ENCLOSURE 1

PROPOSED TECHNICAL SPECIFICATION CHANGE

SEQUOYAH NUCLEAR PLANT UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

(TVA-SQN-TS-88-42)

LIST OF AFFECTED PAGES

Unit 1

2-6
B 2-4
3/4 3-5

Unit 2

2-6
B 2-4
3/4 3-5

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES	
13. Steam Generator Water Level--Low-Low	≥ 18% of narrow range instrument span-each steam generator	≥ 17% of narrow range instrument span-each steam generator	R20
14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	< 40% of full steam flow at RATED THERMAL POWER coincident with steam generator water level ≥ 25% of narrow range instrument span--each steam generator	< 42.5% of full steam flow at RATED THERMAL POWER coincident with steam generator water level ≥ 24.0% of narrow range instrument span--each steam generator	
15. Undervoltage-Reactor Coolant Pumps	≥ 5022 volts-each bus	≥ 4739 volts-each bus	R89
16. Underfrequency-Reactor Coolant Pumps	≥ 56.0 Hz - each bus	≥ 55.9 Hz - each bus	
17. Turbine Trip			
A. Low Trip System Pressure	≥ 45 psig	≥ 43 psig	
B. Turbine Stop Valve Closure	≥ 1% open	≥ 1% open	
18. Safety Injection Input from ESF	Not Applicable	Not Applicable	
19. Intermediate Range Neutron Flux - (P-6) Enable Block Source Range Reactor Trip	≥ 1×10^{-5} % of RATED THERMAL POWER ~~≥ 1×10^{-10} amps~~	≥ 6×10^{-6} % of RATED THERMAL POWER ~~≥ 6×10^{-11} amps~~	
20. Power Range Neutron Flux (not P-10) Input to Low Power Reactor Trips Block P-7	< 10% of RATED THERMAL POWER	< 11% of RATED THERMAL POWER	

R89 R89

Range Channels will initiate a reactor trip at ~~a current level proportional to~~ approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels ~~in the accident analyses~~; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

Overtemperature Delta T

The Overtemperature Delta T trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the High and Low Pressure reactor trips. This setpoint includes corrections for axial power distribution, changes in density and heat capacity of water with temperature and dynamic compensation for piping delays from the core to the loop temperature detectors. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table 2.2-1.

Operation with a reactor coolant loop out of service below the 4 loop P-8 setpoint does not require reactor protection system setpoint modification because the P-8 setpoint and associated trip will prevent DNB during 3 loop operation exclusive of the Overtemperature Delta T setpoint. Three loop operation above the 4 loop P-8 setpoint is permissible after resetting the K1, K2 and K3 inputs to the Overtemperature Delta T channels and raising the P-8 setpoint to its 3 loop value. In this mode of operation, the P-8 interlock and trip functions as a High Neutron Flux trip at the reduced power level.

Overpower Delta T

The Overpower Delta T reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature Delta T protection, and provides a backup to the High Neutron Flux trip. The setpoint includes corrections for changes in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. No credit was taken for operation of this trip in the accident

TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the reactor trip system breakers in the closed position and the control rod drive system capable of rod withdrawal, and fuel in the reactor vessel.

**The channel(s) associated with the protective functions derived from the out of service Reactor Coolant Loop shall be placed in the tripped condition.

#The provisions of Specification 3.0.4 are not applicable.

Source Range Outputs ~~High voltage to detector~~ may be disabled ~~de-energized~~ above the P-6 (Block of Source Range Reactor Trip) setpoint.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours and/or open the reactor trip breakers.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours. R51

b. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1.1. R51

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL and the Power Range, Neutron Flux high trip reduced to less than or equal to 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours.

d. The QUADRANT POWER TILT RATIO, as indicated by the remaining three detectors is verified consistent with the normalized symmetric power distribution obtained by using the movable incore detectors in the four pairs of symmetric thimble locations at least once per 12 hours when THERMAL POWER is greater than 75% of RATED THERMAL POWER.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
13. Steam Generator Water Level--Low-Low	≥ 18% of narrow range instrument span-each steam generator	≥ 17% of narrow range instrument span-each steam generator R7
14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	≤ 40% of full steam flow at RATED THERMAL POWER coincident with steam generator water level ≥ 25% of narrow range instrument span--each steam generator	≤ 42.5% of full steam flow at RATED THERMAL POWER coincident with steam generator water level ≥ 24% of narrow range instrument span--each steam generator R76
15. Undervoltage-Reactor Coolant Pumps	≥ 5022 volts-each bus	≥ 4739 volts-each bus
16. Underfrequency-Reactor Coolant Pumps	≥ 56 Hz - each bus	≥ 55.9 Hz - each bus
17. Turbine Trip		
A. Low Trip System Pressure	≥ 45 psig	≥ 43 psig
B. Turbine Stop Valve Closure	≥ 1% open	> 1% open
18. Safety Injection Input from ESF	Not Applicable	Not Applicable
19. Intermediate Range Neutron Flux, P-6, Enable Block Source Range Reactor Trip	≥ 1×10^{-5} % of RATED THERMAL POWER ~~≥ 1 x 10-10 amps~~	≥ 6×10^{-6} % of RATED THERMAL POWER ~~≥ 6 x 10-11 amps~~
20. Power Range Neutron Flux (not P-10) Input to Low Power Reactor Trips Block P-7	< 10% of RATED THERMAL POWER	≤ 11% of RATED THERMAL POWER

R76 R76

Intermediate and Source Range, Nuclear Flux (Continued)

Range Channels will initiate a reactor trip at ~~at a current level proportional to~~ approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

Overtemperature ΔT

The Overtemperature delta T trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the High and Low Pressure reactor trips. This setpoint includes corrections for axial power distribution, changes in density and heat capacity of water with temperature and dynamic compensation for piping delays from the core to the loop temperature detectors. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table 2.2-1.

Operation with a reactor coolant loop out of service below the 4 loop P-8 setpoint does not require reactor protection system set point modification because the P-8 setpoint and associated trip will prevent DNB during 3 loop operation exclusive of the Overtemperature delta T setpoint. Three loop operation above the 4 loop P-8 setpoint is permissible after resetting the K1, K2, and K3 inputs to the Overtemperature delta T channels and raising the P-8 setpoint to its 3 loop value. In this mode of operation, the P-8 interlock and trip functions as a High Neutron Flux trip at the reduced power level.

Overpower ΔT

The Overpower delta T reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature delta T protection, and provides a backup to the High Neutron Flux trip. The setpoint includes corrections for changes in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the reactor trip system breakers in the closed position, the control rod drive system capable of rod withdrawal, and fuel in the reactor vessel.

**The channel(s) associated with the protective functions derived from the out of service Reactor Coolant Loop shall be placed in the tripped condition.

#The provisions of Specification 3.0.4 are not applicable.

##~~High voltage to detector~~ Source Range Ckts may be ~~de-energized~~ disabled above the P-6 (Block of Source Range Reactor Trip) setpoint.

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours and/or open the reactor trip breakers.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours. R39

b. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1.1. R39

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range, Neutron Flux trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours.

d. The QUADRANT POWER TILT RATIO, as indicated by the remaining three detectors, is verified consistent with the normalized symmetric power distribution obtained by using the movable incore detectors in the four pairs of symmetric thimble locations at least once per 12 hours when THERMAL POWER is greater than 75% of RATED THERMAL POWER.

ENCLOSURE 2

PROPOSED TECHNICAL SPECIFICATION CHANGE

SEQUOYAH NUCLEAR PLANT UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

(TVA-SQN-TS-88-42)

DESCRIPTION AND JUSTIFICATION FOR
MODIFICATION OF THE TRIP SETPOINT AND ALLOWABLE
VALUE UNITS FOR THE INTERMEDIATE RANGE NUCLEAR
FLUX DETECTOR AND CHANGES TO THE APPLICABILITY
REQUIREMENTS FOR THE SOURCE RANGE
NUCLEAR FLUX DETECTOR

ENCLOSURE 2

Description of Change

Tennessee Valley Authority proposes to modify the Sequoyah Nuclear Plant (SQN) Units 1 and 2 technical specifications (TSs) to revise the trip setpoint and allowable value units for the intermediate range (IR) nuclear flux detector and to revise the applicability requirements for the source range (SR) nuclear flux detector.

