEST FORM

8) Serial No. 84-08 (Rev. 1)

1) Technical Specification Reference: Table 3.3.3-2 Item 1d

2) Technical Specification Page No.: 3/4 3-34

3) Subject: Core Spray Time Relay

4) Unit(s) Affected: BSEP-- Both

5) Description of Request (concise detailed description of requested interpretation or problem area): Item 1d of Table 3.3.3-2 indicates that the trip setpoint for the E21-K16A, B time delay relay is 14 ≤t ≤16 sec. This relay is actually a 5 sec. relay. What is the correct value?

6) Originator: Tony Gick Date: 5/15/84

7) Reviewed By: Bob Poulk, Jr. Date: 5/15/84

9) Interpretation: The correct value for E21-K16A, B is 4.5≤t≤5.5. The K16 works in series with the STR 1A2, 1B2 (2.A2, 2.B2) which is a 10 second time relay to provide the 15 sec. delay for core spray. This is in accordance with plant design (PMs 79-185 and 79-186). A T/S change to reflect this has been initiated.

10) Technical Specification Change Required: Yes x (84TSB30) No

11) Prepared By: [signature] Date [illegible]

12) Concurrence: [signature] Date 5/24/85
Director - Regulatory Compliance

13) Approved: [signature] Date 5/24/85
PNSC

ATTACHMENT 1

TECHNICAL SPECIFICATION INTERPRETATION REQUEST FORM

1. Serial No. 84-10, Rev. 6

2. Technical Specification Reference: Figure 3.4.6.1-2, 1-1, 1-3

3. Technical Specification Page No.: 3/4 4-15, 4-14, 4-16

4. Subject: Pressure/Temperature Limits for RPV

5. Unit(s) Affected: BSEP— 1 & 2

6. Description of Request (concise detailed description of requested interpretation or problem area): ~~See attached~~ PLG Have the present RPV Pressure/temperature curves in T/S Figures 3.4.6.1-1, 1-2, and 1-3 been revised to reflect NEDO-24161 and NEDO-24157 referenced by the FSAR? NEDO-24161 and NEDO 24157 specify a re-evaluated end of life neutron fluence.

7. Originator: Phillip Gore Date: 6-26-89

8. Reviewed By: Stephen J. Friend Date: 6/26/89

9. Interpretation: See attached.

RECEIVED

JUN 30 1989

BNP DOC. CONTROL

10) Technical Specification Change Required: Yes BSEP TSB 4 No

11) Prepared By: Phillip Gore/ [illegible] Date 6-26-89 / 6/27/89

12) Concurrence: [illegible] Date 6/28/89

Director - Regulatory Compliance

13) Approved: [illegible] Date 6-29-89

PNSC

The existing T/S curves have not been revised to reflect NEDO-24161 and NEDO-24157, and are not considered sufficiently conservative. A T/S change has been submitted and considers the latest revisions to 10CFR50, Appendices G and H. Pending issuance of the new amendment, this TSI provides conservative curves to be followed.

The attached curves have been calculated using the latest pertinent provisions of 10CFR50, Appendices G and H, ASTM E-185 and the trend equation curve of RG 1.99, Rev. 2. The attached curves have been adjusted to ensure complete conservatism with regards to the existing T/S curves.

Revision 1 to this TSI was issued with the hydrostatic and leak test curves calculated for less than or equal to 4 effective full power years (EFPY). Both units 1 and 2 have surpassed 4 EFPY. Revision 2 to this TSI provided hydrostatic test curves for less than or equal to 5 EFPY. Additionally, revision 2 provided separate pressure/temperature curves for Unit 1 and 2 because Unit 1 has a smaller irradiated RTNDT than Unit 2 as calculated by the trend curve equation.

Revision 3 to this TSI was issued to revise the Unit 2 Hydrostatic and Leak Test curve. The curve was applicable up to 4.75 EFPY vice the then existing curve which was good to 5 EFPY. This revision allowed the hydro to be performed at lower temperatures.

Revision 4 to this TSI is being issued to implement the latest operating curves. The Unit 1 and Unit 2 hydrostatic/leak test curves currently in revision 3 are no longer applicable in that Unit 1 has surpassed 5 EFPY and Unit 2 has surpassed 4.75 EFPY. The curves being implemented by this TSI have been adjusted to ensure complete conservatism with regards to the existing T/S curves.

Revision 5 to this TSI is being issued to implement the latest Unit 1 pressure/temperature curve for Normal Operation With Core Critical (Figure 3.4.6.1-2). This pressure-temperature curve has been revised using the latest NRC guidelines along with actual neutron flux/fluence data.

Revision 6 of this TSI is being issued to revise and clarify the criticality curve for Unit 1 that was transmitted via revision 5. The original criticality curve for Unit 1 ends at approximately 153 degrees on the x-axis and 460 psig on the y-axis (Ref. Figure 3.4.6.1-2 of Tech. Specs.). In revision 5 of this TSI it was determined that the area below the above x-y coordinates was an undefined area. The lower portion of the curve was therefore plotted using the latest NRC guidelines. Regulatory Compliance has determined that the original Unit 1 curve should be drawn straight down from the above listed coordinates. This revision revises the curve issued via revision 5 by removing the cross-hatched area and by adding the lower portion of the curve. The

6

limit has been set at 170^{0} F instead of 153^{0} F to account for neutron embrittlement. This revision will suffice for the interim until the permanent Technical Specification Change Request, which includes the cross-hatched area, is approved by the NRC. 6

ATTACHMENT 1

NUCLEAR SAFETY EVALUATION CHECKLIST

Identification and description of Item Being Evaluated: TSI 84-10, Rev. 6

Circle One

1. Does this item represent a change to the facility as described in the FSAR? Yes No

   If yes, describe the change and the FSAR section(s) involved.

2. Does this item represent a change to procedures as described in the FSAR? Yes No

   If yes, describe the change and the FSAR section(s) involved.

3. Does this item represent a test or experiment not described in the FSAR? Yes No

   If yes, describe the test or experiment.

4. Does this item require a revision to the FSAR? (If yes, submit appropriate change(s) per RCI-04.1, as applicable) Yes No

5. Does the proposed change, test or experiment require a change to the technical specifications? (If yes, submit appropriate change(s) per RCI-02.1) Yes* No

*Technical Specification Change 88TSB04 has been submitted.

Circle One

6. The change, test, or experiment shall be tested against the following criteria:

6.1 Will the probability of occurrence of any accident previously evaluated in FSAR (Chapter 15) be increased? Yes No Uncertain

BASIS* See attached.

6.2 Will the consequences of any accident previously evaluated in the FSAR (Chapter 15) be increased? Yes No Uncertain

BASIS* See attached.

6 3 Will the probability of occurrecne of malfunction equipment important to safety previously evaluated in the FSAR be increased? Yes No Uncertain

BASIS* See attached.

6.4 Will the consequences of malfunction of equipment important to safety previously evaluated in the FSAR be increased? Yes No Uncertain

BASIS* See attached.

6.5 Will the probability of an accident or possibility for malfunction of equipment important to safety of a different type than already evaluated in FSAR be created? Yes No Uncertain

BASIS* See attached.

*BASIS REQUIRED - UTILIZE ADDITIONAL SHEETS AS NECESSARY

ATTACHMENT 1 (Cont'd)

Circle One

6.6 Will the margin of safety as defined in the basis to any technical specifications be reduced? Yes **No** Uncertain

BASIS* See attached.

7. If any response to Question 6 is yes, it is to be assumed that the proposed change or test or experiment constitutes an unreviewed safety quistion within the meaning of 10CFR50.59. Based on this determination, does the subject change or test or experiment constitute an unreviewed safety question? Yes **No**

8. Will the change, test, or experiment have a significant adverse effect on the environment? Yes **No**

9. Will the change, test or experiment raise a potential safety concern on the unit to which it applies, <u>or to the other unit</u>? Yes **No**

10. Does the change, test, or experiment affect or bring about a change to other plant procedures? Yes **No**

NOTE: A copy of this Safety Evaluation shall accompany the package through review.

11. Indicate the sections of the FSAR researched to confirm the determinations made in Items 1, 2, 3, 4, 6, and 8 above.
5.3.2, Chapter 15.

12. Indicate the sections of the technical specifications researched to confirm the determinations, as applicable, made in Items 5 and 6 above.
3/4.4.6, Figures 3.4.6.1-1, 3.4.6.1-2, 3.4.6.1-3, and table 4.4.6.1.3-1.

Prepared by: Phillip Hoe Date: 6-26-89

Title: SR Mech. Specialist

PNSC Approval: ________ Date: ________
(if required)

*BASIS REQUIRED - UTILIZE ADDITIONAL SHEETS AS NECESSARY

SAFETY EVALUATION

6.1 The pressure-temperature curve being implemented by this Technical Specification Interpretation is based on the latest NRC guidelines relative to the RPV pressure-temperature curves (Revision 2 of Reg. Guide 1.99, 10CFR Appendix G, and Appendix G of ASME Section III) along with actual neutron flux/fluence data. Reactor coolant system temperature and pressure are currently utilized to comply with the requirements of TS Section 3/4.4.6 and have been evaluated to confirm that they are representative of the vessel shell temperature and vessel pressure. The reactor coolant system temperature, measured at the recirculation pump suction, is actually lower than that of the vessel shell during various phases of operation (i.e., reactor startup, operation, and immediately following reactor shutdown) because of the effects of gamma heating of the reactor vessel. Therefore, use of recirculation pump suction temperature is more conservative during these operational phases. Since the coolant system data is representative of the vessel shell temperature, the probability of a pressure boundary failure will remain the same and will provide the same limitations of the consequences of a pressure boundary failure. It is therefore concluded that the probability of occurrence of any accident previously evaluated in chapter 15 of the FSAR will not be increased.

6.2 The accidents analyzed in Chapter 15 of the Updated FSAR are not affected by the revised pressure-temperature curve. The curve is designed to provide fracture protection for the reactor vessel coolant boundary. The consequences of a pressure boundary failure are not impacted by this change. Since the curve is based on the most current regulatory guidance and fluence data, it can be concluded that there is not a significant increase in the consequences of an accident previously evaluated.

6.3 The pressure-temperature curve being implemented by this TSI does not change any of the postulated accident scenarios or the accident initiators. Additionally, this change does not adversely affect the operability of any safety related equipment. Therefore, the probability of occurrence of malfunction of equipment important to safety will not be increased.

6.4 As stated above, this change does not alter any of the postulated accident scenarios or accident initiators. Nor does it adversely affect the operability of any safety related equipment. Therefore, the consequences of malfunction of equipment important to safety previously evaluated in the FSAR will not be increased.

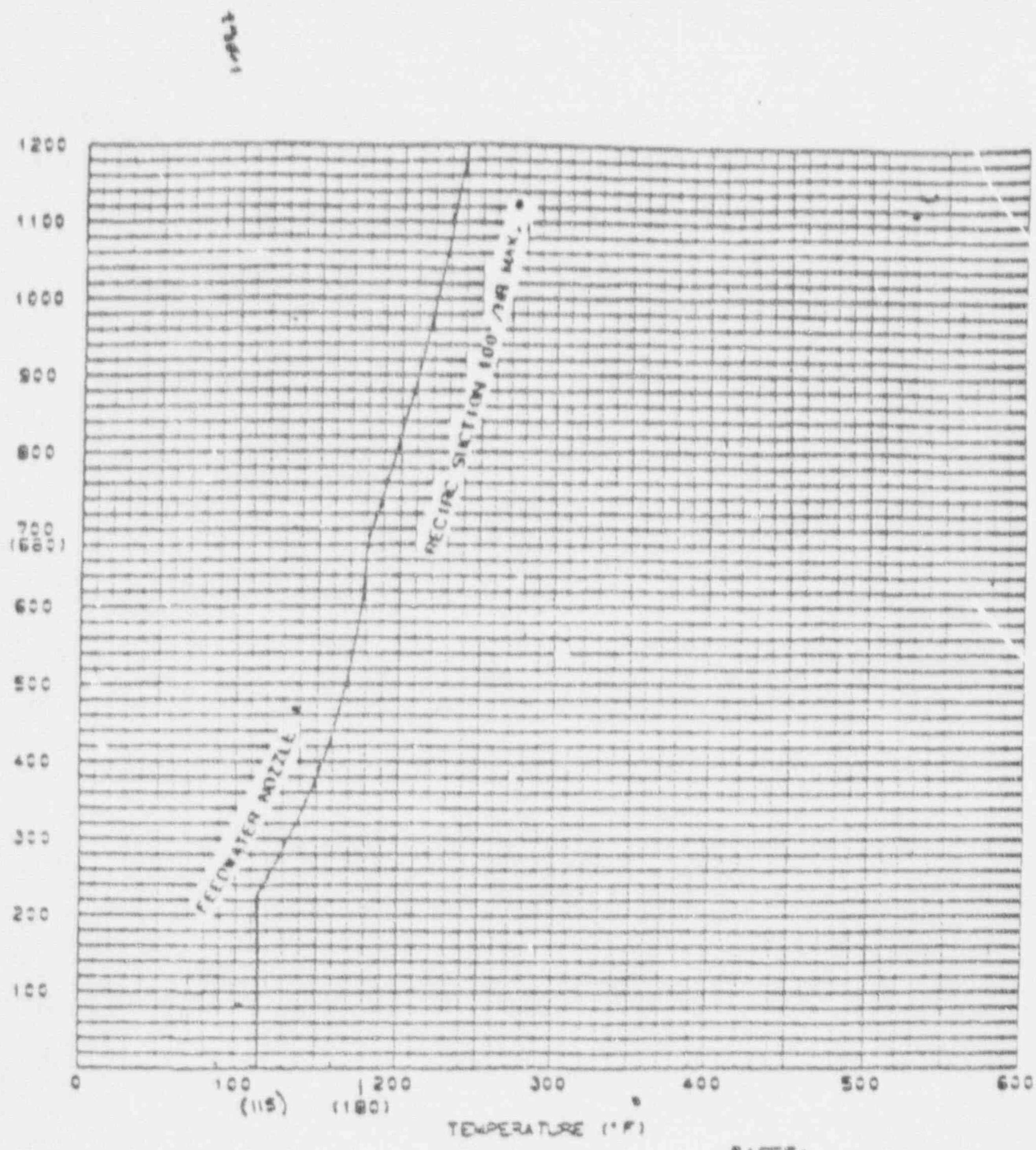
6.5 The accidents analyzed in chapter 15 of the Updated FSAR are not affected by the revised pressure-temperature curve. This curve is designed to provide fracture protection for the reactor coolant pressure boundary and does not create any new accident mode. Accident modes for the reactor coolant pressure boundary, due to nonductile failure are well understood within the industry. The pressure-temperature curve merely provides the protection mechanisms to preclude such a failure. Therefore the probability of an accident or

possibility for malfunction of equipment important to safety of a different type than already evaluated in the FSAR will not be created.