Reason for Change

TVA is replacing the SR and IR neutron monitors as part of the equipment upgrade to comply with Regulatory Guide 1.97 as required by SQN License Conditions 2.C.24 (Unit 1) and 2.C.14 (Unit 2). The new SR/IR monitor is a fission chamber design manufactured by Gamma Metrics. This design does not require high-voltage deenergization as part of the normal SR detector operation. Consequently, the footnote (##) for Table 3.3-1 is being revised to change the high-voltage deenergization wording to say that SR outputs may be disabled. The new IR monitor uses a signal that is in units of relative power. Consequently, the trip setpoint and allowable value units are being changed in Table 2.2-1. Because the new IR detector does not provide output in terms of current, the bases to Section 2.2 are also being revised to delete references to IR detector current signals that are proportional to power levels.

Justification for Change

The new Gamma Metrics SR/IR detectors are being installed to achieve compliance with Regulatory Guide 1.97. The new detectors are Class-1E equipment that is seismically and environmentally qualified.

The new SR equipment is compatible with the rest of the nuclear instrumentation and reactor protection system; however, it includes two improvements over the present design. First, the electronic equipment automatically decreases the high flux at shutdown alarm after a reactor trip until the neutron flux stabilizes. Currently, this function is performed manually as described in the Final Safety Analysis Report, Section 15.2.4.2. Second, the new SR/IR detector does not have to be deenergized at higher power levels. Above the P-6 setpoint, the SR detector output signal is blocked from the reactor trip logic. The SR/IR detector assemblies will remain energized during the full range of power operation. Consequently, the table notation in Table 3.3-1 regarding high-voltage deenergization of the SR detectors has been revised to clarify the wording regarding this feature.

The new IR equipment is compatible with the rest of the nuclear instrumentation and reactor protection system except that the output signal is in units of relative power rather than amperes (A). The P-6 setpoint and allowable value listed in Table 2.2-1 are currently listed in units of A. TVA has performed a calculation to determine the relative power values corresponding to the present trip setpoint and allowable value. A relationship between reactor power and detector current was established using start-up test data from several power levels between 5 and 90 percent power. This relationship was then used to convert the trip setpoint to a relative power value. The computed value was rounded to the next conservative decade for ease of calculation. A corresponding allowable value was then calculated using the previously established setpoint and current-power relationship. Finally, the overlap between the SR/IR detector ranges was checked to ensure sufficient margin between the P-6 setpoint and the SR trip setpoint. It is important to note that the actual setpoint is not changed; only the engineering units have changed. A copy of the TVA calculation is included as an attachment to this enclosure.

In summary, two administrative changes are proposed to support the installation of the Gamma Metrics SR/IR assembly. The first involves the revision of a table notation that is no longer applicable to the design of the new SR detectors. The second involves a change in engineering units for the P-6 setpoint that results from the difference in output signals from the IR detectors.

Environmental Impact Evaluation

The proposed revision involves an administrative change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR 20 and changes to the surveillance requirements. TVA has determined that the proposed change involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite and that there is no significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed change meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement nor environmental assessment needs to be prepared in connection with the issuance of the amendment.

ATTACHMENT

1. TVA Calculation, "Intermediate Range Neutron Flux P-6 Setpoint," Revision 1

2. Safety Evaluation, Revision 3

TVA 10697 (DNE 6-86)

DNE CALCULATIONS **QA Record** I EC - 116

TITLE: INTERMEDIATE RANGE NEUTRON FLUX P-6 SETPOINT		PLANT/UNIT: SQN U 1 & 2
PREPARING ORGANIZATION: UEEC / EEB	KEY NOUNS (Consult RIMS DESCRIPTORS LIST): NEUTRON FLUX, FLUX DETECTORS	
BRANCH/PROJECT IDENTIFIERS: 1,2-XE-92-1	Each time these calculations are issued, preparers must ensure that the original (RO) RIMS accession number is filled in.	

Rev	(for RIMS' use)	RIMS accession number
R0	881117B0001	B25 881110 803
R1	INFO	B25 881117 808
R_	ONLY	
R_		

APPLICABLE DESIGN DOCUMENT(S): SQN-DC-V-27.8 R2

SAR SECTION(S): TABLE 14.1-2 | UNID SYSTEM(S): 92

	Revision 0	R1	R2	R3
ECN No. (or indicate Not Applicable)	L6186	N/A ~~L6186~~ 11/17/88		
Prepared	W.A. Mueller	W.A. Mueller		
Checked	Chandan Balik	Chandan Balik		
Reviewed	F.H. Rim	C. Gerizitis		
Approved	M.R. Sedlacik			
Date	11-10-88	11-17-88		
List all pages added by this revision.				
List all pages deleted by this revision.				
List all pages changed by this revision.				

Use form TVA 10534 if more space required

Safety-related? Yes ☒ No ☐

Statement of Problem

JUSTIFY THE CHANGING OF THE INTERMEDIATE RANGE NEUTRON FLUX, P-6, ENABLE BLOCK SOURCE RANGE REACTOR TRIP SETPOINT ENGINEERING UNITS.

Abstract

These calculations contain an unverified assumption(s) that must be verified later. Yes ☐ No ☒

FSAR COMPLIANCE REVIEW M.R. Sedlacik

CALCULATIONS WERE PERFORMED TO JUSTIFY CHANGING THE ENGINEERING UNITS IN THE TECH SPEC. FOR THE INTERMEDIATE RANGE NEUTRON FLUX P-6 ENABLE BLOCK FROM CURRENT (AMPS) TO % OF RATED THERMAL POWER

R1 FSAR COMPLIANCE REVIEW

☐ Microfilm and store calculations in RIMS Service Center.

☐ Microfilm and return calculations to: Microfilm and destroy. ☐

Address

cc: RIMS, SL 26 C-K

TVA

REVISION LOG

1,2-XE-92-1

Title: INTERMEDIATE RANGE NEUTRON FLUX P-6 SETPOINT

Revision No.	DESCRIPTION OF REVISION	Date Approved
0 -	INITIAL ISSUE. THIS CALCULATION CONTAINS 18 PAGES.	11-10-88
1	INCORPORATED COMMENTS FROM LICENSING CHANGED PAGES 1, 4, 8, 9 & 10 ADDED PAGES, TWO INDEPENDENT REVIEW FORMS REV 1, PAGES - 2A. & ATTACHMENT, PAGES 14A THRU 14E THIS CALCULATION CONTAINS 26 PAGES	11-17-88

CALCULATION DESIGN VERIFICATION (INDEPENDENT REVIEW) FORM

1,2-XE-92-1 — Calculation No.

Rev 0 — Revision

Method of design verification (independent review) used (check method used):

1. Design Review ____________
2. Alternate Calculation ____________
3. Qualification Test ____________

Justification (explain below):

Method 1: In the design review method, justify the technical adequacy of the calculation and explain how the adequacy was verified (calculation is similar to another, based on accepted handbook methods, appropriate sensitivity studies included for confidence, etc.).

Method 2: In the alternate calculation method, identify the pages where the alternate calculation has been included in the calculation package and explain why this method is adequate.

Method 3: In the qualification test method, identify the QA documented source(s) where testing adequately demonstrates the adequacy of this calculation and explain.

Reference documents Attached

① Westinghouse Functional Requirement Document / Page 8

② Westinghouse PLS Document Rev 4 dated MAY 1986

Ⓐ P-6 Permissive Page 17

Ⓑ Source Range Reactor Trip Page 11

③ Westinghouse Figure 10.1-1 Neutron Detectors and Ranges of Operation

After Reviewing the Documents Listed Above in my engineering judgement this calculation is technically adequate.

(Time 5:45PM) Wednesday — Steve Stork — Design Verifier (Independent Reviewer) — 11/09/88 Date

CALCULATION DESIGN VERIFICATION (INDEPENDENT REVIEW) FORM

1,2-XE-92-1 — Calculation No.