6.6 Pressure-temperature curves are designed to provide a specific margin of safety. This margin is required to be at least as great as that specified in the ASME Boiler and Pressure Vessel Code, Section III, Appendix G, and Appendix G to 10CFR50. The revised curve is based on the latest NRC guidelines (Regulatory Guide 1.99, Rev. 2). along with actual neutron flux/fluence data for the Brunswick Units. Thus the curves provide a greater confidence level than the present curves. It can therefore be concluded that the margin of safety as defined in the basis to any technical specification will not be reduced.

PRESSURE-TEMPERATURE LIMITS
REACTOR VESSEL
BSEP UNIT NO. 1

NORMAL OPERATION WITH CORE NOT CRITICAL



BASES:

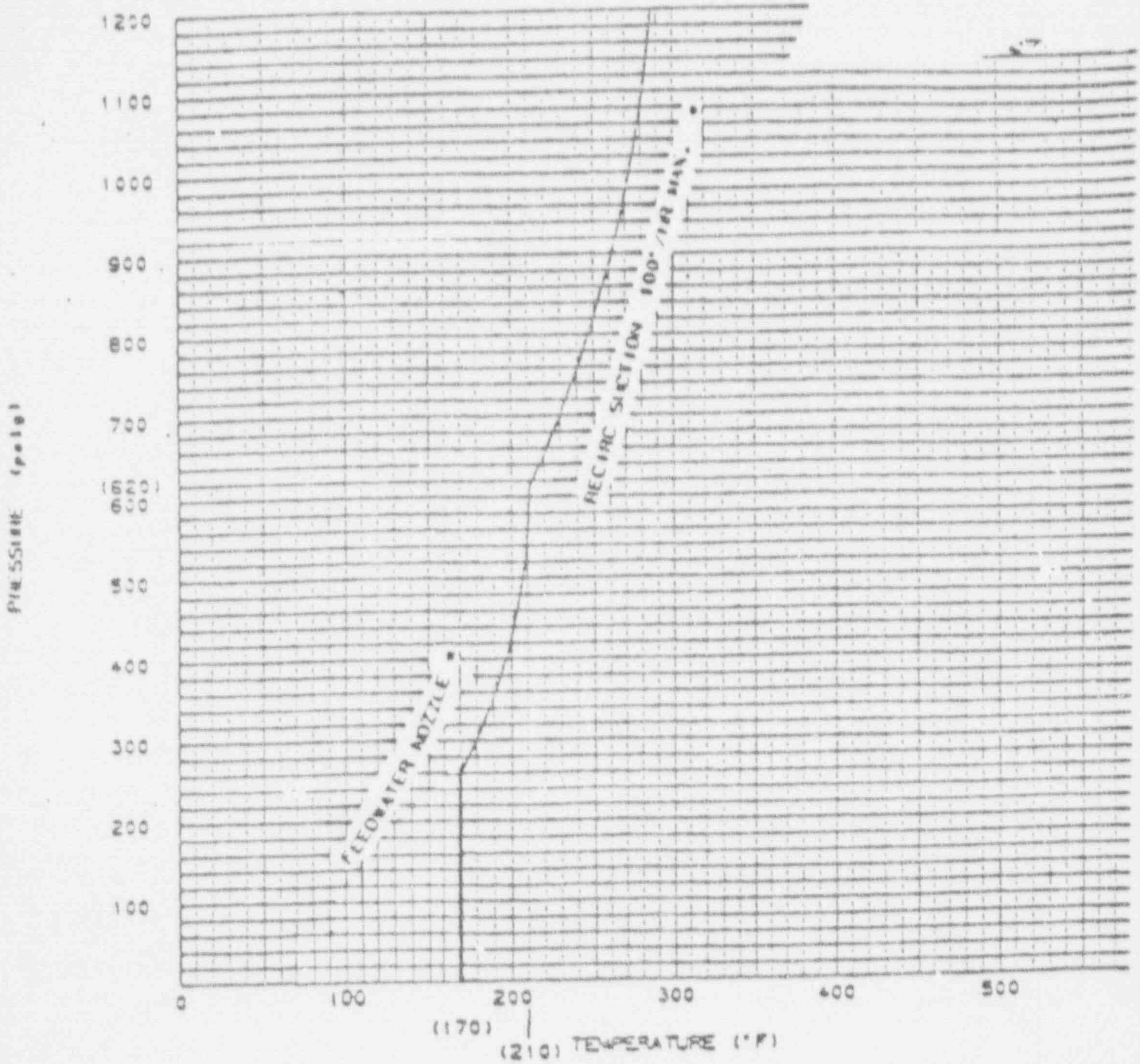
1. FUEL IN REACTOR
2. ≤ 16 EFPY
3. 7.1×10^{17} N/CM² > 1 MEV
4. RT_{NDT} = 81.6° (1/4 T)
5. 15 PSI INSTRUMENT LOCATION CORRECTION INCLUDED
6. PROPOSED REG. GUIDE 1.99 REV.2

FIGURE 3.4.6.1-1

OPERATE TO RIGHT AND/OR BELOW LIMITING LINES
* INDICATES BOTH HEATUP AND COOLDOWN RATE
PRESSURE AND TEMPERATURE INTERSECTIONS NOTED BY PARENTHESES

FIGURE 3.4.6.1-2
PRESSURE-TEMPERATURE LIMITS
REACTOR VESSEL

NORMAL OPERATION WITH CORE CRITICAL



BASES.

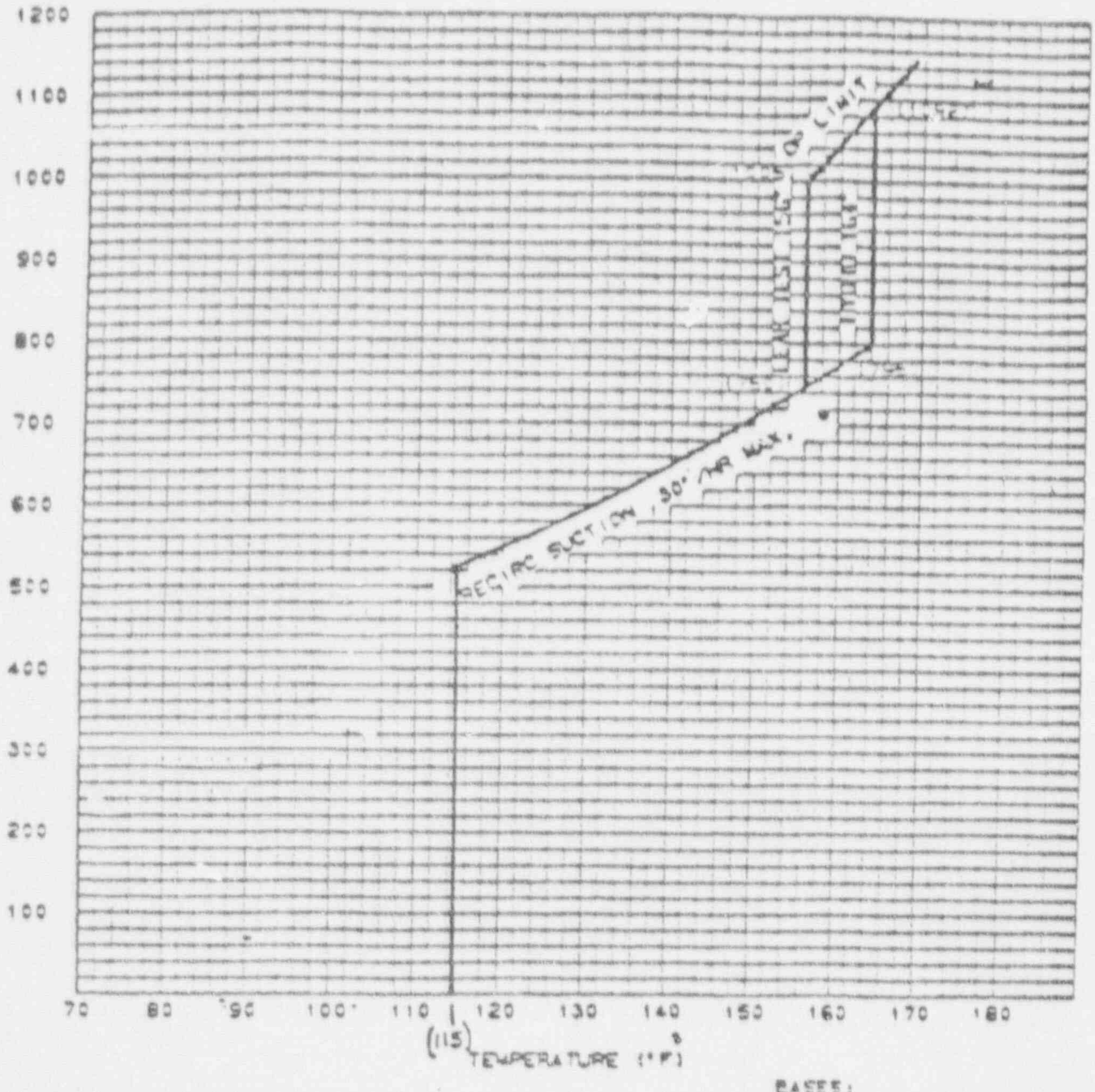
1. FUEL IN REACTOR
2. ≤ 16 EFPY,
3. 7.1×10^{17} n/cm^2 > 1 MEV
4. RT_{NDT} = 81.4° (1/4 T)
5. 13 PSI INSTRUMENT LOCATION CORRECTION INCLUDED
6. REG. GUIDE 1.99 REV. 2

NOTES.

1. OPERATE TO RIGHT AND/OR BELOW LIMITING LINES
2. * INDICATES BOTH HEATUP AND COOLDOWN RATE
3. PRESSURE AND TEMPERATURE INTERSECTIONS NOTED BY PARENTHESES
4. ~~OPERATION IN CROSS-HATCHED AREA PERMITTED ONLY WHEN WATER LEVEL IS WITHIN NORMAL RANGE FOR POWER OPERATION~~ 6/29/87

PRESSURE-TEMPERATURE LIMITS
REACTOR VESSEL
BSEP UNIT NO. 1

HYDROSTATIC AND LEAK TESTS



BASES:

1. FUEL IN REACTOR
2. REACTOR NOT CRITICAL
3. PROPOSED REG. GUIDE 1.99, REV. 2
4. 8 EFPY
5. 3.5 x 10^{17} n/cm² > 1 MEV
6. RT_{NDT} = 56° (1/4T)
7. 15 PSI INSTRUMENT LOCATION CORRECTION INCLUDED

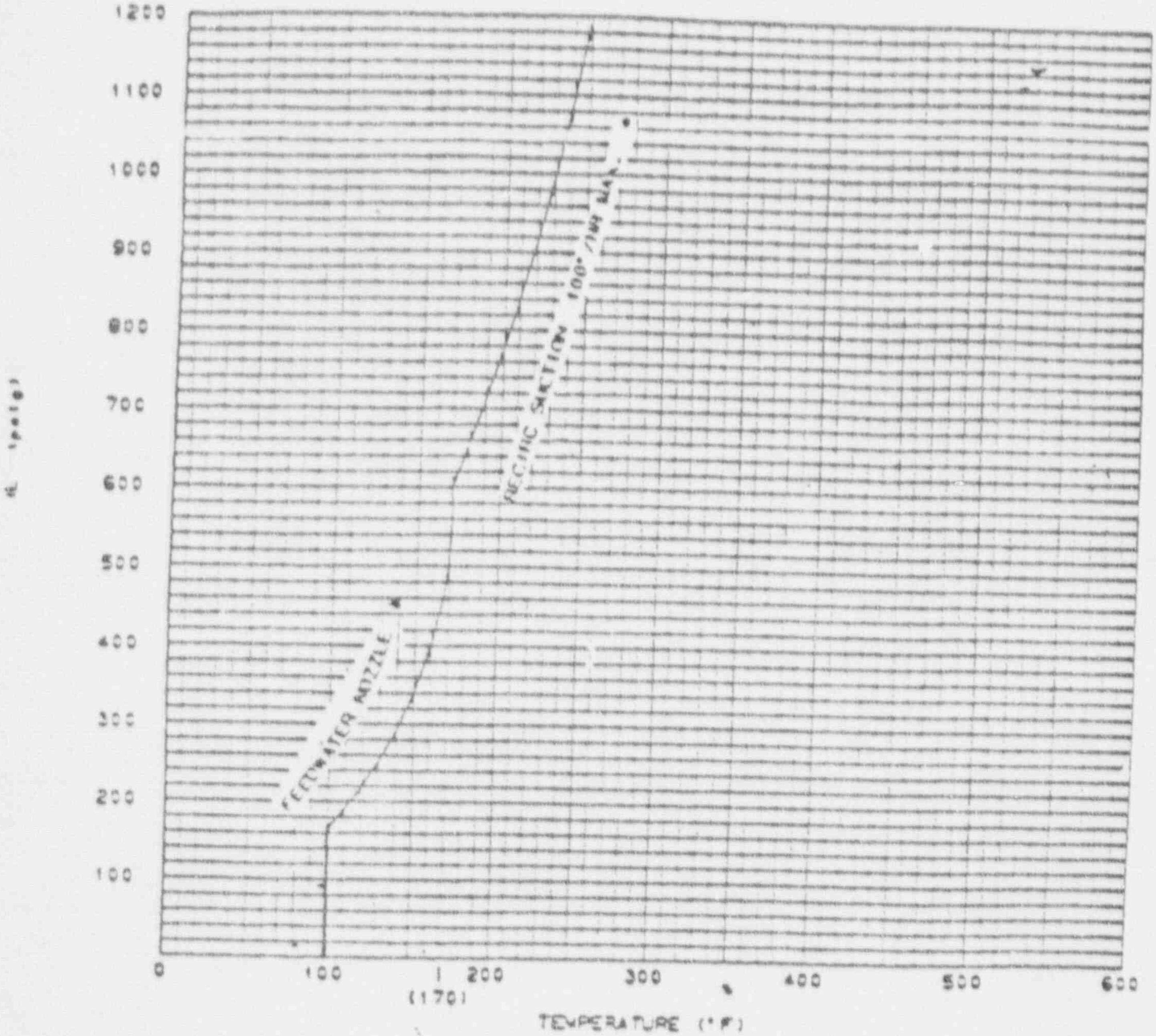
FIGURE 3.4.6.1-3

NOTES:

OPERATE TO RIGHT AND/OR BELOW LIMITING LINES
* INDICATES BOTH HEATUP AND COOLDOWN RATE
PRESSURE AND TEMPERATURE INTERSECTIONS NOTED BY PARENTHESES [illegible] INDICATED TEMPERATURE REQUIRED IF TEST PRESSURE WERE EXCEEDED.

PRESSURE-TEMPERATURE LIMITS
REACTOR VESSEL
BSEP UNIT NO. 2

NORMAL OPERATION WITH CORE NOT CRITICAL



BASES:

1. FUEL IN REACTOR
2. ≤ 16 EFPY
3. 7.1 X 10^{17} N/CM² > 1 MEV
4. RT_{NDT} = 93° (1/4 T)
5. 15 PSI INSTRUMENT LOCATION CORRECTION INCLUDED
6. PROPOSED REG. GUIDE 1.99 REV.2

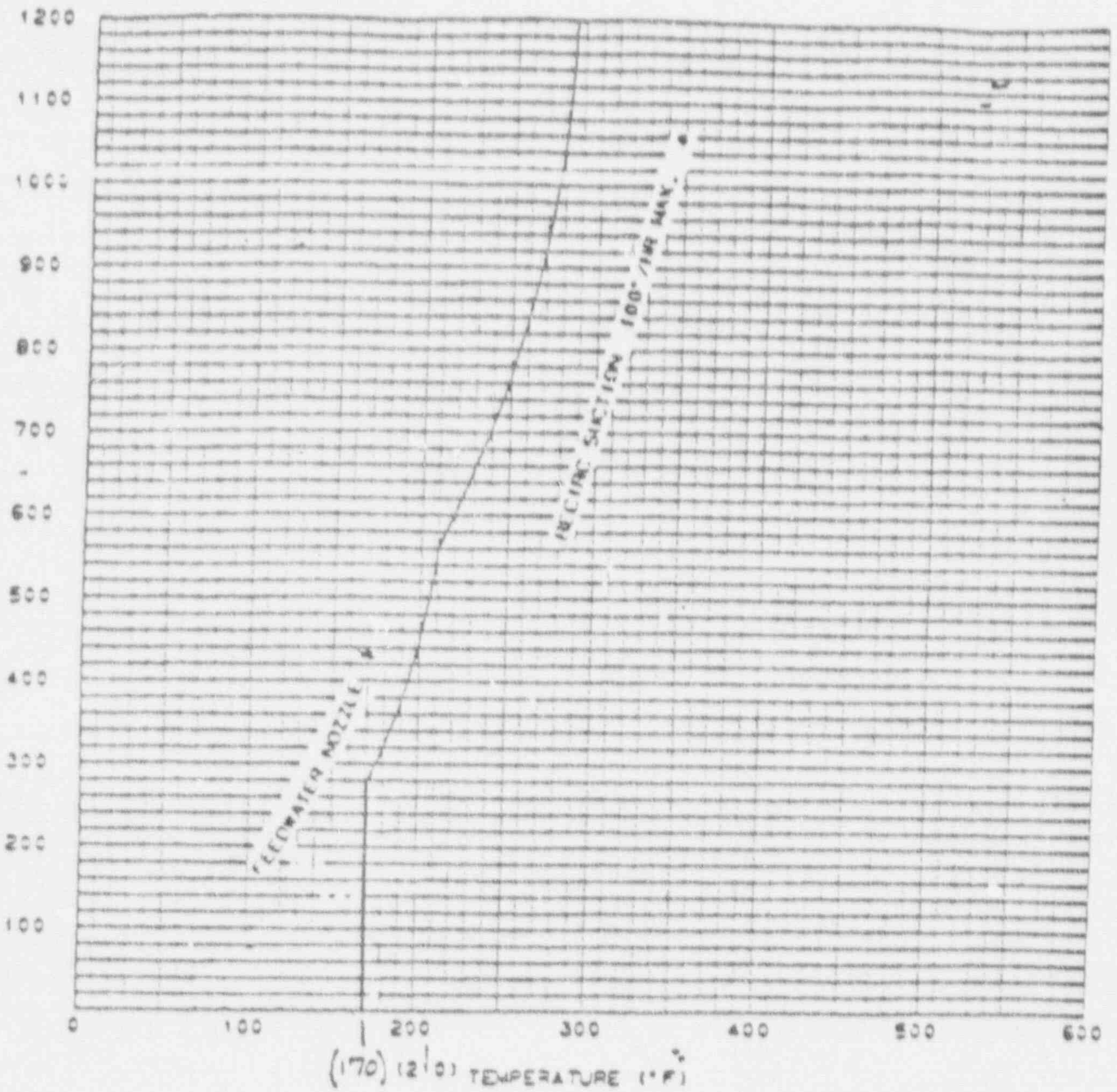
FIGURE 3.4.6.1-1

S:

- OPERATE TO RIGHT AND/OR BELOW LIMITING LINES
- * INDICATES BOTH HEATUP AND COOLDOWN RATE
- PRESSURE AND TEMPERATURE INTERSECTIONS NOTED BY PARENTHESES

REACTOR VESSEL
BSEP UNIT NO. 2

NORMAL OPERATION WITH CORE CRITICAL



BASES:

1. FUEL IN REACTOR
2. ≤ 16 EFPY
3. 7.1 X 10[17] N/CM² > 1 MEV
4. RT_{NDT} = 93° (1/4 T)
5. 15 PSI INSTRUMENT LOCATION CORRECTION INCLUDED
6. PROPOSED REG. GUIDE 1.99 REV.2

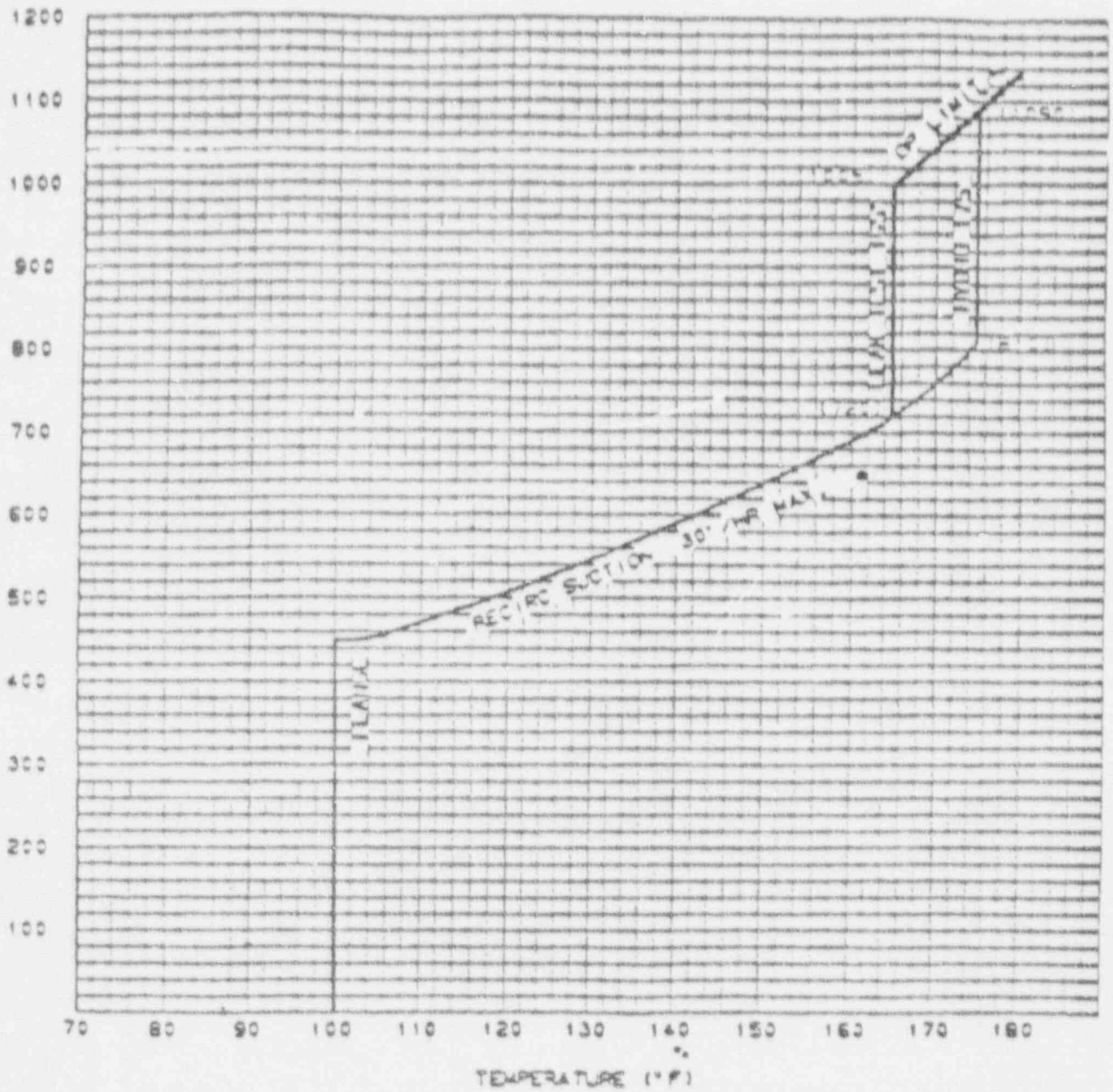
FIGURE 3.4.6.1-2

1:

OPERATE TO RIGHT AND/OR BELOW LIMITING LINES
* INDICATES BOTH HEATUP AND COOLDOWN RATE
PRESSURE AND TEMPERATURE INTERSECTIONS NOTED BY PARENTHESES
~~[illegible]~~
~~[illegible]~~

REACTOR VESSEL
BSEP UNIT NO. 2

HYDROSTATIC AND LEAK TESTS



BASES:

1. FUEL IN REACTOR
2. REACTOR NOT CRITICAL
3. PROPOSED REG. GUIDE 1.99, REV. 2
4. 8 EFPY
5. 3.5 x 10[17] n/cm², 1 MEV
6. RT_{NDT} = 77° (1/4T)
7. 15 PSI INSTRUMENT LOCATION CORRECTION INCLUDED

FIGURE 3.4.6.1-3

NOTE:

- OPERATE TO RIGHT AND/OR BELOW LIMITING LINES
- * INDICATES BOTH HEATUP AND COOLDOWN RATE
- PRESSURE AND TEMPERATURE INTERSECTIONS NOTED BY PARENTHESES
- UPPER LIMIT INDICATES TEMPERATURE REQUIRED IF TEST PRESSURE WERE EXCEEDED.

ATTACHMENT 1

TECHNICAL SPECIFICATION INTERPRETATION REQUEST FORM

8) Serial No. 84-19

1) Technical Specification Reference: Table 3.3.1-1

2) Technical Specification Page No.: 3/4 3-3

3) Subject: RPS-Turbine Control Valve Fast Closure Control Oil Pressure Low Channels (Required no.

4) Unit(s) Affected: BSEP-- Unit 2 only

5) Description of Request (concise detailed description of requested interpretation or problem area): T/S 3.3.1, Table 3.3.1-1 Item 10 requires 4 operable channels per trip system, however, the BSEP design has only 2 channels per trip system. How many channels per trip system are actually required?

6) Originator: D. E. Cobb Date: 10/21/84

7) Reviewed By: [signature] Date: 10/29/84

9) Interpretation: The correct number of required channels per trip system is 2. BASIS: The number in the T/S represents a typographical error introduced when the page was retyped for T/S change request involving digital to analog modifications. When the NRC issued the amendment (#97), the error was also incorporated. The applicable submittal dated 1-26-83 did not discuss a change to the turbine control valve switches nor was this portion of the page "barred" (Cont

10) Technical Specification Change Required: Yes (Editorial) ✓ No

11) Prepared By: D. E. Cobb Date 10/29/84

12) Concurrence: [signature] Date 10/29/84

Director - Regulatory Compliance

13) Approved: [signature] Date 10/29/84

PNSC

9) Interpretation (Cont.)

indicating a change. (The Safety Evaluation issued with the amendment did not address changing the number of channels nor was this portion of the new amendment page "barred" indicating a change). Thus the number of required channels involving the turbine control valve pressure switches is an error and should be 2 instead of 4.

ATTACHMENT 1

TECHNICAL SPECIFICATION INTERPRETATION REQUEST FORM

8) Serial No. 84-20

1) Technical Specification Reference: 4.1.4.2b.

2) Technical Specification Page No.: 3/4 1-16

3) Subject: RSCS Operability

4) Unit(s) Affected: BSEP-- 1 & 2

5) Description of Request (concise detailed description of requested interpretation or problem area): T/S 4.1.4.2b requires demonstrating RSCS OPERABLE by attempting to select and move an out-of-sequence control rod in each of the other three rod groups as soon as RSCS is automatically initiated when reducing THERMAL POWER. However, between 22% full power and 50% rod density, the RSCS has only two out-of-sequence rod groups. How can this requirement be satisfied?

6) Originator: D. E. Cribb Date: 10/29/84

7) Reviewed By: K. E. Enzor Date: 10/28/84

9) Interpretation: The surveillance requirement is satisfied by verifying the rod block is functioning on the two out-of-sequence groups as soon as RSCS is initiated. The remaining rod block shall be verified as soon as 50% rod density is achieved and the RSCS is blocking 3 out-of-sequence groups. Additionally, the group notch control inhibit function shall be verified as soon as the RSCS is automatically initiated (as specified in GE-STS 4.1.4.2b2). (Cont.)

10) Technical Specification Change Required: Yes ✓ No ____

11) Prepared By: D. E. Cribb Date 10/29/84

12) Concurrence: K. E. Enzor Date 10/28/84
Director - Regulatory Compliance

13) Approved: D. [illegible] Date 10/29/84
PNSC

9) Interpretation Cont.

BASIS: The BSEP T/S does not agree with the RSCS design or the GE-STS guidance. The GE-STS does not specify the number of block rod groups. ~~This interpretation has been discussed with the resident inspectors and they agree with the proposed actions.~~ TEC 10/29/87

ATTACHMENT 1

TECHNICAL SPECIFICATION INTERPRETATION REQUEST FORM

1. Serial No. 85-01 (Rev 1)

2. Technical Specification Reference: 3.6.3, Table 3.6.3-1

3. Technical Specification Page No.: 3/4 6-14

4. Subject: Primary Containment Isolation Valves

5. Unit(s) Affected: BSEP-- Both

6. Description of Request (concise detailed description of requested interpretation or problem area):

The above reference lists primary containment isolation valves.

Are these the only valves to which this specification is applicable?

7. Originator: J. W. Chase Date: 6-4-83

8. Reviewed By: J. W. Chase Date: 6-4-83

9. Interpretation: The valves identified in SD-12 Tables 2.4.2 and 2.4.3 are considered primary containment isolation valves per T/S 3.6.3.