1 — Revision

Method of design verification (independent review) used (check method used):

1. Design Review ✓
2. Alternate Calculation ______
3. Qualification Test ______

Justification (explain below):

Method 1: In the design review method, justify the technical adequacy of the calculation and explain how the adequacy was verified (calculation is similar to another, based on accepted handbook methods, appropriate sensitivity studies included for confidence, etc.).

Method 2: In the alternate calculation method, identify the pages where the alternate calculation has been included in the calculation package and explain why this method is adequate.

Method 3: In the qualification test method, identify the QA documented source(s) where testing adequately demonstrates the adequacy of this calculation and explain.

Rev 1 changes only which are for clarification purposes

Daniel L Bobbington — Design Verifier (Independent Reviewer)

11-16-88 — Date

CALCULATION DESIGN VERIFICATION (INDEPENDENT REVIEW) FORM

1,2 - XE-92 -1
Calculation No.

1
Revision

Method of design verification (independent review) used (check method used):

1. Design Review ✓
2. Alternate Calculation ______
3. Qualification Test ______

Justification (explain below):

Method 1: In the design review method, justify the technical adequacy of the calculation and explain how the adequacy was verified (calculation is similar to another, based on accepted handbook methods, appropriate sensitivity studies included for confidence, etc.).

Method 2: In the alternate calculation method, identify the pages where the alternate calculation has been included in the calculation package and explain why this method is adequate.

Method 3: In the qualification test method, identify the QA documented source(s) where testing adequately demonstrates the adequacy of this calculation and explain.

REVIEWED FOR REVISION 1 OF THIS CALCULATION WHICH INCORPORATED EDITORIAL CHANGES AND FOUND ACCEPTABLE

Blair
Design Verifier
(Independent Reviewer)

11/16/85
Date

CALCULATION DESIGN VERIFICATION (INDEPENDENT REVIEW) FORM

J,Z-XE-92-1 — Calculation No.

0 — Revision

Method of design verification (independent review) used (check method used):

1. Design Review ✓
2. Alternate Calculation ____
3. Qualification Test ____

Justification (explain below):

Method 1: In the design review method, justify the technical adequacy of the calculation and explain how the adequacy was verified (calculation is similar to another, based on accepted handbook methods, appropriate sensitivity studies included for confidence, etc.).

Method 2: In the alternate calculation method, identify the pages where the alternate calculation has been included in the calculation package and explain why this method is adequate.

Method 3: In the qualification test method, identify the QA documented source(s) where testing adequately demonstrates the adequacy of this calculation and explain.

THIS CALCULATION IS REVIEWED FOR DESIGN INPUT DATA AND TECHNICAL ADEQUACY AND FOUND ACCEPTABLE

[signature] — Design Verifier (Independent Reviewer)

11/1/88 — Date

GENERAL COMPUTATION SHEET

DISCIPLINE

United Engineers & Constructors

NAME OF COMPANY: TVA - SQN UNIT/S 1 & 2

SUBJECT: INTERMEDIATE RANGE NEUTRON FLUX P-6 SET POINT

CALC. LETTER NO.		REV.	COMP. BY	CHECKED
PRELIM		0	DATE 11/9/88	DATE
FINAL	1,2-78-92-1			
VOID				
SHEET 1 OF 14		1	DATE 11/16/88	DATE 11/16/88
JO 8806.129				

TABLE OF CONTENTS

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United Engineers & Constructors
A Raytheon Company

CALCULATION CONTROL SHEET

SHEET 2 OF 14

CALCULATION SET NO.	1,2 - XE - 92 - 1
PRELIMINARY	
FINAL	✓
VOID	

PROJECT TITLE: SQN

DISCIPLINE: DNE - EEB

STRUCTURE OR SYSTEM: EXCORE DETECTORS

SUBJECT: INTERMEDIATE RANGE NEUTRON FLUX P-6 SETPOINT

DESIGN CLASSIFICATION: CLASS 1E

☒ Safety Related
☐ Non - Safety Related

STARTED BY: W. A. MUELLER — DATE: 11/9/88

AUTHORIZED BY: F.M. RIVA — DATE: 11/9/88

CHECKED BY: FMR for CHANDRAN BALA — DATE: 11/9/88

PROBLEM STATEMENT

JUSTIFY THE CHANGING OF THE INTERMEDIATE RANGE NEUTRON FLUX, P-6, ENABLE BLOCK SOURCE RANGE REACTOR TRIP SETPOINT ENGINEERING UNITS.

DESIGN BASIS AND ASSUMPTIONS

SEE SECTION 2.0

ADMINISTRATIVE CLOSE OUT

TOTAL NUMBER OF SET COMPUTATION SHEETS: ~~14~~ 20

LATEST REVISION OF CALC. SET: ~~0~~ 1

DATE NOTED: 11/9/88

CALCULATION FINISHED BY: (W.A. MUELLER) (Signature) — DATE: 11/9/88

CALCULATION RELEASED BY: (F.M. RIVA) (Supervisor - Design / Administrative) — DATE: 11/9/88

CONCURRED BY: (F.M. RIVA) (Manager (Staff) or SDE) — DATE: 11/9/88

Form 5605 Rev 12/16/83

United Engineers & Constructors
A Raytheon Company

CALCULATION REVISION CONTROL SHEET

Sheet 2A of 14

CALCULATION SET NO	
PRELIMINARY	
FINAL	✓
VOID	

PROJECT TITLE: SQN DISCIPLINE: DNE - EEB

REVISION NO. 1

STRUCTURE OR SYSTEM: EXCORE DETECTORS

SUBJECT: INTERMEDIATE RANGE NEUTRON FLUX P-6 SETPOINT

DESIGN CLASSIFICATION: CLASS 1E ☒ Safety Related ☐ Non-Safety Related

REASON FOR REVISION: ADDITION OF COMMENTS BY LICENSING

REVISION STARTED BY: (W.A. MUELLER) Signature DATE 11/16/88

REVISION AUTHORIZED BY: DATE 11/16/88

CALCULATION CONTROL SHEET NOTED FOR NEW REVISION BY: B Pai (B PAI) 11-16-88 Signature

PROBLEM STATEMENT

SEE CALC CONTROL SHEET.

DESIGN BASIS

SEE SECTION 2.0

ADMINISTRATIVE CLOSE OUT

TOTAL SHEETS THIS REVISION: 13 Quantity

SHEET NOS.: REVISED: 1, 4, 8, 9, 10 ADDED: (2) R.V. FORMS, 2A, 14A THRU 14E
List Sheets Revised Added Deleted

REVISION FINISHED BY: (W.A. MUELLER) Signature DATE 11/16/88

REVISION RELEASED BY: F. RIVA Supervisor - Design / Administrative DATE 11/16/88

CONCURRED BY: F. RIVA Manager (Staff) or SDE DATE 11/16/88

United Engineers & Constructors
A Raytheon Company

CALCULATION SUMMARY & REFERENCE SHEET

SHEET 3 OF 14

CALCULATION SET NO	1,2-NE-RCH-1
PRELIMINARY	
FINAL	✓
VOID	
SHEET 3 OF 14	
J.O.	3806.129

REV	COMP BY / DATE	CHK'D BY / DATE
0	[illegible]	CB

PROJECT TITLE: SQN   DISCIPLINE: DNE - EEB

STRUCTURE OR SYSTEM: Excore Detectors

SUBJECT: Intermediate Range Neutron Flux P-6 Setpoint

DESIGN CLASSIFICATION: Class 1E   ☒ Safety Related ☐ Non-Safety Related

SUMMARY / CONCLUSIONS

See Section 7.0

REFERENCES: (SPECIFICATIONS, DRAWINGS, CODES, CALCULATION SETS, TEXTS, REPORTS, COMPUTER DATA, PSAR ETC.)

See Section 3.0

GENERAL COMPUTATION SHEET

(DISCIPLINE)

United Engineers & Constructors
A Raytheon Company

NAME OF COMPANY TVA - SQN UNIT/S 1 & 2

SUBJECT INTERMEDIATE RANGE NEUTRON FLUX P-6 SET POINT

CALC SET NO		REV	COMP BY	CHK'D BY
PRELIM				
FINAL	1,2-XG-02-1	0	DATE 11/4/85	DATE
VOID				
SHEET	4 OF 14	1	DATE 11/16/88	DATE 11/16/88
JO	8806.129			

FSAR COMPLIANCE REVIEW

A REVIEW OF TABLE 14.1-2 INDICATES NO DISCREPANCIES BETWEEN THIS CALCULATION AND THE FSAR.