10) Technical Specification Change Required: Yes X(84TSB08) No

11) Prepared By: Bob Lill Jr. Date 11/6/87

12) Concurrence: [signature] Date 11/6/87
Director - Regulatory Compliance

13) Approved: [signature] Date 11/17/87
PNSC

RECEIVED

NOV 18 1987

BNP DOC. CONTROL

RECEIVED

MAY 7 1985

NRP DOC. CONTROL

ATTACHMENT 1

TECHNICAL SPECIFICATION INTERPRETATION REQUEST FORM

8) Serial No. 85-04 (35)

1) Technical Specification Reference: 3.1.5/3.3.2

2) Technical Specification Page No.: 3/4 1-18, 3/4 3-12

3) Subject: SLC Operability

4) Unit(s) Affected: BSEP-- Both

5) Description of Request (concise detailed description of requested interpretation or problem area): Specification 3.1.5 requires that SLC be operable in modes 1, 2, & 5 while Specification 3.3.2 requires the RWCU isolation due to SLC initiation be operable in modes 1, 2, & 3. When must the isolation instrumentation for RWCU be operable.

6) Originator: George Milligan Date: 1/24/83

7) Reviewed By: N/A Date:

9) Interpretation: RWCU isolation is required in modes 1, 2, 3, & 5. Basis: As SLC is required in modes 1, 2, & 5, instrumentation required to maintain it operable must be in service; therefore, instrumentation required to isolate RWCU on an SLC initiation must be operable in modes 1, 2, 3 & 5.

10) Technical Specification Change Required: Yes X (84TSB07) No

11) Prepared By: [illegible] Date [illegible]

12) Concurrence: [illegible] Date [illegible]
Director - Regulatory Compliance

13) Approved: [illegible] Date 5/2/85
PNSC

ATTACHMENT 1

TECHNICAL SPECIFICATION INTERPRETATION REQUEST FORM

8) Serial No. 85-10 (19)

1) Technical Specification Reference: Table 3.3.5.7-1

2) Technical Specification Page No.: 3/4 3-61

3) Subject: Fire Detection Instruments

4) Unit(s) Affected: BSEP-- Both

5) Description of Request (concise detailed description of requested interpretation or problem area): The fire detection instruments specified by Table 3.3.5.7-1 for Zone 4 of the AOG building are not consistent with instruments actually installed.

6) Originator: Bill Leonard Date: 11/12/82

7) Reviewed By: N/A Date:

9) Interpretation: The minimum instruments operable for Zone 4 of the AOG Building are 2 flame detectors, 5 heat detectors and 0 smoke detectors.

BASIS: These minimum instruments were specified in a request for a TSC submitted on Sept. 7, 1982 to the NRC. This TSC request was inappropriately superseded by later TSC requests to the NRC dated Dec. 13, 1982 and Oct. 17, 1983. These later requests were based on out-dated information, contained a (Cont.)

10) Technical Specification Change Required: YesX (85TSB16) No

11) Prepared By: Michael Filer Date 5/20/85

12) Concurrence: R S [illegible] Date 5/20/85
Director - Regulatory Compliance

13) Approved: S. Alan Bishop Date 5/24/85
PNSC

9) Interpretation (Cont.)

typographical error, and were issued by the NRC as Amendments 66 and 92. A TSC has been initiated to correct the AOG fire detection instruments identified by Table 3.3.5.7-1.

ATTACHMENT 1

TECHNICAL SPECIFICATION INTERPRETATION REQUEST FORM

8) Serial No. 85-12

1) Technical Specification Reference: Table 3.3.2-3, Item 3.a

2) Technical Specification Page No.: 3/4 3-23

3) Subject: RWCU Response Time

4) Unit(s) Affected: BSEP-- Both

5) Description of Request (concise detailed description of requested interpretation or problem area): The response time for RWCU differential flow specified by T/S Table 3.3.2-3, Item 3.a is ≤13 seconds. However, there are 45 second timers in the logic train.

6) Originator: A. . Curtner Date: 6/12/85

7) Reviewed By: P. D. Musser Date: 6/19/85

9 Interpretation: The response time for the RWCU differential flow is ≤ seconds. However, this time does not include the 45 second timers.

BASIS: The RWCU differential flow instrumentation is part of the steamline break protection provided for the RWCU system. The instrumentation compares the inlet and outlet flows to ensure the leakage from the system is below a specified minimum (≤53 gal/min.). The system incorporates a timer to (Cont.)

10) Technical Specification Change Required: Yes X (TSC 84TSB08) No

11) Prepared By: [signature] Date 7/24/85

12) Concurrence: [signature] Date 7/24/85
Director - Regulatory Compliance

13) Approved: [signature] Date 7/29/85
PNSC

9) Interpretation

prevent spurious isolation during system evolutions. This timer is set at a value high enough to allow the system evolutions to occur, and below the time used in the GE analysis for a line break on this line.

ATTACHMENT 1

TECHNICAL SPECIFICATION INTERPRETATION REQUEST FORM

1. Serial No. 85-13

2. Technical Specification Reference: 3.7.2

3. Technical Specification Page No.: 3/4 [illegible]-3

4. Subject: Control Room Emergency Filtration System (CBEFS) Operability

5. [illegible] Affected: [illegible]

6. Description of Request (concise detailed description of requested interpretation or problem area):

When chlorine or fire detectors are out of service or when these detectors are tripped so as to provide their required input to the CBEFS, is the CBEFS considered operable?

7. Originator: J. R. Harrell (BESU) Date: 11/27/85

8. Reviewed By: J. M. Brown Date: 11/27/85

9. Interpretation: The control building emergency filtration system is operable.

BASIS: The design criteria for the CBEFS is 10CFR50 Appendix A Criterion 19.

"Criterion 19-Control room. A control room shall be provided from which actions can be taken to operate the nuclear power unit safely under normal conditions and to maintain it in a safe condition under accident conditions, including loss-of-coolant accidents. Adequate radiation protection shall be provided to permit access and occupancy of the control room under accident

10) Technical Specification Change Required: Yes X (85TSB24/ 85TSB37) No ____

11) Prepared By: [illegible] Date 12/03/85

12) Concurrence: [illegible] Date 12/3/85
Director - Regulatory Compliance

13) Approved: [illegible] Date 12/5/85
PNSC

9) Interpretation (cont.)

conditions without personnel receiving radiation exposures in excess of 5 rem whole body, or its equivalent to any part of the body, for the duration of the accident."

There is no requirement in Design Criteria 19 for either fire or chlorine detection. These sub-systems are a part of the plant specific design of the CBEFS at Brunswick as described in the FSAR; however, they are not required to meet Design Criteria 19.

In the event of an actuation of the fire detection system, the CBEFS is placed in the same operational status as would be required by a high radiation signal; therefore, there is no question of operability. If a fire detector were to fail such that it would not provide an automatic start of the CBEFS, system design would still allow an automatic start on a high radiation signal. Therefore, neither a fire detector failure nor initiation can create a problem with CBEFS operability.

In the event of a chlorine detector failure or actuation, the CBEFS would align itself into the recirculation mode of operation. In this mode, normal makeup of outside air (1000 scfm) to the system is isolated. With this loss of makeup air, the reduction in positive pressure within the control room allows an increase in the in-leakage of unfiltered air from approximately 275 scfm (CBEFS running) to approximately 1375 scfm (CBEFS isolated). A Control Room Habitability Evaluation performed on BSEP by NUS (per TMI III.D.3.4) evaluated the dose which would be received by control room personnel during a LOCA with different values of unfiltered in-leakage. This evaluation determined that the dose from airborn radioactivity in the control room will peak and level off at 2.8 rem thyroid and 0.004 rem whole body for unfiltered in-leakage of 100,000 scfm or greater. When combined with the other sources of radiation to the control room, the sum totals are 0.415 rem whole body and 2.8 rem thyroid. Based on this, neither a chlorine detector failure nor a CBEFS isolation from the chlorine detection system will create a problem with CBEFS operability per Design Criterion 19.

ATTACHMENT 1

TECHNICAL SPECIFICATION INTERPRETATION REQUEST FORM

1. Serial No. 86-01
2. Technical Specification Reference: 3.1.3.5
3. Technical Specification Page No.: 3/4.1-8
4. Subject: Control Rod Scram Accumulators
5. Unit(s) Affected: BSEP -- Unit 1 and Unit 2
6. Description of Request (concise detailed description of requested interpretation or problem area): What action is required when two or more control rod scram accumulators are declared inoperable in Conditions 1 or 2?
7. Originator: S. L. Russell, STA Date: 1-28-86
8. Reviewed By: M. R. Foss, SOS Date: 1-28-86
9. Interpretation: With two or more control rod scram accumulators inoperable in Modes 1 and 2, operation is outside the defined boundaries of the ACTION statement and T/S 3.0.3 must be entered. To get out of T/S 3.0.3, the number of inoperable accumulators must be reduced to less than or equal to one. This may be done by either correcting the problem or by inserting the associated control rod(s) and either electrically or hydraulically disarming the control rod(s

10) Technical Specification Change Required: Yes X (86TSB02) No ____

11) Prepared By: [signature] Date 02/12/86

12) Concurrence: [signature] Date 2/13/86
Director - Regulatory Compliance

13) Approved: James W. Chase Date 2/13/86
PNSC

The final statement of ACTION a states, "otherwise, be in at least HOT SHUTDOWN within the next 12 hours." This statement only applies if ACTION items a.1 or a.2 are not completed within the required 8 hours. This statement has no bearing on the ACTION to be taken if more than one accumulator is inoperable.

BASIS: The purpose of the scram accumulators is to ensure that sufficient energy is available to insert the control rod in the most unfavorable condition. This condition would be a LOCA with a loss of off-site power where the CRD system and vessel pressure would both be lost. In this condition, the only available source for control rod insertion would be the accumulator. Insertion prevents a pattern of inoperable accumulators that would result in less reactivity insertion on a scram than has been analyzed.

Inserting and electrically or hydraulically isolating such a control rod removes it from the inoperable accumulator specification and places you in the inoperable control rod (3.1.3.1) specification.

ATTACHMENT 1

TECHNICAL SPECIFICATION INTERPRETATION REQUEST FORM

1. Serial No. - 86-02

2. Technical Specification Reference: Section 3/4 3.2 Table 3.3.2-1

3. Technical Specification Page No.: 3/4 3-13 and 3/4 3-16

4. Subject: Secondary Containment Requirement in Condition 5

5. [illegible] Affected: BSEP -- Unit 1 (one) and Unit 2 (two)

6. Description of Request (concise detailed description of requested interpretation or problem area): If D12-RM-N010A and/or N010B are inoperable while in Mode 5, must Secondary Containment Integrity be established with the SBGT's operating within one hour? (Action statement 23)

7. Originator: Ricky D. Tart   Date: 1-26-86

8. Reviewed By: A. Hegler   Date: 1-27-86

9. Interpretation: If the N010A and/or N010B is inoperable in Mode 5, the ACTION which should be taken is "Establish Secondary Containment Integrity with the standby gas treatment system operations within one hour or enter the ACTION statements for 3.6.5.1, 3.6.5.2, and 3.6.6.1 as applicable.

10) Technical Specification Change Required: Yes X (86TSB03) No ____

11) Prepared By: [signature]   Date 02/12/86

12) Concurrence: [signature]   Date 2/12/86
Director - Regulatory Compliance

13) Approved: [signature]   Date 2/13/86
PNSC

BASIS:

The intent of ACTION statement 23 to T/S 3.3.2 is to ensure that those functions initiated by N010A(B) are established, as the initiating instrumentation is inoperable. These instruments are used to isolate the secondary containment damp and to start the SBGT's on a high reactor building exhaust radiation signal. Each of the functions required by ACTION 23 [SECONDARY CONTAINMENT INTEGRITY, SBGT's, secondary containment dampers (implied)] have their own unique ACTION requirements, and all are required in the same modes. Should ACTION 23 not be able to be followed (any of the three functions cannot be performed), the implication would be that T/S 3.0.3 must be entered; however, you are already in Cold Shutdown.

ACTION 23 [illegible] a defined set of functions should N010A(B) be inoperable. If the required ACTION's for those functions (T/S 3.6.5.1, 3.6.5.2, 3.6.6.1) are taken when they are inoperable, then the intent of ACTION 23 is met.

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ATTACHMENT 1

TECHNICAL SPECIFICATION INTERPRETATION REQUEST FORM

1. Serial No. 87-01 (Rev. 1)

2. Technical Specification Reference: 3.8.1.1

3. Technical Specification Page No.: 3/4 8.1

4. Subject: Required Off Site AC Power Sources

5. Unit(s) Affected: BSEP-- Both

6. Description of Request (concise detailed description of requested interpretation or problem area): What constitutes "two physically independent circuits between the offsite transmission network and the onsite Class IE distribution system," as it applies to OPERABILITY and ACTION statements a,b, ~~and~~ c and D.

7. Originator: Bob Poulk, Jr. Date: 6/19/89

8. Reviewed By: Earl Enzor Date: 6/19/89

9. Interpretation: For this LCO statement to be satisfied, the following must be maintained for each unit:

1) two operable incoming transmission lines to ~~both~~ the switchyard~~s~~ capable of supplying both the UAT and SAT.

2) an operable distribution system from both the UAT and SAT to BOP buses 1(2)C and 1(2)D.

3) an operable distribution system to supply power from the 1(2)C and 1(2)D to its respective emergency bus.

10) Technical Specification Change Required: Yes ____ No x

11) Prepared By: Bob Poulk Jr. Date 6/19/89

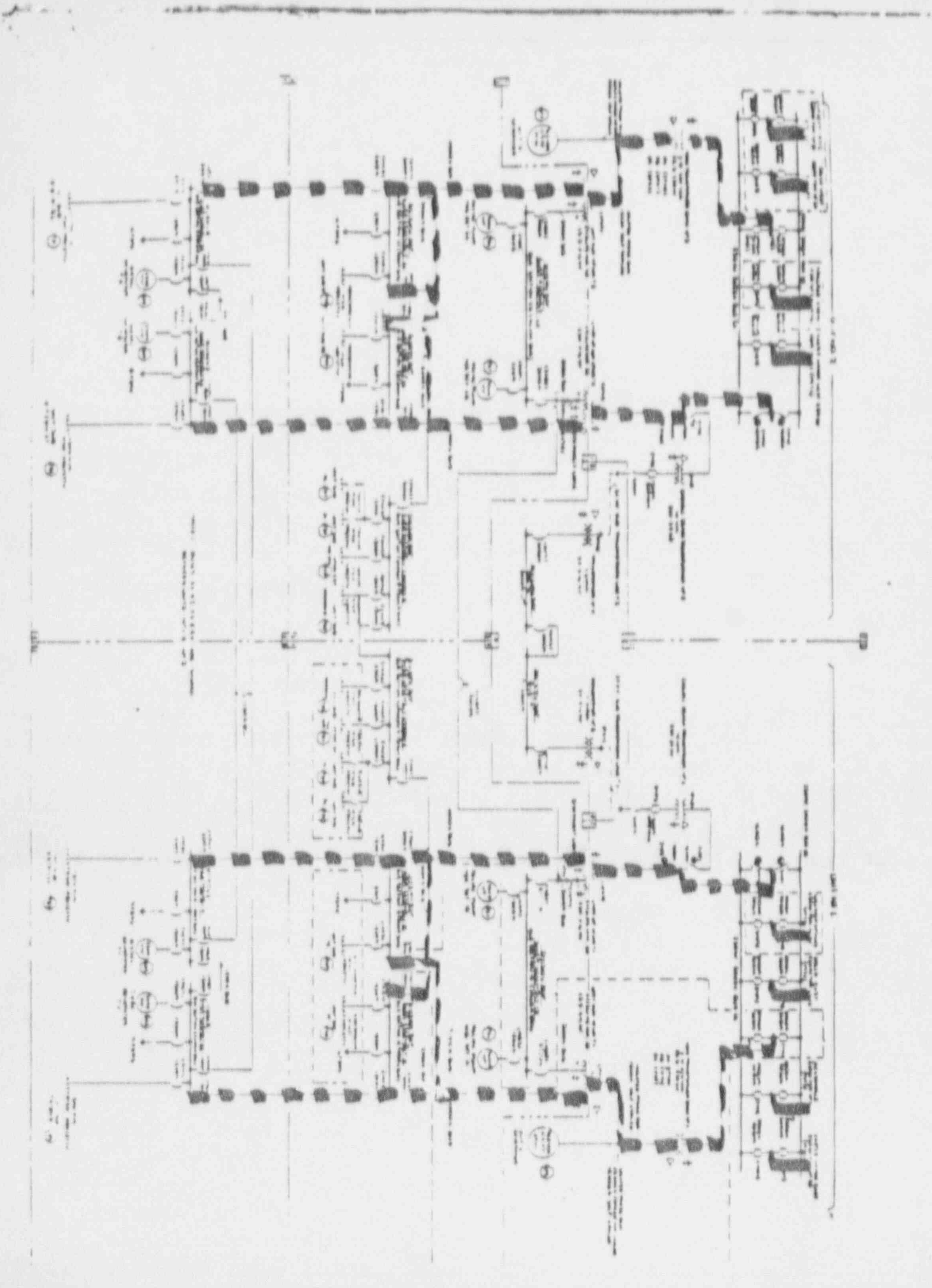
12) Concurrence: [signature] Date 6/19/89
Director - Regulatory Compliance

13) Approved: [signature] Date 6/19/89
PNSC

BASIS:

The original interpretations provided by TSI's 85-08 (alternate/normal) and 87-01 (two independent off-site power sources) was based on the design information provided in the FSAR as to how BNP satisfies the requirements of General Design Criteria 17. The plant design document took credit for two off-site power feeds to the switchyard as meeting this criteria. Following a technical specification submittal and subsequent discussions with the NRC on this matter, it has been determined that the requirement for two independent off-site power sources is met by the Unit Auxiliary Transformer [(UAT) alternate] and the Startup Auxiliary Transformer [(SAT) normal]. This was addressed by the NRC in their response to the technical specification change request dated May 25, 1989.

It is recognized that the UAT can not normally supply the emergency buses when the generator is off line. Credit for UAT operability is taken by the ability to backfeed through the UAT following the removal of the generator disconnects.



ATTACHMENT 1

TECHNICAL SPECIFICATION INTERPRETATION REQUEST FORM

1. Serial No. 87-02

2. Technical Specification Reference: 3.5.3.2 and 3.6.2.2

3. Technical Specification Page No.: 3/4 5-7 and 3/4 6-11

4. Subject: Operation with inoperable RHR room coolers

5. [illegible] Both

6. Description of Request (concise detailed description of requested interpretation or problem area): What action, if any, should be taken if one or both RHR room coolers are inoperable?

7. Originator: Bob Poulk, Jr. Date: 8/27/87

8. Reviewed By: Earl Enzor Date: 8/27/87

9. Interpretation: (See Attached)

10) Technical Specification Change Required: Yes 87TSB19 No

11) Prepared By: Bob Poulk Jr. Date 8/27/87

12) Concurrence: K. Enzor Date 8/25/87
Director - Regulatory Compliance

13) Approved: C. J. Date 8/28/87
PNSC

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9) Interpretation

Limiting Condition for Operation

3.5.3.2 c. One LPCI room cooler

3.6.2.2. The suppression pool cooling mode of residual heat removal (RHR) system shall be OPERABLE with two independent cooling loops, each loop consisting of two pumps, one heat exchanger, and one room cooler.

APPLICABILITY: Condition 1, 2, 3, 4*, and 5*

ACTION (for room cooler inoperability only):

a. In Condition 1, 2 or 3:

1. With one LPCI subsystem room cooler inoperable:

a. Restore the inoperable LPCI subsystem room cooler to OPERABLE status within 7 days, or;
b. Demonstrate the OPERABILITY of the remaining redundant LPCI subsystem room cooler by performing Surveillance Requirement 4.X within 2 hours and at least once per 24 hours thereafter for the next 7 days;
c. Restore the inoperable LPCI subsystem room cooler to OPERABLE status within 14 days or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With both LPCI subsystem room coolers inoperable, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In Condition 4* and 5* with one or more LPCI subsystems room coolers inoperable, take the ACTION required by Specification 3.5.3.1. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.X Each LPCI subsystem room cooler shall be demonstrated OPERABLE:

a. At least once per 24 hours when required by ACTION a.1.b above by:

1. Manually starting the LPCI subsystem room cooler and verifying air flow and service water flow from the cooler unit.

*Not applicable when two CSS subsystems are OPERABLE per Specification 3.5.3.1.

BASIS: See EER 86-0460

ATTACHMENT 1

TECHNICAL SPECIFICATION INTERPRETATION REQUEST FORM

1. Serial No. 87-03

2. Technical Specification Reference: 3.6.6.3

3. Technical Specification Page No.: 3/4 6-29

4. Subject: Oxygen Concentration

5. Unit(s) Affected: BSEP-- Both

6. Description of Request (concise detailed description of requested interpretation or problem area): Specification 3.6.6.3.b requires "within 24 hours prior to a scheduled reduction of thermal power to less than 15% of rated thermal power." How is this specification to be applied to "scheduled reduction"?

7. Originator: [illegible] Date: 11/06/87

8. Reviewed By: K. E. Enzor Date: 11/06/87

9. Interpretation: This specification is intended to allow for the deinerting of the drywell prior to decreasing power to less than 15% or scheduled power reductions requiring drywell entry. This is applicable for both planned and exigent power decreases. The requirement remains that power must be less than 15% or the drywell oxygen concentration must be less than 4% within the 24 hours, or the action statement entered. This has been concurred with by Region II and NRR.

10) Technical Specification Change Required: Yes 87TSB25 No ____

11) Prepared By: [signature] Date 11/6/87

12) Concurrence: [signature] Director - Regulatory Compliance Date 11/6/87

13) Approved: [signature] PNSC Date 11/12/87

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ATTACHMENT 1

TECHNICAL SPECIFICATION INTERPRETATION REQUEST FORM

1. Serial No. 88-01

2. Technical Specification Reference: Table 4.3.1-1 (1a, 1b, 2a) Note "d" Table 4.3.6-1 (1d, 3a,b,c,d, 4a,b,c,d) Note "c"

3. Technical Specification Page No.: 3/4 3-7, 3/4 3-8a, 3/4-3-63, 3/4 3-63a

4. Subject: NEUTRON MONITORING SURVEILLANCE REQUIREMENT

5. Unit(s) Affected: BSEP-- 1 AND 2

6. Description of Request (concise detailed description of requested interpretation or problem area):

SEE ATTACHED SHEET FOR ADDITIONAL INFORMATION

Note "d" would indicate that the referenced surveillances must be performed within 12 hours after entering OPERATIONAL CONDITION 2. Can the surveillances be performed in other OPERATIONAL CONDITIONS and still meet T/S?

7. Originator: [signature] EXT. 2704 Date: ORIGINAL SIGNED 4/25/88

8. Reviewed By: PORC user Date: original signed 4/25/88

9. Interpretation:

SEE ATTACHED SHEET FOR ADDITIONAL INFORMATION

Those surveillances identified above may be performed in conditions other than "...within 12 hours after entering OPERATIONAL CONDITION 2".

10) Technical Specification Change Required: Yes 88-5812 No

11) Prepared By: [signature] Date 5/20/88

12) Concurrence: [signature] Date 5/20/88

Director - Regulatory Compliance

13) Approved: [signature] Date 5/20/88

PNSC

Attachment to Tech Spec interpretation request form

Item No. 6:

The tech spec surveillance requirements listed above refer to a Note "d" which states that the APRM, IRM, and SRM trip functions must be checked within 12 hours of entering Operational Condition 2 from Operational Condition 1. However, the note does not take into account switching back and forth from Condition 1 and 2 within a seven day period.

As an example, it is possible the above listed tech spec trip functions are in surveillance because the applicable MST's were performed in the previous seven days to satisfy going to Operational Condition 2 from Operational Condition 3 or 4. The Unit is then taken to Operational Condition 1 and immediately back to Operational Condition 2 still within the seven day period.

It is believed that note "d" was intended to require checking these trip functions after an extended period of being in Operational Condition 1, i.e., greater than seven days; in which case the trip functions should be checked if the intent is to remain in Operational Condition 2 for any extended period of time.

It should not be a requirement to perform the trip function surveillance again within the same seven day period if it is intended to switch Operational Conditions while these trip functions still fall within the seven day surveillance requirement.

The request is to exempt performing the note "d" required surveillance if the trip functions have been checked by the applicable MST's within there normal tech spec frequency and the unit was taken from Operational Condition 1 to Operational Condition 2.

Item No. 9:

Footnote "d" as referenced above was provided to allow the performance of those identified surveillances within 12 hours after entering OPERATIONAL CONDITION 2, due to extensive technical requirements to perform them prior to entering MODE 2. The intent of that footnote is not to prevent the performance of those surveillances when in OPERATIONAL CONDITION 1 where it is possible to do so. Therefore, if a surveillance test is performed in OPERATIONAL CONDITION 1 and is still within it required scheduling periodicity, the test need not be performed again "within 12 hours after entering OPERATIONAL CONDITION 2."

Footnote "d" was requested by BNP as a T/S change in 1985 to allow the performance of these suveillances after entering OPERATIONAL CONDITION 2 due to difficulties with performing them in OPERATIONAL CONDITION 1 prior to shutting down. Before Amendment 96 and 121 were approved on 3/86, performance of this surveillance was required in OPERATIONAL CONDITION 1. A TSC will be issued to clarify the intent of Footnote "d" as reflected in Amendment 96 and 121.

ATTACHMENT 1

TECHNICAL SPECIFICATION INTERPRETATION REQUEST FORM

1. Serial No. 89-01

2. Technical Specification Reference: Definitions

3. Technical Specification Page No.: 1-2

4. Subject: Core Alteration

5. Unit(s) Affected: BSEP-- Both

6. Description of Request (concise detailed description of requested interpretation or problem area): What constitutes a "core alteration" as referenced in the definitions?

7. Originator: R. M. Poulk, Jr. Date: 2/17/89

8. Reviewed By: K. E. Enzor Date: 2/17/89

9. Interpretation: A core alteration is the addition, removal, relocation, or movement of fuel, sources, incore instruments, or reactivity controls in the reactor core with the head removed and fuel in the vessel by other than its normal controling mechanism. Examples: 1) An SRM or Control Rod repositioned or moved by its normal drive mechanism in its normal operating channel is not a core alteration. 2) An SRM or a control blade assembly (cont.)

10) Technical Specification Change Required: Yes X ([illegible]) No ____

11) Prepared By: [signature] Date 2/17/89

12) Concurrence: [signature] Date 2/17/89
Director - Regulatory Compliance

13) Approved: [signature] PNSC Date 2-[illegible]

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9. Interpretation (cont.)

removed from or relocated in the reactor vessel is a core alteration. 3) The addition, removal, or relocation of sources and/or fuel by any mechanism is a core alteration. 4) Removal of an LPRM string or an IRM is a core alteration; however, withdrawal or insertion of the IRM using its normal drive system is not a core alteration. 5) Withdrawal of a control rod with the CRD system and then the removal of the drive unit is not a core alteration. 6) Insertion, removal, or relocation of blade guides is a core alteration.

BASIS: See Attached

BASIS

Neither the DEFINITIONS section nor the REFUEL section of the Brunswick Technical Specifications (BTS) provides adequate guidance for this concern. To develop this interpretation, reviews of BTS history and NRR approved NUREG's, discussions with other BWR's, and a review of the requirements that could not be met with the current BTS with a different interpretation, were conducted.

The original BTS contained a definition for "Alteration of the Reactor Core" which was the forerunner of the current Standard Technical Specifications (STS). Contained within that definition is the following clarification: "*Normal control rod movement with the control rod drive hydraulic system is not defined as a core alternation. Normal movement of incore instrumentation is not defined as a core alteration.*"

In 1977, Brunswick began operation under the STS. The definition of CORE ALTERATION as provided in the STS was worded differently from that in the previously used TS; however, as the STS had to be accepted as presented and there was no indication that the intent was different, operation at Brunswick continued with the understanding that control rod movement was not considered a CORE ALTERATION.

Support that the intent was not changed is reflected in NRR approved NUREG's 1141 (Fermi) and 1162 (Perry), and other NUREGs for various BWR's (plant specific STS's). These STS's either provide wording in the DEFINITION section for CORE ALTERATION which mirror the wording contained within the original BTS or they provide allowed exceptions within the REFUELING section of technical specifications.

Brunswick conducted a survey of other utilities to determine their understanding of rod movement using the normal hydraulic system as a CORE ALTERATION. The initial input indicates that the Brunswick interpretation is consistent with the industry practice (seven utilities responded thus far).

No basis for the term CORE ALTERATION is provided in STS; however, bases are provided for the individual REFUEL specifications. The basis for CONTROL ROD POSITION states "*...all control rods be inserted during CORE ALTERATIONS* (sic. which implies that the two are not synonymous) *ensures that fuel will not be loaded into a cell without a control*

rod and prevents two positive reactivity changes from occurring simultaneously." In addition, were normal movement of control rods and SRMs/IRM to be considered a CORE ALTERATION, surveillances required by STS for one-rod-out interlock and SRM operability could not be performed due to the requirement to move these components.

Therefore, based on the available history and current NRR approved methodology for CORE ALTERATION, this interpretation defines a consistent and safe definition for Brunswick.