FSAR COMPLIANCE REVIEW R1

THE EDITORIAL CHANGES IN THIS REVISION DO NOT IMPACT ANY PORTION OF THE FSAR NOR DO THEY CHANGE ANY PREVIOUS REVIEW. R1

GENERAL COMPUTATION SHEET

(DISCIPLINE)

United Engineers & Constructors
A Raytheon Company

NAME OF COMPANY: TVA - SQN   UNIT/S: 1&2

SUBJECT: Intermediate Range Neutron Flux P-6 Set Point

CALC SET NO	REV	COMP BY	CHK'D BY
PRELIM			
FINAL 1,2-XE-92-1	0	DATE 11/9/85	DATE 11/4/85
VOID			
SHEET 5 OF 14			
JO 8806.129		DATE	DATE

1.0 Purpose

Justification for changing the Intermediate Range Neutron Flux, P-6, enable Block Source Range Reactor Trip Setpoint Engineering Units.

2.0 Technical Requirements

P-6 is a protection interlock derived from the intermediate range Nuclear Instrumentation System (NIS). With reactor power above the P-6 setpoint the source range NIS reactor trip may be blocked to allow continued power escalation. Therefore, the setpoint for P-6 must be set sufficiently below the source range trip to give the operators time to actuate the block and at the same time be above the minimum usable signal in the intermediate range NIS.

The present Westinghouse Design utilizes the following Methodology of establishing the P-6 setpoint for the intermediate range.

> The source range reactor trip is set at 10^5 counts/sec in the source range NIS which corresponds to about 2×10^{-9} amps in the intermediate range NIS. The process range of the intermediate range NIS is from 10^{-11} amps to 10^{-3} amps with the lower end of the range corresponding to about 6×10^2 counts/sec in the source range NIS. This leaves a range of 10^{-11} amps to 2×10^{-9} amps in which to set P-6. Midway in this range (10^{-10} amps, the current P-6 setpoint) provides a little more than 1 decade below the source trip to allow the operators to block the trip and 1 decade above the lower end of the intermediate range NIS process range to achieve a reasonably good signal.

Determination of the Sequoyah Plant specific setting of the power permissive P-6 setpoint can be calculated by measuring the detector current for different reactor levels at or below 75% power & interpolating for the power level AT 10^{-10} amps.

7.0 Conclusions

The Gamma Metrics Design to be installed provides a process range of 10^{-8} to 200% rated thermal power. This provides an additional two decades of overlap with the source range (10^{-8} to 10^{-6} % RTP.)

Therefore, based on the following calculations a value of 1×10^{-5} % RTP shall be used as the trip setpoint which is functionally equivalent to 10^{-10} amps. The values calculated for the P-6 setpoint (using Plant Specific Data) are in good agreement with the 1×10^{-5} RTP setpoint. This setpoint value provides adequate margin below the source range trip to give the operators time to ACTUATE a trip BLOCK and at the same time ENSURE a conservative signal overlap of the intermediate range drawer (3 decades).

(DISCIPLINE)

United Engineers & Constructors
A Raytheon Company

NAME OF COMPANY TVA-SQN UNIT/S 1 & 2

SUBJECT INTERMEDIATE RANGE NEUTRON FLUX P-6 SET POINT

CALC SET NO		REV	COMP BY	CHK'D BY
PRELIM				
FINAL	1,2-XE-92-1 ✓	0	DATE 11/9/93	DATE 11/9/93
VOID				
SHEET	6 OF 14		DATE	DATE
JO	8806129			

SECTION 3.0

SOURCE OF DESIGN INPUT INFORMATION (REFERENCES)

REF #	ATT #	REFERENCE (RIMS#)
1	1	SQNP SU-8.5.1 - Start Up Data Sheets for 10%, 29%, 50% & 75% Power (4 Pages)

(DISCIPLINE)

United Engineers & Constructors
A Raytheon Company

	CALC SET NO	COMP BY	CHKD BY
PRELIM			
FINAL	42-XE-92-1	DATE 11/9/88	DATE 11/9/88
VOID			
SHEET 7 OF 14		DATE	DATE
JO 3806-129			

NAME OF COMPANY TVA-SQN UNIT/S 1 & 2

SUBJECT INTERMEDIATE RANGE NEUTRON FLUX P-6 SET POINT

4.0 DATA TAKEN FROM SU-8.6.1 (2 of 2)

ACTUAL VALUE (AMPS)			AVERAGE VALUE (AMPS)
CHAN N35	3.6×10^{-5}	@ 10% POWER	3.5×10^{-5}
CHAN N36	3.4×10^{-5}		
CHAN N35	1.09×10^{-4}	@ 29% POWER	1.12×10^{-4}
CHAN N36	1.15×10^{-4}		
CHAN N35	1.66×10^{-4}	@ 50% POWER	1.695×10^{-4}
CHAN N36	1.73×10^{-4}		
CHAN N35	2.549×10^{-4}	@ 75% POWER	2.6×10^{-4}
CHAN N36	2.65×10^{-4}		

GENERAL COMPUTATION SHEET

(DISCIPLINE)

United Engineers & Constructors

NAME OF COMPANY: TUA - SQN UNIT/S: 1&2

SUBJECT: INTERMEDIATE RANGE NEUTRON FLUX P-6 SET POINT

CALC SET NO		REV	COMP BY	CHK'D BY
PRELIM				
FINAL	1/2-XE-42-1 ✓	0	DATE 11/9/82	DATE 11/10/82
VOID				
SHEET	8 OF 14	1	DATE 11/16/82	DATE 11/16/82
JO	8806-129			

SECTION 5.0

POWER PERMISSIVE P-6 SETPOINT

10% POWER

AVERAGE = 3.5×10^{-5} AMPS

$$\frac{x}{10\%} = \frac{10^{-10}\text{ AMPS}}{3.5 \times 10^{-5}\text{ AMPS}}$$

$x = 2.857 \times 10^{-5}$ % POWER

29% POWER

AVERAGE = 1.12×10^{-4}

$$\frac{x}{29\%} = \frac{10^{-10}\text{ AMPS}}{1.12 \times 10^{-4}\text{ AMPS}}$$

$x = 2.589 \times 10^{-5}$ % POWER

50% POWER

AVERAGE = 1.695×10^{-4} AMPS

$$\frac{x}{50\%} = \frac{10^{-10}\text{ AMPS}}{1.695 \times 10^{-4}\text{ AMPS}}$$

$x = 2.949 \times 10^{-5}$ % POWER

75% POWER

AVERAGE = 2.6×10^{-4} AMPS

$$\frac{x}{75\%} = \frac{10^{-10}\text{ AMPS}}{2.6 \times 10^{-4}\text{ AMPS}}$$

$x = 2.885 \times 10^{-5}$ % POWER

∴ THE P-6 SETPOINT OF 10^{-10} AMPS IS FUNCTIONALLY EQUIVALENT TO 1×10^{-5} % RTP. | R1

~~∴ A P-6 SETPOINT OF 1×10^{-10} AMPS = 1×10^{-5} % RTP~~
~~IS A CONSERVATIVE ROUNDING~~

COMPUTATION SHEET

(DISCIPLINE)

United Engineers & Constructors
A Raytheon Company

NAME OF COMPANY: TVA - SQN   UNIT/S: 1&2

SUBJECT: INTERMEDIATE RANGE NEUTRON FLUX P-6 SET POINT

CALC SET NO		REV	COMP BY	CHK'D
PRELIM				
FINAL	12-XE-92-4	0		
VOID			DATE 11/9/88	DATE 11/10/8
SHEET 9 OF 14		1		
JO 8806.129			DATE 11/16/88	DATE 11/16/88

SECTION 5.0 (CONT'D)