Appendix 5

Evaluation of Brunswick Technical Specification Interpretations

EVALUATION OF BRUNSWICK TECHNICAL SPECIFICATION INTERPRETATION

A. Technical Specification Interpretation Request Form-Serial No. 84-06, (Revision 5)(Both Units)

The subject of this Technical Specification Interpretation (TSI) is Service Water System Operability.

B. Background

Technical Specification (TS) 3.7.1.2 states that the service water system nuclear header shall be operable with at least three operable service water pumps. However, the TS does not specify what service water pumps must be operable in order to satisfy the TS requirements.

The TSI states that the service water system for each unit has five pumps. Two are designated nuclear service water pumps and three are designated conventional service water pumps. The two nuclear pumps service the nuclear header and the three conventional pumps service the conventional header. Each conventional service water pump can service the nuclear header by closing a valve associated with the conventional header and opening a valve associated with the nuclear header. All ten service water pumps are located in the service water building.

The licensee made the interpretation that the Unit 1 and 2 service water system shall be operable with at least:

a. Four operable nuclear service water pumps for the site capable of supplying service water to the nuclear headers; and in addition,

b. Two operable Unit 1 (2) conventional service water pumps, each powered from a different division and capable of supplying both the Unit 1 (2) service water nuclear header and conventional header.

The licensee performed a failure analysis for the TSI to support the interpretation. The licensee evaluated the assumed failure of each of the four safety related 4160 volt buses/diesel generators to ensure cooling water to (a) essential components of the unit experiencing the loss of coolant accident and (b) the unit experiencing a safe shutdown condition. The licensee concluded that the safety function would be fulfilled.

The original TSI was approved by the plant nuclear safety committee on March 12, 1984. The TSI was modified a number of times since that time. A Diagnostic evaluation of the Brunswick Plant was conducted early in 1989. One of the systems evaluated by the team was service water. The team found serious deficiencies with the system. Examples included single failure vulnerability, nuclear header to conventional header service water leakage, pump motor reliability and unavailable preoperational/startup test data. The licensee developed a justification for continued operation (EER No. 89-0135, Rev 0, 5/4/89) until such time that design changes could be implemented in the plant. The TSI discussed here is Revision 5 which was approved by the Plant Nuclear Safety Committee on May 5, 1989. This revision reflects the licensee's JCO.

The licensee recognized that the TS required a TS change. One was submitted on November, 1988. As a result of the DET evaluation and the licensee's subsequent evaluation, the licensee requested the staff to put the TS change request on-hold. The licensee subsequently withdrew the application on October 30, 1989, on the basis that a new application would be submitted in February 1990, at which time the licensee's evaluations would be completed.

C. Evaluation

The present TS lacks definition as to which three pumps must be operable. This is important becuase the nuclear service water pumps start on an accident signal (if not already running) and the conventional pumps do not. The TS bases do not provide clarity. The present TS would permit the two nuclear service water pumps for each unit to be out of service, and the licensee would not be in an action statement. This is not appropriate. TS that reflect the design basis of the system are necessary.

The licensee's TSI is a step in the right direction. The requirements specified in the interpretation ensures that the plant's design basis can be met. However, the TSI Revision 5 is not a final substitute for the present TS. The licensee can change the TSI at anytime without staff approval. It should be noted that this TSI is a revision to the TSI reviewed by the DET.

The ability of the service water system to fulfill its design function was a major concern identified by the Diagnostic Evaluation Team during a Spring 1989 inspection. The licensee was notified that a violation is under consideration for escalated enforcement action in Inspection Report 89-34, dated November 30, 1989. An Enforcement Conference was held on December 15, 1989.

D. Conclusion

There is no immediate safety concern at this time, as long as the TSI Revision 5 remains in-effect. There may have been an immediate safety concern prior to Revision 5. If a further revision is adopted, it should have at least the same level of safety. A TS change that reflects the design basis of the system is necessary.

Escalated Enforcement is pending.

E. Reference

As stated

F. Principal Contributor: E. G. Tourigny

EVALUATION OF BRUNSWICK TECHNICAL SPECIFICATION INTERPRETATION

A. TECHNICAL SPECIFICATION INTERPRETATION REQUEST FORM - SERIAL NO. 84-08 (REV. 1) (BOTH UNITS)

The subject of this Technical Specification Interpretation (TSI) is concerned with the Core Spray Time Relay

B. BACKGROUND

Brunswick Technical Specification (TS) Table 3.3.3-2, Item 1d indicates that the trip setpoint for the E21-K16A,B time delay relay is $14 \leq t \leq 16$ sec. The TSI states that this relay is actually a 5 sec. relay. This TSI was approved by the Plant Nuclear Safety Committee on May 24, 1985.

The licensee justified this TSI on the basis that the K16 relay is in series with the STR 1A2, 1B2 (2.A2, 2.B2), which is a 10 second time relay, to provide the 15 second delay for core spray. The correct time delay value of the K16A,B relays is $4.5 \leq t \leq 5.5$ seconds.

C. EVALUATION

The staff has verified that the 15 second delay for core spray is consistent with the safety analysis of the plant as documented in the Brunswick Updated Final Safety Analysis Report (UFSAR).

The staff noted that even though the TSI value assigned to E21-K16A or B is 4.5 t 5.5 sec., the core spray time delay specified in the UFSAR is 15 seconds. Time delay relays STR 1A2, 1B2 (2.A2, 2.B2) in series with the E21-K16A relay provide the additional 10 second delay for the core spray time delay. Since the UFSAR specifies the 15 second delay, the TS should be changed to show the correct relays and/or times for the core spray system.

The staff noted that although the 1984 TSI indicated that a TS change was initiated, one was not submitted to the NRC.

D. CONCLUSION

The staff determined that the overall delay time of 15 seconds for core spray is consistent with the safety analysis of the Brunswick plant and should remain in the TS.

The staff considers the TSI incomplete because it does not address the design of the core spray time delay. A TS change should be submitted to the NRC to include the STR 1A2, 1B2 (2.A2, 2.B2) relays in the TS, as separate items or combined with the K16 relay.

The staff concludes that a breakdown in the licensee's administrative controls appears to have occurred because no TS change was proposed to the NRC since the TSI was formalized on May 24, 1985.

E. REFERENCE

As stated

F. PRINCIPAL CONTRIBUTOR

OTSB

EVALUATION OF BRUNSWICK TECHNICAL SPECIFICATION INTERPRETATION

A. Technical Specification Interpretation Request Form - Serial No. 84-10 (Revision 6)(Both Units)

The subject of this Technical Specification Interpretation (TSI) is Pressure/Temperature Limits for the Reactor Pressure Vessel.

B. Background

The licensee recognized that existing technical specification pressure-temperature limit figures have not been revised to reflect current General Electric basis documents. Upon review of the documents, the licensee determined that the current TS figures are not sufficiently conservative. The original TSI was written in 1984. The licensee submitted a TS change request by application dated October 26, 1988, as supplemented March 30, 1989, June 13, 1989, and August 4, 1989. The figures contained in the TSI (Revision 6) are to be used in the interim. The pressure-temperature limits of the current TSI are based upon the latest NRC staff guidance (Revision 2 of RG 1.99) and 10 CFR 50, Appendix G. The figures are contained in the TSI. This interpretation was approved by the Plant Nuclear Safety Committee on June 29, 1989.

C. Evaluation

Pressure-temperature limits are imposed upon reactor coolant system components so that they are not overstressed during cyclic operation.

The pressure-temperature limit figures in the TSI are exactly the same and/or bounded by the proposed and existing TS pressure-temperature limit figures. The staff's evaluations of the licensee's proposed technical specifications are almost complete. No significant issues have been identified that would preclude issuance of the amendment.

D. Conclusion

The licensee's TSI does not represent an immediate safe ncern, since the TSI figures are exactly the same and/or bounded by the existing and proposed TS figures. The proposed TS figures should be approved in the near future.

Although a TS change has been submitted to the staff and should be issued in the near future, the TSI has been in-place as one revision or another since 1984. This appears to be a breakdown in the licensee's administrative controls.

E. Reference

As stated

F. Principal Contributor: E. G. Tourigny

EVALUATION OF BRUNSWICK TECHNICAL SPECIFICATION INTERPRETATION

A. TECHNICAL SPECIFICATION INTERPRETATION REQUEST FORM - SERIAL NO. 84-19 (Unit 2 Only)

The subject of this Technical Specification Interpretation (TSI) is concerned with the number of required Reactor Protection System (RPS) - turbine control valve fast closure control oil pressure low channels.

B. BACKGROUND

Brunswick Technical Specification (TS) 3.3.1, Table 3.3.1-1, Item 10 requires 4 operable channels per trip system; however, the TSI states that the plant design has only 2 channels per trip system. This TSI was approved by the Plant Nuclear Safety Committee on October 29, 1984.

The licensee justified this TSI on the basis that the number in the Brunswick TS represents a typographical error introduced when the page was retyped for TS change requests involving digital to analog modifications. This error was incorporated in license amendment number 97 issued by the NRC.

C. EVALUATION

The staff's review of similar TS for Unit 1 showed that there are only two channels per trip system, which is consistent with the licensee's interpretation for Unit 2. The staff also reviewed the Updated Final Safety Analysis Report (UFSAR), Table 7.2.2-2 entitled "Reactor Protection System Minimum Number of Channels Required for Functional Performance in Run Mode." The number of channels per trip system shown in this table of the UFSAR for the turbine control valve fast closure is 2.

The staff noted that although the 1984 TSI indicated that a TS change was needed, one was not submitted to the NRC.

D. CONCLUSION

The staff determined that the correct number of channels per trip system for the turbine control valve fast closure is 2.

Although there is no immediate safety concern, a TS change should be submitted to the NRC so the Brunswick TS can be corrected.

The staff concludes that a breakdown in the licensee's administrative controls appears to have occurred because no TS change was proposed to the NRC since the TSI was formalized on October 27, 1984.

E. REFERENCE

As stated

F. PRINCIPAL CONTRIBUTOR

OTSB

EVALUATION OF BRUNSWICK TECHNICAL SPECIFICATION INTERPRETATION

A. Technical Specification Interpretation Request Form - Serial No. 84-20 (Both Units)

The subject of this Technical Specification interpretation (TSI) is Rod Sequence Control System (RSCS) Operability.

B. Background

Technical Specification 4.1.4.2b requires demonstrating the Rod Sequence Control System operable by attempting to select and move an out-of-sequence control rod in each of the other three groups as soon as RSCS is automatically initiated when reducing thermal power. However, between 22% full power and 50% rod density, the RSCS has only two out-of-sequence rod groups. The RSCS design for certain conditions therefore cannot support the technical specification requirement.

The surveillance requirement is satisfied by verifying the rod block is functioning on the two out-of-sequence groups as soon as RSCS is initiated. The remaining rod block shall be verified as soon as 50% rod density is achieved and the RSCS is blocking 3 out-of-sequence groups. Additionally, the group notch control inhibit function shall be verified as soon as the RSCS is automatically initiated (as specified in GE-STS 4.1.4.2b).

The licensee's basis for the TSI is that the TS do not agree with the RSCS design or the GE - STS guidance. In addition, the GE-STS does not specify the number of blocked rod groups.

The TSI was approved by the Plant Nuclear Safety Committee on October 29, 1984.

C. Evaluation

The RSCS restricts rod movement to minimize the individual worth of control rods to lessen the consequences of a control rod drop accident. Control rod movement is restricted through the use of rod select, insert, and withdraw blocks. The RSCS is a hardwired, redundant backup system to the rod worth minimizer (RWM). RSCS is required to be operable below 20% power.

According to the UFSAR, the licensee uses 22% power to ensure that the TS of 20% is met. In addition, the RSCS is automatically in and out of service when the preset power level is reached. The present power level is 30%. The licensee uses two ranges for the RSCS: 100% to 50% rod density and 50% rod density to present power level. The first range uses a four RSCS group approach (sequence control). The second range uses a single notch approach (group notch control). One hundred percent rod density is when all rods are in.

The TS specifies that the RSCS shall be demonstrated operable by attempting to select and move an out-of-sequence control rod in each of the other three rod groups:

a. In CONDITION 2 prior to the start of control rod withdrawal for a reactor startup, and

b. As soon as the RSCS is automatically initiated during control rod insertion when reducing thermal power.

The b part of the TS cannot be met in the 50% rod density to preset power level range because of the system design. The licensee's approach is acceptable until 50% rod density is achieved, at which time the licensee fully adheres to the TS. In addition, the rod worth minimizer adds additional assurance that the plant is being operated safely. Thus, there is no immediate safety concern.

Although the 1984 TSI indicated that a TS change was needed, one was not submitted.

Other licensees have requested the deletion of the TS for RSCS and received approval by staff. This was an effort led by the BWR Owners Group. Brunswick may desire to take a similar approach.

D. Conclusion

The inconsistency between the TS and the plant design for the range of 50% of rod density and present power level does not present an immediate safely concern, because the licensee fully complies with the TS at the 50% rod density point and the RWM is acting as a backup.

A breakdown in the licensee's administrative control appears to have occurred because no TS change has been submitted since the TSI was formalized on October 29, 1984. The staff believes that the TS should be corrected or deleted.

E. Reference:

As stated

F. Principal Contributor E. G. Tourigny

EVALUATION OF BRUNSWICK TECHNICAL SPECIFICATION INTERPRETATION

A. Technical Specification Interpretation Request Form - Serial No. 85-01 (Revision 1)(Both Units)

The subject of this Technical Specification Interpretation (TSI) is Primary Containment Isolation Valves.

B. Background

Technical Specification (TS) Table 3.6.3-1 lists the primary containment isolation valves and their associated isolation times and group numbers. A question was asked by plant personnel as follows "Are these the only valves to which this specification is applicable". A TSI was written to address this which states "The valves identified in SD-12 Tables 2.4.2 and 2.4.3 are considered primary containment isolation valves per TS 3.6.3". SD-12 is a plant procedure.

Revision 1 of the TSI was approved by the Plant Nuclear Safety Committee on November 17, 1987.

A TS change was submitted to NRC in February 1988, which, among other requests, would delete Table 3.6.3-1 valves from the TS. The application was completely superseded in September 1989.

C. Evaluation

Table 2.4.2 of SD-12, Revision 13, May 8, 1989, entitled "Containment Isolation by Penetration," contains a listing of all penetrations, and for each penetration, valve numbers and containment isolation valve group information are illustrated.

Table 2.4.3 of SD-12 entitled "Containment Isolation by Valve Number" contains a listing of all valves; and, for each valve, the penetration associated with it.

TS Table 3.6.3-1 contains a listing of automatic primary containment isolation valves, their isolation group number, and their isolation time.