ALLOWABLE VALUE

INTERPOLATION MAY BE USED BECAUSE IR CURRENT AND POWER ARE LINEAR ~~LOGARITHMIC~~ LOGARITHMIC FUNCTIONS THAT | R1
RESPOND DIRECTLY AND PROPORTIONALLY WITH EACH OTHER.
THE (x) % RTP THAT IS EQUIVALENT TO 6×10^{-11} AMPS IS DETERMINED AS FOLLOWS:

$$\text{GIVEN } 1 \times 10^{-5}\ \%\text{RTP} = 10^{-10}\ \text{AMPS}$$

$$\therefore \frac{x}{1 \times 10^{-5}\ \%\text{RTP}} = \frac{6 \times 10^{-11}\ \text{AMPS}}{10^{-10}\ \text{AMPS}}$$

$$x = \frac{(1 \times 10^{-5}\ \%\text{RTP})(6 \times 10^{-11}\ \text{AMPS})}{1 \times 10^{-10}\ \text{AMPS}}$$

$$x = \frac{6 \times 10^{-16}\ \text{RTP}}{1 \times 10^{-10}}$$

$$x = 6 \times 10^{-6}\ \%\text{RTP}$$

GENERAL COMPUTATION SHEET

(DISCIPLINE)

United Engineers & Constructors
A Raytheon Company

CALC SET NO		REV	COMP BY	CHKD BY
PRELIM		0		
FINAL	1,2-KE-92-1		DATE 11/9/88	DATE 11/10/88
VOID				
SHEET 10 OF 14		1		
JO 8806.129			DATE 11/16/88	DATE 11/16/88

NAME OF COMPANY: TVA - SQN UNIT/S 1&2

SUBJECT: INTERMEDIATE RANGE NEUTRON FLUX P-6 SET POINT.

SECTION 5.0 (CONT'D)

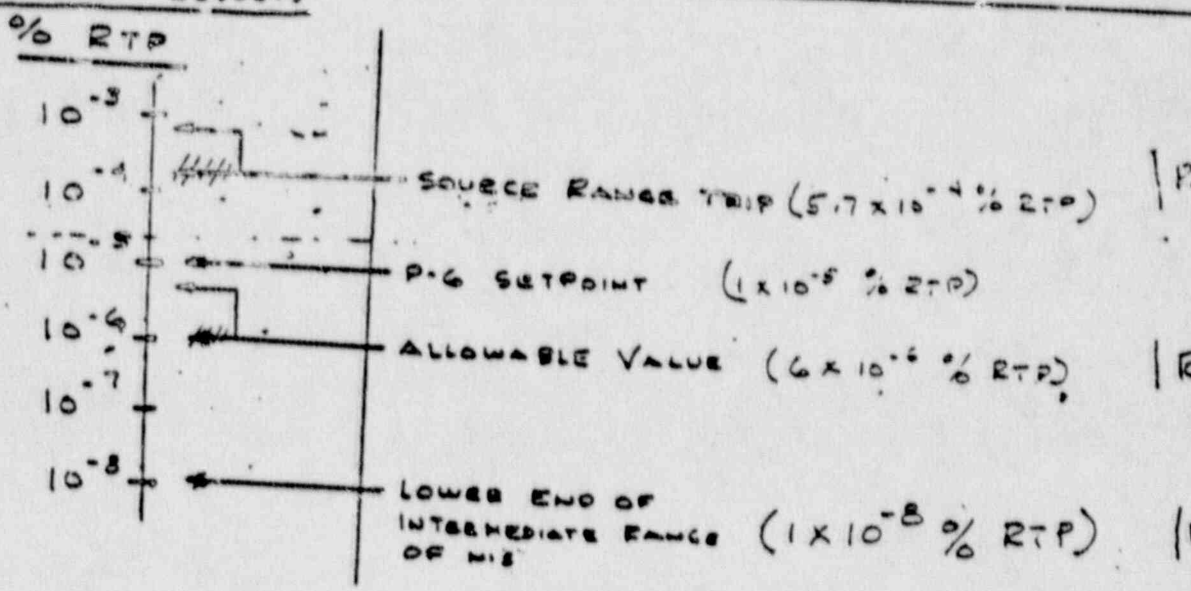
TO BE ABLE TO SHOW THESE BLOCK & TRIP SETPOINTS ON THE SAME SCALE FOR COMPARASON PURPOSES WE CALCULATE THE ACTUAL SOURCE RANGE TRIP IN % RTP BY PERFORMING THE FOLLOWING: | R

SOURCE RANGE TRIP CALCULATION

$$\frac{x \text{ \% PWR } (10^5 \text{ CPS})}{10 \text{ \% PWR}} = \frac{2 \times 10^{-9} \text{ AMPS}}{3.5 \times 10^{-5} \text{ AMPS}}$$

$$\therefore 10^5 \text{ CPS} \approx 5.7 \times 10^{-4} \text{ \% PWR}$$

6.0 SUMMARY OF RESULTS



SQNP
SU-8.5.1 - Units 1 & 2
Data Sheet 1
Page 1 of 1
Rev. 1

Date 10/3/80 Time 0510 Unit 1

INTERMEDIATE RANGE

Channel N35

Main Control Board	2.9 x10-5 amps
NI Drawer	3.6 x10-5 amps

Channel N36

Main Control Board	3.0 x10-5 amps
NI Drawer	3.4 x10-5 amps

POWER RANGE

Channel N41

Main Control Board	11 %
NI Drawer	9.6 %

Channel N42

Main Control Board	9.5 %
NI Drawer	9.4 %

Channel N43

Main Control Board	10 %
NI Drawer	9.5 %

10/3/80

Channel N44

Main Control Board	9.0 %
NI Drawer	9.5 %

Remarks:

Data By G.L. Supt Ed / 10/3/80

Checked By R.W. Fortenberry / 10/3/80

SQHP
SC-3.5.1 - Units 1 & 2
Data Sheet 2
Page 1 of 1
Rev. 1

Date 10/31/80 Power 29% Unit 1

Item No.	Parameter	Test Point	Volts at (Time) 1435	Volts at (Time) 1500	Volts at (Time) 1515	Average Volts	Scaling Factor	Value	Units	Value
1	N-35	TP3 to TP4	8.784	8.800	8.809	8.798	$10^{(.8V-11)}$ =	1.09×10^{-4}	amps	
2	N-36	TP3 to TP4	8.819	8.822	8.832	8.824	$10^{(.8V-11)}$ =	1.16×10^{-4}	amps	
3	N-41 Det. A	Back of meter	.0995	.0986	.1020	.100	$10^{-3}V$ =	1×10^{-4}	uA	100
4	N-41 Det. B	Back of meter	.1472	.1415	.1421	.144	$10^{-3}V$ =	1.44×10^{-4}	uA	144
5	N-41 Power	TP306 to TP305	2.509	2.432	2.473	2.471	12V =	29.6	%	
6	N-42 Det. A	Back of meter	.1054	.1023	.1057	.1045	$10^{-3}V$ =	1.045×10^{-4}	uA	104.5
7	N-42 Det. B	Back of meter	.1582	.1508	.1514	.1535	$10^{-3}V$ =	1.535×10^{-4}	A	153.5
8	N-42 Power	TP306 to TP305	2.537	2.469	2.474	2.493	12V =	29.9	%	
9	N-43 Det. A	Back of meter	.1031	.1022	.1035	.1029	$10^{-3}V$ =	1.029×10^{-4}	A	102.9
10	N-43 Det. B	Back of meter	.1494	.1444	.1431	.1456	$10^{-3}V$ =	1.456×10^{-4}	A	145.6
11	N-43 Power	TP306 to TP305	2.530	~~[illegible]~~ 2.527	2.475	2.510	12V =	30.1	%	
12	N-44 Det. A	Back of meter	.1024	.1047	.1029	.1033	$10^{-3}V$ =	1.033×10^{-4}	A	103.3
13	N-44 Det. B	Back of meter	.1525	.1507	.1463	.1498	$10^{-3}V$ =	1.498×10^{-4}	A	149.8
14	N-44 Power	TP306 to TP305	2.534	2.526	2.464	2.508	12V =	30	%	

NOTE: V = reading in volts

REMARKS: Det. Tracols
N41A: 0.5 mA B: 0.5 mA N42A: 0.5 mA B: 0.5 mA N43A: 0.5 mA B: 0.5 mA N44A: 0.5 mA B: 0.5 mA