A comparison was made between the automatic isolation valves listed in SD-12 and the valves listed in Table 3.6.3-1 of the TS. It was noted that there were automatic isolation valves listed in the TS that were not listed in SD-12 and there were automatic isolation valves listed in SD-12 that were not included in the TS. For example, the following TS valves were not found in SD-12: E-41-F041; E11-F040; E-11-F079A and B; E11-F080A and B. Likewise, there were more than thirty automatic isolation valves listed in SD-12 that were not included in the TS table. Because there were TS valves missing from SD-12 and SD-12 was being used to meet the TS, the operability of the valves was not demonstrated. Likewise, since the plant has more automatic isolation valves in SD-12 than covered by the TS table, the TS table is incomplete.

This TSI was also reviewed by the Diagnostic Evaluation Team during a spring 1989 inspection, and this evaluation confirms the Team's concern. In addition, the licensee filed a Licensee Event Report (LER) (1-89-16) on July 14, 1989 which also confirms the results of this evaluation. Lastly, a violation was identified on this subject in Inspection Report No. 89-34, dated November 30, 1989. The NRC identified violation was not cited because criteria specified in Section V.A. of the NRC Enforcement Policy were satisfied.

An immediate safety concern does not exist because the licensee is presently following the TS. However, the licensee violated the TS when certain valves were not included in SD-12 and the TSI specified following SD-12 instead of the TS.

D. Conclusion

There is no immediate safety concern at this time. There could have been an immediate safety concern for that period of time that certain valves were not included in SD-12. A TS change is under review to, among other things, remove the valve listing from the TS and include the listing in a plant controlled document subject to 10 CFR 50.59.

E. Reference

As stated

F. Principal Contributor:

E. G. Tourigny

EVALUATION OF BRUNSWICK TECHNICAL SPECIFICATION INTERPRETATION

A. Technical Specification Interpretation Request Form - Serial No. 85-04 (Both Units)

The subject of this Technical Specification Interpretation (TSI) is Standby Liquid Control (SLC) system Operability and relationship to Reactor Water Cleanup (RWCU) Operability.

B. Background

Brunswick Technical Specification 3.1.5 requires that the SLC system be operable in conditions 1, 2, and 5. Technical Specification 3.3.2 requires the Reactor Water Cleanup system isolation due to SLC initiation be operable in condition 1, 2, and 3. The licensee's interpretation is RWCU isolation is required in conditions 1, 2, 3, and 5. This interpretation was approved by the Plant Nuclear Safety Committee on May 2, 1985.

The licensee justifies the interpretation as follows. As SLC is required in conditions 1, 2, and 5, instrumentation required to maintain it operable must be in service; therefore, instrumentation required to isolation RWCU or SLC initiation must be operable in conditions 1, 2, 3, and 5.

C. Evaluation

RWCU should be isolated when SLC is actuated because the ion-exchange resins in the RWCU system would remove boron and not give the desired reactivity effect. The licensee is adding a condition when RWCU must isolated if SLC is initiated, namely 5. In this case, the licensee is going beyond the technical specifications requirements.

The staff believes that RWCU isolation should occur when SLC is acutated in Conditions 1, 2, and 5. In the case of standard technical specifications, it should be noted that condition 5 also has the caveat "where a control rod is withdrawn".

Although the 1985 TSI indicated that a TS change was needed, one was not submitted to the NRC.

D. Conclusion

The staff has no objection to the licensee's interpretation. Adding RWCU isolation during condition 5 represents no safety concern.

A breakdown in the licensee's administrative controls appears to have occurred. No TS change was submitted since the TSI was formalized on May 2, 1985.

The staff believes that a TS change is necessary in this case. The licensee should have consistency of conditions between the specifications.

E. Reference

As stated

F. Principal Contributor: E. G. Tourigny

EVALUATION OF BRUNSWICK TECHNICAL SPECIFICATIONS INTERPRETATION

A. Technical Specification Interpretation Request Form - Serial No. 85-10 (Both Units)

The subject of this Technical Specification Interpretation (TSI) is Fire Protection Instruments.

B. Background

TS Table 3.3.5.7-1 specifies for Zone 4 of the AOG Building the following minimum operable instruments: one flame detector, 6 heat detectors, and 6 smoke detectors

The licensee states in the TSI that fire detection instruments specified by Technical Specification (TS) 3.3.5.7-1 for Zone 4 of the Augmented Offgas (AOG) building are not consistent with the instruments actually installed. The minimum instruments operable for Zone 4 of the AOG building are 2 flame detectors, 5 heat detectors, and no smoke detectors. These minimum instruments were specified in a request for a TS change submitted on September 7, 1982 to the NRC. This TS request was unappropriately superseded by later TS change requests to the NRC dated December 17, 1982 and October 17, 1983. These later requests were based on out-dated information, contained a typographical error, and were issued by the NRC as Amendments 66 and 92. A TS change has been initiated to correct the Zone 4 for detection instruments identified by Table 3.3.5.7-1.

This interpretation was approved by the Plant Nuclear Safety Committee on May 24, 1985.

C. Evaluation

The technical specifications should reflect actual plant equipment. A review of UFSAR Table 9.5.1-2, entitled "Detection System Summary", identified ionization, thermal (heat) and flame detectors as being in Zone 4 of the AOG Building. Photoelectric (smoke) detectors are not listed. The type of detection instruments contained in the UFSAR is consistent with the licensee's TSI information; however, the UFSAR does not identify the number of detectors by type.

The licensee's fire hazards analysis in section 9.5.1.5 of the UFSAR stated that the exposures present in the AOG Building are not severe and because of an absence of either train, none could be lost due to a fire. Thus, a fire in the AOG Building, Zone 4 would not inhibit safe shutdown of the plant.

D. Conclusion

The inconsistency between the number of minimum operable fire detection instruments in the AOG Building, Zone 4 does not present an immediate safety concern, because the AOG Building is not needed for safe shutdown. A breakdown in the licensee's administrative controls appears to have occured because no TS change was submitted since TSI was formalized on May 24, 1985. The staff believes that the TS should be corrected. The TSI is correct.

E. Reference

As stated

F. Principal Contributor: E. G. Tourigny

EVALUATION OF BRUNSWICK TECHNICAL SPECIFICATION INTERPRETATION

A. Technical Specification Interpretation Request Form -Serial No. 85-12 (Both Units)

The subject of this Technical Specification Interpretation (TSI) is Reactor Water Clean Up (RWCU) Response Time.

B. Background

The response time for Reactor Water Clean Up differential flow specified by Technical Specification (TS) Table 3.3.2-3, item 3.a is less than or equal to 13 seconds. However, there are 45 second timers in the logic trains and this appears to be an inconsistency, according to plant personnel. The licensees interpretation is that the response time for the RWCU differential flow is less than or equal to 13 seconds and this time does not include the 45 second timer. The licensee's basis is as follows. The RWCU differential flow instrumentation is part of the steamline break protection provided for the RWCU system. The instrumentation compares the inlet and outlet flows to ensure the leakage from the system is below a specified minimum of less than or equal to 53 gallons per minute. The system incorporates a timer to prevent spurious isolation during system evaluations. This timer is set at a value high enough to allow the system evolutions to occur, and below the time used in the GE analysis for a line break on this line.

This TS change was submitted in an admendment application dated February 28, 1988 as supplemented September 20, 1989. The TS change is under review.

The TSI was approved by the Plant Nuclear Safety Committee on July 29, 1985.

C. Evaluation

The staff reviewed UFSAR section 7.3.1.1.6.17 "Cleanup System High Differential Flow." High differential flow in the cleanup system measured between a point immediately outside the primary containment and points downstream from the filter-demineralizers could indicate a break between these points. The automatic closure of the cleanup system isolation valves prevents excessive loss of reactor coolant and release of significant amounts of radioactive material. A break downstream from the filter-demineralizers would be less consequential because of the low radioactivity of the water at that point. The high differential flow isolation trip setting was selected high enough to avoid spurious isolations yet low enough to provide timely detection and isolation. A 45 second time delay is provided to allow the RWCU system to ride through

normal flow transients without a differential flow isolation occurring. This setting is high enough to prevent spurious isolations and low enough to prevent safety problems due to having a High Energy Line Break (HELB) without being isolated within design limits. The UFSAR specifies the flow trip setpoint of ≤ 53 gallons/minute

Based upon the above FSAR statements, it appears that the RWCU response time due to high flow conditions is less than or equal to 45 seconds. Thus the response time of less than or equal to 13 seconds specified in the TS is incorrect or incomplete. The TS change is under review.

D. Conclusion

The TSI does not cause an immediate safety concern because the licensee takes into account the 45 seconds in its analysis of line breaks. In addition, the licensee has submitted a TS change which is under review. The TSI was formalized in July 1985, and it took the licensee more than three years to request the change formally. A breakdown in their administrative controls appears to have taken place.

E. Reference

As stated

F. Principal Contributor: E. G. Tourigny

EVALUATION OF BRUNSWICK TECHNICAL SPECIFICATOIN INTERPRETATION

A. Technical Specification Interpretation Request Form - Serial No. 85-13 (Both Units)

The subject of this Technical Specification Interpretation (TSI) is Control Building Emergency Filtration System (CBEFS) Operability.

B. Background

Brunswick posed the question: Is the CBEFS considered operable when chlorine or fire detectors are out of service or when these detectors are tripped so as to provide their required input to the CBEFS. The licensee believes that the CBEFS is operable under such conditions. The licensee provides the following basis.

The design criterion for the CBEFS is 10 CFR 50, Appendix A, General Design Criteria (GDC) 19. There is no requirement in GDC 19 for either fire or chlorine detection. These sub-systems are a part of the plant specific design of the CBEFS at Brunswick as described in the FSAR; however, they are not required to meet GDC 19.

In the event of an actuation of the fire detection system, the CBEFS is placed in the same operational status as would be required by a high radiation signal; therefore, there is no question of operability. If a fire detector were to fail such that it would not provide an automatic start of CBEFS, system design would still allow an automatic start on a high radiation signal. Therefore, neither a fire detector failure nor initiation can create a problem with CBEFS operability.

In the event of a chlorine detector failure or actuation, the CBEFS would align itself into the recirculation mode of operation. In this mode, normal makeup of outside air (1000 scfm) to the system is isolated. With this loss of makeup air, the reduction in positive pressure within the control room allows an increase in the in-leakage of unfiltered air from approximately 275 scfm (CBEFS running) to approximately 1375 scfm (CBEFS isolated). A Control Room Habitability Evaluation performed on BSEP by NUS (per TMI III.D.3.4) evaluated the dose which would be received by control room personnel during a LOCA with different values of unfiltered in-leakage. This evaluation determined that the dose from radioactivity airborne in the control room will peak and level off at 2.8 rem thyroid and 0.004 rem whole body for unfiltered in-leakage of 100,000 scfm or greater. When combined with the other sources of radiation to the control room, the sum totals are 0.415 rem whole body and 2.8 rem thyroid. Based on this, neither a chlorine detector failure nor a CBEFS isolation from the chlorine detection system will create a problem with CBEFS operability per GDC 19.

This interpretation was approved by the Plant Nuclear Safety Committee on December 5, 1985.

C. Evaluation

The control room habitability system at Brunswick protects control room operators against such postulated releases of radioactive materials, toxic gases, and products of combustion. The single control room at Brunswick services both units.

The licensee incorrectly assumes that GDC 19 only addresses radiation protection of operators. It goes beyond radiation protection concerns. In addition, GDC 3 and 4 applies. Lastly, NUREG-0737, Item III.D.3.4, specifically included toxic gas releases as a scenario control room operators must be protected against.

The licensee stated that in the event of an actuation of the fire detection system, the CBEFS is placed in the same operational status as would be required by a high radiation signal. Therefore, the staff agrees that the CBEFS is operable.

The licensee stated that if a fire detector were to fail such that it would not provide an automatic start of the CBEFS, system design would still allow an automatic start on a high radiation signal. Therefore, the staff agrees that the CBEFS is operable, so far as a radiation type accident, such as a LOCA, is concerned. However, the licensee would have to follow the technical specifications for a failed fire detector or multiple detectors. The licensee would have to make a CBEFS operability determination from a fire protection perspective.

The licensee states that in the event of a chlorine detector failure or actuation, the CBEFS would align itself into the recirculation mode of operation. Therefore, the staff believes that the CBEFS is operable, as far as a radiation type accident, such as LOCA, is concerned. However, the licensee would have to follow the technical specification for a failed chlorine detector or multiple detectors. The licensee would have to make a CBEFS operability determination from a chlorine release perspective.

The licensee determined that a TS change was necessary. One has not been submitted since the TSI was put in effect in December 1985.

D. Conclusion

The licensee's TSI does not present an immediate safety concern. However, the licensee does not have a firm grasp of the applicable GDC's and others Brunswick TS that have a bearing on CBEFS operability. In addition, Detectors failing low versus high should be addressed if the differences are material.

A breakdown in the licensee's administrative control appears to have occurred because no TS change has been submitted since the TSI was formalized on December 5, 1985.

E. References

As stated

F. Principal Contributor: E. G. Tourigny

EVALUATION OF BRUNSWICK TECHNICAL SPECIFICATION INTERPRETATION

A. TECHNICAL SPECIFICATION INTERPRETATION REQUEST FORM - SERIAL NO. 86-01 (BOTH UNITS)

The subject of this Technical Specification interpretation (TSI) is concerned with actions to be taken when more than one control rod scram accumulator is declared inoperable in OPERATIONAL MODES 1 or 2.

B. BACKGROUND

With two or more control rod scram accumulators INOPERABLE in OPERATIONAL MODES 1 and 2, operation is outside the defined boundaries of Brunswick Technical Specification (BTS) 3.1.3.5. Therefore BTS 3.0.3 must be entered. The TSI restates this fact. It also states that to exit the actions of BTS 3.0.3, the number of inoperable accumulators must be reduced to less than or equal to one.

The TSI would allow 3.0.3 to be satisfied by (1) correcting the problem or (2) inserting the associated control rods and either electrically or hydraulically disarming the control rod and taking the ACTIONS required by BTS 3.1.3.1.

The licensee justified the TSI on the basis that inserting and electrically or hydraulically isolating such a control rod removes it from the inoperable accumulator specification and transfers control to the control rod specification (BTS 3.1.3.1).

C. EVALUATION

The current BWR Standard Technical Specifications (STS) ACTIONS for more than one control rod accumulator INOPERABLE (STS 3.1.3.5), specifies that the ACTIONS for INOPERABLE control rods (STS 3.1.3.1) be followed. The intent of the TSI was to institute, administratively, actions which are equivalent to ACTIONS which are allowed by the STS. The ACTIONS in the STS are safe and an amendment in this area to the BTS to be consistent with the STS would likely be approved by the NRC. Therefore, the ACTIONS which the TSI institutes are equivalent to the STS and do not create a safety problem. However, the suggestion of the TSI that BTS 3.0.3 can be met by inserting and disarming the control rods associated with the inoperable accumulators is not acceptable. BTS 3.0.3 can only be met by shutting down or restoring the inoperable equipment to operable status, meeting the remedial measures specified in the LCO's ACTION statements or entering an OPERATIONAL MODE in which the LCO is not applicable.