DVM # 418927 Calibration due date 1/31/81

Data By Ruah Sammons 10/31/80

Checked By [illegible] 1/1/81

Date 11/29/80 Power 75% Unit 1

Item No.	Parameter	Test Point	Volts at (Time) 1700	Volts at (Time) 1715	Volts at (Time) 1730	Average Volts	Scaling Factor	Value	Units
1	N-35	TP3 to TP4	9.260	9.256	9.259	9.258	$10^{(.8V-11)}$ =	$2.549\ 10^{-4}$	amps
2	N-36	TP3 to TP4	9.281	9.276	9.280	9.279	$10^{(.8V-11)}$ =	$2.65\ 10^{-4}$	amps
3	N-41 Det. A	Back of meter	.271	.270	.270	.270	$10^{-3}V$ =	270	μA
4	N-41 Det. B	Back of meter	.315	.311	.313	.313	$10^{-3}V$ =	313	μA
5	N-41 Power	TP306 to TP305	6.333	6.280	6.311	6.308	12V =	75.7	%
6	N-42 Det. A	Back of meter	.280	.279	.280	.280	$10^{-3}V$ =	280	μA
7	N-42 Det. B	Back of meter	.338	.333	.336	.336	$10^{-3}V$ =	336	μA
8	N-42 Power	TP306 to TP305	6.320	6.284	6.297	6.300	12V =	75.6	%
9	N-43 Det. A	Back of meter	.276	.277	.276	.276	$10^{-3}V$ =	276	A
10	N-43 Det. B	Back of meter	.320	.316	.318	.318	$10^{-3}V$ =	318	A
11	N-43 Power	TP306 to TP305	6.323	6.290	6.302	6.305	12V =	76.7	%
12	N-44 Det. A	Back of meter	.274	.275	.275	.275	$10^{-3}V$ =	275	A
13	N-44 Det. B	Back of meter	.326	.322	.325	.324	$10^{-3}V$ =	324	A
14	N-44 Power	TP306 to TP305	6.325	6.292	6.324	6.314	12V =	75.8	%

NOTE: V = reading in volts

DVM # 486348 Calibration due date 12/27/80

REMARKS:

Data By R.W. Fatenberry 11/29/80

Checked By R.P. Gibbs 1/7/81

CALC # 1,2-XE-92-1 R1

TENNESSEE VALLEY AUTHORITY
SEQUOYAH NUCLEAR PLANT
UNIT NUMBERS 1 AND 2

PRECAUTIONS, LIMITATIONS AND SETPOINTS
FOR
NUCLEAR STEAM SUPPLY SYSTEMS

REVISION 9

MAY, 1981

(plus revised pages 28A.1 - 28.A.4 and 79)

APPROVED WITH CORRECTION
AS NOTED
This approval ... the Contract ... for the ... and dimensions ... noted hereon ...
DATE MAR 30 1984
T.V.A. NUCLEAR ENGR BRANCH
By: CHIEF NUCLEAR ENGINEER
Ans'd by Ltr. # 7601

WESTINGHOUSE ELECTRIC CORPORATION
Nuclear Energy Systems
P. O. Box 355
Pittsburgh, Pennsylvania 15230

~~APPROVED WITH CORRECTION~~
~~AS NOTED~~
~~DATE ... 1982~~
~~T.V.A. NUCLEAR ...~~
~~BY: CHIEF NUCLEAR ...~~
~~Ans'd by Ltr. # 7211~~

PROJECT SQN DATE MAR 30 1984
CONTRACT 8PC00-01984 FILE N2M-2-X
DRAWING NO. PLS
SHEET — REV 9 UNIT 1&2

PAGE 14A THRU 14T

CALC # 1,2-XE-92-1 E

2. mismatch
(FB-510B, FB-520B, FB-530B, FB-540B, FB-511B, FB-521B, FB-531B, FB-541B) — 38% of rated steam flow per steam generator

3. turbine trip
steam generator Hi level signal for feedwater valve closure, turbine trip and feedwater pump trip
(LB-517A, LB-527A, LB-537A, LB-547A, LB-518A, LB-528A, LB-538A, LB-548A, LB-519A, LB-529A, LB-539A, LB-549A) — 75% of level span

II. Permissive and Interlock Circuits

A. P-6 (allows manual block of source range high level reactor trip)
(NC-35D, NC-36D) — 10^{-10} amperes

is equiv. to 1×10^{-5} A

B. P-7 (automatically blocks various "at power" trips at low power)
1. low neutron flux (See P-10)
2. low turbine load (See P-13)

C. P-8 (allows one loop loss of flow below setpoint)
(NC-41N, NC-42N, NC-43N, NC-44N) — 35% of full power (4 loop operation)

D. P-9 (blocks reactor trip on turbine trip below setpoint nuclear power level)
(NC-41S, NC-42S, NC-43S, NC-44S) — 50% of full power

E. P-10 (allows manual block of power range (low setpoint) trip, intermediate range trip, and C-1; blocks source range trip and provides a portion of P-7 signal)
(NC-41M, NC-42M, NC-43M, NC-44M) — 10% of full power

F. P-11 (allows manual block of safety injection actuation on low pressurizer pressure.

(See I.1.A.4 above)

PAGE 14B THRU 14F

Calc 1,2-XE-92-1 R1

3b. Low steam line pressure
(PB-516A, PB-526A, PB-536A, PB-546A) 600 psig
Lead time constant
(PY-516B, PY-526B, PY-536B, PY-546B) 50 seconds
Lag time constant
(PY-516B, PY-526B, PY-536B, PY-546B) 5 seconds

3c. Low-Low T_{avg}
(TB-412D, TB-422D, TB-432D, TB-442D) 540°F

4. Automatic reset of manual block on high pressurizer pressure (P-11)
(PB-455B, PB-456B, PB-457B) 1970 psig

5. Containment high pressure
(PB-934B, PB-935B, PB-936B) 1.54 psig

6. Time delay on SI manual reset 1 minute

B. Steam Line Isolation

1. High steam line flow (See I.1.A.3 above)
2. High-high containment pressure
(PB-934A, PB-935A, PB-936A, PB-937A) 2.81 psig

C. Containment Spray Actuation

1. High-high containment pressure (See I.1.B.2 above)

is equivalent to 5.7×10^{-4} % F

2. Reactor Trips

A. Nuclear Instrumentation

1. Source range high level
(NC-31D, NC-32D) 10^5 counts/second
2. Intermediate range high level
(NC-35F, NC-36F) Current equivalent to 25% of full power
3. Power range, low range, high level
(NC-41P, NC-42P, NC-43P, NC-44P) 25% of full power