D. CONCLUSION

The staff has determined that the licensee is misapplying the requirements of BTS 3.0.3 with regards to ways of terminating the actions of the specification.

Although there is no immediate safety concern, a TS change should be submitted to the NRC to bring BTS 3.1.3.5 into conformance with the STS.

E. REFERENCE

As stated

F. PRINCIPAL CONTRIBUTOR

OTSB

EVALUATION OF BRUNSWICK TECHNICAL SPECIFICATION INTERPRETATION

A. <u>Technical Specification Interpretation Request Form - Serial No. 86-02 (Both Units)</u>

The subject of this Technical Specification Interpretation (TSI) is Secondary Containment Integrity Requirement in Condition 5.

B. <u>Background</u>

Technical Specification 3/4.3.2 identifies operability and surveillance requirements for isolation actuation instrumentation. Radiation monitor in the reactor building exhaust are addressed. An action statement is provided if the minimum number of channels are not operable. If one channel is inoperable, it must be repaired within two hours or ACTION 23 must be taken. If both channels are inoperable, ACTION 23 must be taken.

The licensee poses the question: If [radiation monitor] D12-N010A and/or N010B are inoperable in Mode [Condition] 5, must Secondary Containment Integrity be established with the Standby Gas Treatment (SBGT) System operating within one hour (Action 23). The licensee provides the following interpretation.

If the N010A and/or N010B is inoperable in Mode 5, the ACTION which should be taken is "Establish Secondary Containment Integrity with the standby gas treatment system operating within one hour or enter the ACTION statements for 3.6.5.1, 3.6.5.2, and 3.6.6.1 as applicable.

The intent of ACTION statement 23 to T/S 3.3.2 is to ensure that those functions initiated by N010A(B) are established, as the initiating instrumentation is inoperable. These instruments are used to isolate the secondary containment dampers and to start the SBGT's on a high reactor building exhaust radiation signal. Each of the functions required by ACTION 23 [SECONDARY CONTAINMENT INTEGRITY, SBGT's, secondary containment dampers (implied)] have their own unique ACTION requirements, and all are required in the same modes. Should ACTION 23 not be able to be followed (any of the three functions cannot be performed), the implication would be that T/S 3.0.3 must be entered, however, you are already in Cold Shutdown.

ACTION 23 dictates a defined set of functions should N010A(B) be inoperable. If the required ACTION's for those functions (T/S 3.6.5.1, 3.6.5.2, 3.6.6.1) are taken when they are inoperable, then the intent of ACTION 23 is met.

The licensee determined that a TS change was needed. The licensee submitted a TS amendment on February 13, 1989. This TSI was approved by Plant Nuclear Safety Committee on February 13, 1986.

C. Evaluation

Each radiation monitor can isolate the secondary containment dampers and start the SBGT system (both trains). There is only one channel per trip system. The inoperable channel need not be place in the tripped condition as long as it can be restored to operable status within two hours. If if cannot be restored within two hours, secondary containment must be established with the standby gas treatment system operating within one hour. If both are inoperable, ACTION 23 must be followed.

The staff does not agree with the licensee's interpretation. The licensee takes one of two actions: ACTION 23 or enter ACTION statements for 3.6.5.1, 3.6.5.2, and 3.6.1, as applicable. The staff believes that "or" should be replaced with "and". All applicable action statements must be completed (isolation instrumentation, secondary containment integrity, secondary containment isolation dampers, and standby gas treatment system). Since ACTION statement 23 has the shortest time frame for completion, it should be completed prior to completion of the other action statements. The various times associated with the ACTION statements should not be taken sequentially.

The licensee submitted a TS change which is currently under review.

Although the licensee submitted a TS change request, the licensee's TSI has been in effect for a number of years; and a breakdown in administrative controls appears to have taken place.

D. Conclusion

The staff does not believe that an immediate safety concern exists when the licensee takes ACTION 23, but does believes that an immediate safety concern could exist if the licensee takes the other ACTION's while in Condition 5 and ACTION 23 is not also taken while irradiated fuel is being moved. A breakdown in the licensee's administrative controls appears to have occurred because it took the licensee a number of years to request a TS change.

E. Reference

As stated

F. Principal Contributor E. G. Tourigny

EVALUATION OF BRUNSWICK TECHNICAL SPECIFICATION INTERPRETATION

A. TECHNICAL SPECIFICATION INTERPRETATION REQUEST FORM - SERIAL NO. 87-01 (Rev. 1) (Both Units)

The subject of this Technical Specification Interpretation (TSI) is what constitutes "two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system."

B. BACKGROUND

Brunswick Technical Specification (TS) 3.8.1.1 requires "...two physically independent circuits between the offsite transmission network and the onsite Class 1E distribution system..." In order to meet this requirement the TSI states that the following must be maintained for each unit:

1) two operable incoming transmission lines to the switchyard capable of supplying both the Unit Auxiliary Transformer (UAT) and Station Auxiliary Transformer (SAT).

2) an operable distribution system from both the UAT and SAT to Balance of Plant (BOP) buses 1(2)C and 1(2)D.

3) an operable distribution system to supply power from the 1(2)C and 1(2)D buses to the emergency (E) buses.

The license justifies this TSI on the basis of the design information provided in the FSAR describing how Brunswick meets the requirements of GDC 17 and discussions and correspondence with the NRC.

C. EVALUATION/CONCLUSION

The staff has verified through FSAR figure 8.3.1-1 that the requirements as set forth above assure two physically independent circuits to the Class 1E distribution system. Thus the TSI is in conformance with the current staff positions with regards to the configuration aspect of GDC 17. However, as a part of the Diagnostic Team Inspection followup affort, the staff requested by letter dated November 1, 1989, additional information on how Brunswick satisfies the requirements of GDC-17. The staff is waiting for CP&L's response. It should be noted that the above evaluated TSI is a revision to the TSI reviewed by the DET.

D. REFERENCES

As stated

E. PRINCIPAL CONTRIBUTOR

TSB

EVALUATION OF BRUNSWICK TECHNICAL SPECIFICATION INTERPRETATION

A. TECHNICAL SPECIFICATION INTERPRETATION REQUEST FORM - SERIAL NO. 87-02 (BOTH UNITS)

The subject of this Technical Specification Interpretation (TSI) is the action that should be taken if the residual heat removal (RHR) room coolers are inoperable.

B. BACKGROUND

Brunswick Technical Specifications (BTS) do not include a technical specification for the RHR room coolers. The TSI provides the administrative controls for the RHR room coolers in the form of Limiting Conditions for Operation (LCO), Action Statements and Surveillance Requirements.

C. EVALUATION

The RHR room coolers are support systems necessary for the operation of the RHR/LPCI system and decisions about the operability of the room coolers and RHR/LPCI system are the safety responsibility of the licensee. The licensee is allowed to develop and implement administrative controls and procedures for the operation of support systems. In the decision process, the licensee must rely on the definition of operability and the UFSAR. The key question is whether one RHR room cooler is capable of handling the heat load from the design basis accident and maintaining the room temperature within the operating limits for the space which contains both RHR/LPCI subsystems. The NRC staff examined the UFSAR, the TSI, and discussed the system and the RHR/LPCI system space with the resident inspector. The available information is inconsistent; therefore, the NRC is unable to answer this key question. If one room cooler can handle the accident heat load for both RHR/LPCI subsystems, remedial actions like those in the TSI are reasonable. If one room cooler can not handle the accident heat load for both RHR/LPCI subsystems, the remedial actions in the TSI are inadequate because they are not consistent with the actions for an inoperable RHR or LPCI subsystem (BTS 3.5.3.2). The allowed outage time for one inoperable LPCI subsystem is 7 days; the TSI proposes 14 days for one inoperable room cooler. Also, the 7 day allowed outage time for an inoperable LPCI subsystem relies on the operability of the core spray system.

D. CONCLUSIONS

The staff has determined that the licensee needs to evaluate the RHR room cooler design to determine its capabilities to maintain RHR area tempeature within design limits during a DBA. After this determination has been made the TSI should be adjusted to conform to the results of the analysis. This justification should be included in the TSI.

E. REFERENCES

As stated

F. PRINCIPAL CONTRIBUTOR

OTSB

EVALUATION OF BRUNSWICK TECHNICAL SPECIFICATION INTERPRETATION

A. TECHNICAL SPECIFICATION INTERPRETATION REQUEST FORM - SERIAL NO. 87-03 (BOTH UNITS)

The subject of this Technical Specification Interpretation (TSI) is the application of the words "scheduled reduction" in Specification 3.6.6.3.b which requires the oxygen concentration to be within a certain limit whenever thermal power is above 15% of rated thermal power.

B. BACKGROUND

Brunswick Technical Specification (TS) 3.6.6.3 requires the primary containment oxygen concentration be less than 4% by volume whenever thermal power is greater than 15% of rated thermal power (RTP). The LCO is applicable in Mode 1 during the time period of 24 hours after exceeding 15% RTP to 24 hours prior to reducing power to less than 15% RTP.

The TSI states that specification 3.6.6.3.b is intended to allow for the deinerting of the drywell prior to decreasing power to less than 15% RTP or scheduled power reductions requiring drywell entry. This is applicable for both planned and exigent power decreases.

C. EVALUATION/CONCLUSION

The staff has reviewed this position and concurs with it. A TS bases page change was submitted by the licensee on the subject. It is under review.

D. REFERENCES

As stated

E. PRINCIPAL CONTRIBUTOR

OTSB

EVALUATION OF BRUNSWICK TECHNICAL SPECIFICATION INTERPRETATION

A. TECHNICAL SPECIFICATION INTERPRETATION REQUEST FORM - SERIAL NUMBER 88-01 (BOTH UNITS)

The subject of this Technical Specification Interpretation (TSI) is concerned with the timing of the performance of the required surveillances for APRM, IRM, and SRM trip functions upon entering OPERATIONAL CONDITION 2.

B. BACKGROUND

Brunswick Technical Specification Table 4.3.1-1 Items 1a, 1b, and 2a - Note "d" and Table 4.3.4-1 Items 1d, 3a, b, c, d, and 4a, b, c, d - Note "d" require that when changing from OPERATIONAL CONDITION 1 to OPERATIONAL CONDITION 2, the required weekly surveillances specified in the tables should be performed within 12 hours after entering OPERATIONAL CONDITION 2.

The TSI states that Note "d" was intended to require checking these trip functions after being in OPERATIONAL CONDITION 1 for an extended period of time, i.e., greater than seven days; in which case the trip functions should be checked if the intent is to remain in OPERATIONAL CONDITION 2 for any extended period of time.

C. EVALUATION

The surveillance requirements associated with Note "d" state that the APRM, IRM, and SRM trip functions must be checked within 12 hours of entering OPERATIONAL CONDITION 2 from OPERATIONAL CONDITION 1. The intent of the footnote is not to prevent the performance of those surveillances when in OPERATIONAL CONDITION 1 where it is possible to do so. However, the note does not specifically address switching back and forth between OPERATIONAL CONDITION 1 and 2 within a seven day period.

The staff's position is that as long as a component's surveillance has been performed within its scheduled frequency, the surveillances do not have to be redone upon entering OPERATIONAL Conditions where the component is required to be OPERABLE. Therefore, if the surveillance requirement was completed within the last 7 days, it does not have to be performed again upon reentry into MODE 2.

The Staff noted that although the 1988 TSI indicated that a TS change was needed, one was not submitted to the NRC.

D. CONCLUSION

If a surveillance test is performed prior to entering or while in OPERATIONAL CONDITION 1 and is still within its required 7 day surveillance interval, the test need not be performed again "within 12 hours after entering OPERATIONAL CONDITION 2."

Even though the TSI states that a Technical Specification change would be submitted, the staff does not see the need for a technical specification change.

E. REFERENCE

As stated

F. PRINCIPAL CONTRIBUTOR

OTSB

EVALUATION OF BRUNSWICK TECHNICAL SPECIFICATION INTERPRETATION

A. TECHNICAL SPECIFICATION INTERPRETATION REQUEST FORM - SERIAL NO. 89-01 (BOTH UNITS)

The subject of this Technical Specification Interpretation (TSI) is what constitutes a "CORE ALTERATION" as referenced in the definitions.

B. BACKGROUND

The Brunswick Technical Specifications uses the definition of CORE ALTERATION as provided in the STS. The TSI states that the original Brunswick Technical Specifications (BTS), prior to the BTS change in 1977 to the STS, contained a definition for CORE ALTERATION with the following clarification; "Normal control rod movement with the control rod drive hydraulic system is not defined as a core alternation. Normal movement of incore instrumentation is not defined as a core alteration." Since there was no indication that the intent was different, operation at Brunswick continued with the understanding that control rod movement was not considered a CORE ALTERATION.

In addition the TSI classified the following items as a CORE ALTERATION or not a CORE ALTERATION: 1) An SRM or Control Rod repositioned or moved by its normal drive mechanism in its normal operating channel is not a CORE ALTERATION. 2) An SRM or a control blade assembly removed from or relocated in the reactor vessel is a CORE ALTERATION. 3) The addition, removal, or relocation of sources and/or fuel by any mechanism is a CORE ALTERATION. 4) Removal of an LPRM string or an IRM is a CORE ALTERATION; however, withdrawal or insertion of the IRM using its normal drive system is not a CORE ALTERATION. 5) Withdrawal of a control rod with the CRD system and then the removal of the drive unit is not a CORE ALTERATION. 6) Insertion, removal, or relocation of blade guides is a CORE ALTERATION.

C. EVALUATION

In a memorandum from Gus C. Lainas, Assistant Director for Region II Reactor, Division of Reactor Projects I/II, NRR to Albert F. Gibson, Division of Reactor Safety Region II, dated July 7, 1987, NRR stated that based on its review of the Technical Specifications for a number of plants, the NRR staff found that the definition of CORE ALTERATIONS in TSs for more recent plants (such as Susquehanna 2, Fermi 2, River Bend and Perry) and recent license amendments for some plants (such as Hatch 1) includes a statement that normal movements of the SRMs, IRMs, TIPs, or special moveable detectors are not considered CORE ALTERATIONS.

D. CONCLUSION

The staff has determined that the interpretation of CORE ALTERATION provided in the TSI is in conformance with current staff positions as cited in the July 7, 1987 Lainas memorandum.

E. REFERENCE

As stated

F. PRINCIPAL CONTRIBUTOR

OTSB