Page 14C thru 14E

CALC# 1,2-XE-92-1

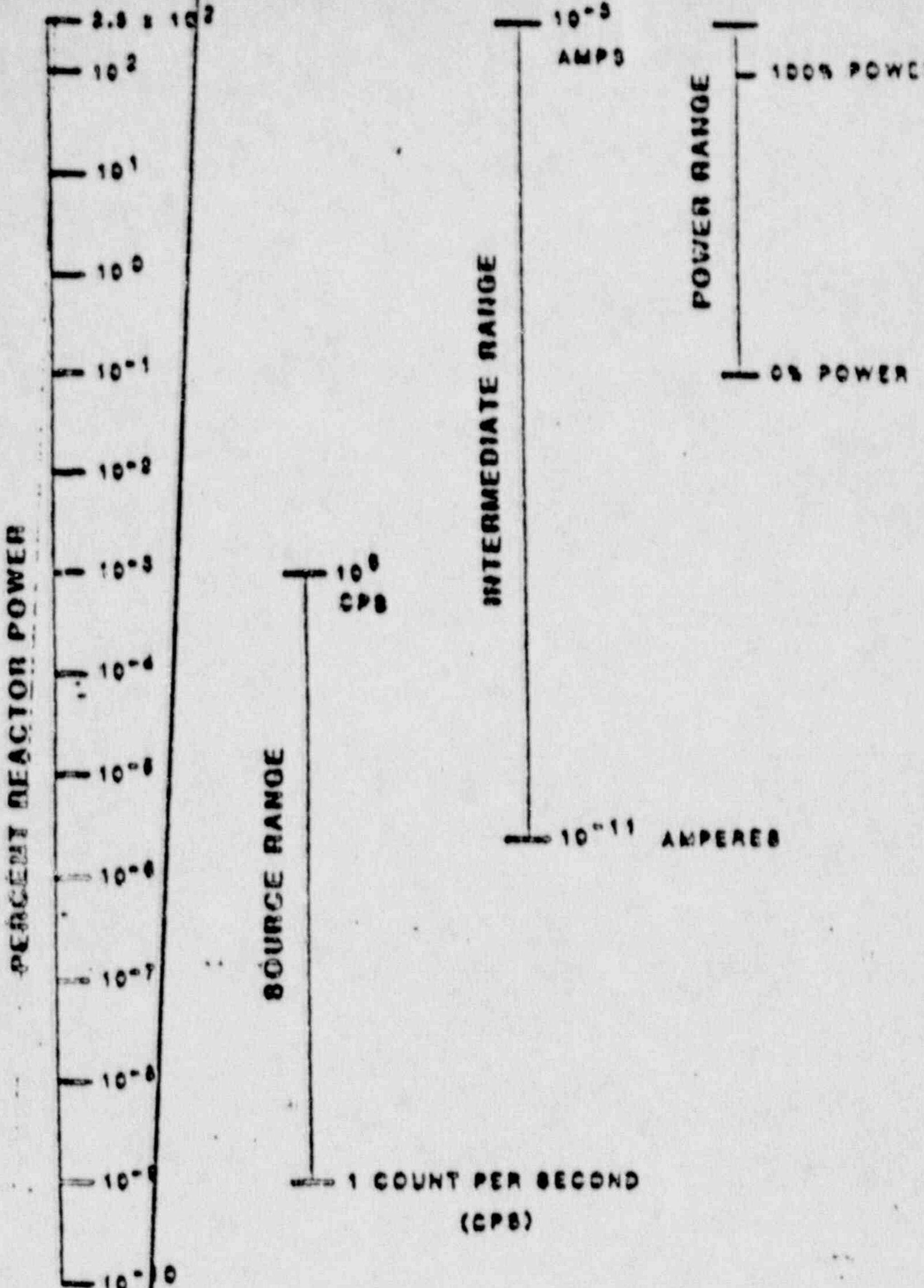


Figure 10.1-1 Neutron Detectors and Ranges of Operation

PAGE 14 B THRU 14 E

generated from detecting the signal onward. Where applicable, this requirement should be met with all lead, lag, and filter time constants set to OFF.

1.14 Controller Transfer Functions

Not Applicable

1.15 Setpoints

Variable	Range of Setting
Intermediate Range High Neutron Flux Reactor Trip	5 to 30% full power
Source Range High Neutron Flux Reactor Trip	$\sim 10^{-5}$ to $\sim 10^{-3}$% of full power
Intermediate Range Rod Withdrawal Stop (C-1)	5 to 25% at full power
P-6	$\sim 10^{-5}$ to $\sim 10^{-3}$% of full power

5.7×10^{-4}% is inside the span

1×10^{-4} is inside this span

All settings with the exception of time constants shall be continuously adjustable within their range and all time constants shall be continuously adjustable or adjustable in increments such that any setpoint can be obtained within ± 10% of the setpoint value.

For the P-10 setpoint see Nuclear Power Range Protection (Document 2).

1.16 Requirements for Test and Calibration

All protection channels should be supplied with sufficient redundancy to provide the capability for channel calibration and test at power. In the case of 1/N logic a bypass must be provided to prevent a reactor trip during test.

ECN NO. L6186 R3
48 OF ___

QA Record

NEP-6.6
Attachment 3
Page 1 of ~~5~~ ~~33~~ ~~47~~ 37 11/12/88 ~~39~~ ~~43~~ ~~45~~ 63

SAFETY EVALUATION FORM

1. To Sequoyah Nuclear Plant, Daisy, TN	3. USQ? ☐ Yes ☑ No
2. From R16 SNP	

Sheet 1

4. Safety Evaluation Number
ECN L6186
9. RIMS Accession Number
R0 SQP 850117 503
R1 B25 881114 532
R2 B25 881117 575
R3 B3789 1127 801
R4
R5

Rev No.	Tot pp	5. Prepared	6. Reviewed	7. Approved	8. Date Appd
0	3	R.A. Edlund	R.R. Campbell	LH Chin/VAB	1/14/85
1	37 ~~33~~	T Fleischer	[illegible]	[illegible]	11-14-88
2	39	T Fleischer	[illegible]	[illegible]	11-17-88
3	63 ~~45~~	Jeff Newton [illegible]	[illegible]	[illegible]	11-22-89
4					
5					

10. Project and Affected Unit(s) Sequoyah Units 1, 2	11. PMP or DCN Number ECN L6186	PMP or DCN Revision
12. FCR, SCR, MCR, DCR, DCN, or CAQR Number DCR 1156	Date of Document(s) 3/11/82	
13. Other Document Identifier None	Date of Document N/A	
14. Special Requirements? ☐ Yes ☑ No See Sheet No. N/A	15. Potential Tech Spec Change ☑ Yes ☐ No	See Sheet No. 26-44 R3

16. References (include system number and name as appropriate) CH1

System 92 - Neutron Monitoring

References:
1. Reg Guide 1.97 R2
2. Design Criteria SQN-DC-V-27.8 "Neutron Monitoring System"

(Continued on Sheet 2)

17. Description of Proposed Activity (Change, Test, or Experiment)

This ECN upgrades to electrical class 1E the source and intermediate range neutron monitors. This modification is being performed to comply with Sequoyah's commitment to provide a means of monitoring plant status and environs after an accident within the guidelines specified in the USNRC's Regulatory Guide 1.97, Revision 2. Reference 24, 25, 28, 29, and 30 R3

(continued on sheet 3)

cc (Attachments):
RIMS, SL 26 C-K

ECN NO. L6186 R3

49 OF ____

Sheet 2
Safety Evaluation
No. ECN L6186
NEP 6.6

ADDITIONAL INFORMATION

16. References (Continued)

3. FSAR Section 7.1.2.1.3
4. FSAR Sections 7.2.1.1.2, 7.2.1.1.3
5. Design Criteria SQN-DC-V-19.0 "Post Accident Monitoring"
6. Design Criteria SQN-DC-V-27.9 "Reactor Protection System"
7. Design Criteria SQN-DC-V-1.0 "General Civil Design Criteria"
8. SQN PAM Appendix F "Design Criteria for Qualification of Seismic Class I and Seismic Class II Mechanical and Electrical Equipment.
9. 47B601-92 Series I-Tabs
10. Design Criteria SQN-DC-V-12.2 "Separation of Electrical Equipment and Wiring"
11. Tech Specs
12. Design Criteria SQN-DC-V-26.2 "Environmental Qualification to 10CFR50.49
13. Calculation 1,2-XE-92-1, Rev. 1 RIMS No. B25 881117 808.
14. Design Criteria SQN-DC-V-2.3 "Containment Vessel"
15. Design Criteria SQN-DC-V-11.3 "Power, Control, and Signal Cable for use in Category I structures
16. Design Criteria SQN-DC-V-13.10 "Seismic Qualification of Conduit"
17. FSAR Tables 7.2.1-1 through 4 and 8.3.1-11, 12, 13, 15, 16
18. FSAR chapter 15.0
19. Plant Procedure PHYSI-13 "Fire"
20. Westinghouse Test Reports, Nuclear Instrumentation System Isolation Amplifier WCAP-7506-L, NEB810126303 and WCAP-7819 NEB8102040314
21. Nuclear Engineering Calculation, "Equipment Required for 10CFR50, Appendix R" SQN-SQ54-127 Rev. 10 (RIMS No. B25880829501)
22. Gamma Metrics Neutron Flux Monitoring Instruction Manual No. 72, Rev. 4, Contract No. 835545
23. Gamma-Metrics Test Report No. 135, Rev. 0, Isolation Testing of Single Channel Isolators, Contract No. 835545
24. Gamma-Metrics Test Report No. 010, Rev. 1, Neutron Flux Monitoring Qualification, Contract No. 835545
25. Gamma-Metrics Test Report No. 096, Rev. 1, Source and Intermediate Range Rack Mount Signal Processors Qualification, Contract No. 835545
26. Demonstrated Accuracy Calculation SQN-EEB-PS-TI28-0001 R3
27. Pipe Rupture Calculation SQN-CEB-SCG-4E00168
28. Test Report No. 12, Rev. 0, Gamma-Metrics RCS series Neutron Flux Monitoring Seismic Qualification Report and MSLB/LOCA Test Report, Contract No. 835545
29. Gamma-Metrics Test Report No. 31, Test Plan for Qualification of Gamma-Metrics RCS series of Neutron Flux Monitoring Systems per IEEE STD 323.1974, Contract No. 835545
30. Gamma-Metrics Test Report No. 40, Rev. 3, Test Report for Class 1E Qualification of Mineral Insulated Cable in the Detector Cable Assembly, Contract No. 835545

ADDITIONAL INFORMATION

16. References (Continued)

31. Gamma-Metrics Document No. 133, Rev. 1 Shutdown Monitor, Contract No. 835545
32. TI-81 NIS calibration for restart following core load, Rev. 4
33. Design Criteria SQN-DC-V-12.1 "Flood Protection Provisions"
34. Design Criteria SQN-DC-V-11.6 "120V A-C Vital Instrument Power System"
35. Gamma-Metrics Test Report No. 26, Rev. 3, "Optical Isolator, Fiber Optic Transmission System and RCS-211 Amplifier Qualification.
36. Gamma-Metrics Test Report No. 27, Rev. 0, "Seismic Test for Optical Isolator, Fiber Optic Transmission System and RCS-211 Amplifier".
37. 120 VAC Vital Inverter Loading Calculation SQN-CPS-021, RIMS No. B87 891011 006.
38. AI-17 Drilling, Cutting, Chipping and Excavating, Revision 13.
39. DIM-SQN-DC-V-27.8-3, RIMS No. B37 891101 802. R3
40. DIM-SQN-DC-V-27.9-7, RIMS No. B37 891101 801.
41. Westinghouse Letter - TVA-89-963, RIMS No. B25 891030 010.
42. HVAC Cooling Load Calculation, Auxiliary Bldg., EGTS Room Elev. 734 ft., RIMS No. B25 891102 504.
43. HVAC Cooling Load Calculation, Auxiliary Bldg., Elev. 714 ft., RIMS No. B25 891102 505.
44. HVAC Cooling Load Calculation, Auxiliary Bldg., General, RIMS No. B25 891102 503.
45. HVAC Cooling Load Calculation, Reactor Bldg., Lower Containment, RIMS No. B25 891102 506
46. VCPS Loading Evaluation, SQN-CPS-022 Rev. 0, RIMS No. B87 891120 002

17. Description of Proposed Activity (Continued)

None of the existing outputs, functions, and interfaces to other systems, such as the Reactor Protection System, will be functionally changed by this upgrade. This ECN will resolve all outstanding Appendix R commitments for the source range channels.

The specific modifications performed by this ECN are described below and will be performed on both Sequoyah Units 1 and 2.

1. Replace the non-1E Westinghouse source and intermediate range neutron detectors with Class 1E Gamma-Metrics source and intermediate range detectors.

ADDITIONAL INFORMATION

17. Description of Proposed Activity (Continued)

2. Reroute cable and conduit, and locate junction boxes and other hardware above containment building flood level from the detectors to containment penetrations 31 (Unit 2 Channel I), 23 (Unit 2 Channel II), 43 (Unit 1 Channel I), and 48 (Unit 1 Channel II). Special pressure tight precut cable from the detector through the containment penetration to the amplifier shall be installed in accordance with Gamma-Metrics Instruction Manual, Reference No. 22.

3. Replace the non-1E Westinghouse pre-amplifier with a 1E Gamma-Metrics amplifier assembly.

4. Replace the cable from Unit 1 penetrations 43 and 48 and Unit 2 penetrations 23 and 31 to the new amplifier assemblies described in number 3 above.

5. Replace the cables and conduit routed from the amplifiers to the NIS racks located in the control building. One channel will be routed on elevation 734 above the design basis flood level.

6. Replace the non-1E Westinghouse intermediate and source range signal processing drawers with Gamma-Metrics intermediate and source range drawers. The new Gamma-Metrics drawers will provide the required electronics to provide qualified signals for compliance with Reference 1.

7. Install a shutdown monitor on each source range neutron monitoring channel to automatically adjust the high flux at shutdown alarm setpoint downward for flux decay during shutdown. It will identify and report flux increases that indicate a loss of reactor shutdown margin. This will eliminate manual adjustment of this high flux at shutdown setpoint (Reference 22 and 31). |R3

ADDITIONAL INFORMATION

17. Description of Proposed Activity (Continued)

8. Route an isolated temporary cable from the Auxiliary Control Room L-10 to Main Control Room panel M-19 to provide redundant neutron flux information to the main control room from the Appendix R backup source range neutron monitor during implementation of items 1-6 above. Only one channel of the source and intermediate range flux monitors will be worked at one time. The first channel to be replaced, post-mod tested, and documented, shall be declared operable prior to removing the second channel from service. The backup and one operating source range channels will ensure the operator has sufficient neutron flux information during implementation of this ECN. The temporary cable shall be removed after both new channels are declared operable and prior to Mode 3. R3

9. Replace the Appendix R Westinghouse backup source range detector, cabling, and electronics with an optically isolated signal from the new Gamma-Metrics Amplifier to a source/power range processor and display. This will reduce maintenance, spare parts storage, and enhance the back-up Control Room neutron flux readout. This will require a revision to Reference No. 21 by minicalculations to the U1C4 and U2C4 Appendix R Calculations SQN-SQS2-0094 and -0100, respectively. R3

10. Replace containment penetrations 31 (Unit 2) and 43 (Unit 1) with 1E qualified penetrations. Penetrations 23 (Unit 2) and 48 (Unit 1) have already been replaced under ECN L6490.

11. Westinghouse drawings 5655D26 sheets 3 and 4 and 108D438 sheets 2, 5 and 43 that are affected by this ECN will be changed under DCR-3094 for Units 1 and 2.

12. The new control room indicator and recorder scales required for this ECN are being purchased and their installation coordinated with DCN M01496, Unit 1, and M01497, Unit 2. R3

13. Westinghouse PLS and setpoint methodology documents that are affected by this ECN will be changed under DCR-3094 for Units 1 and 2.

14. The Technical Support Center Data System (TSCDS) computer software will be modified to reflect the new range and units (10^{-8} to 200% RTP) for the intermediate range instruments. The calculation of the intermediate and source range start up rates (SUR) will be modified to use time tagged levels rather than assume the data is supplied from the input multiplexers every two seconds.

ADDITIONAL INFORMATION

17. Description of Proposed Activity (Continued)

15. All drilling, cutting and chipping will be performed in accordance with AI-17. AI-17 states that the description of the work shall include all information required to locate the exact spot for drilling, chipping or cutting by relating to elevations, distance from column lines and other location references used on plant drawings. Additionally, a review of all effected drawings will be performed to verify embedded piping, electrical conduit, cable troughs, duct work, tunnels or other plant equipment will not be jeopardized and shall be listed under the drawings reviewed section of the permit. Submittal of an AI-17 permit is to serve a verification that the location of all known embedded piping, electrical conduit, cable troughs, reinforcing steel, duct work, tunnels or other plant equipment will be identified before the drilling, cutting, or shipping is started. Reference No. 38.

16. Implementing the proposed activity will require revisions to:

FSAR Sections:	4.4.5.3	Tables:	8.3.1-11	and Figure:	7.2.1-1
15.2.4.2	7.2.1.1.2.c		8.3.1-12		sheets 3 and 4
7.5 (in to-to)	7.2.1.1.3		8.3.1-13		
8.3.1.2.3	7.2.1.1.8		8.3.1-15		
	7.2.2.2.3		8.3.1-16		
	7.2.4		15.2.4-1		
	15.2.4.3				

R3

Tech Specs (See Question 27)

Table 3.3-1 Notes	Table 4.3-7	Bases, Limiting Safety System
Table 3.3-10	Table 2.2-1	Settings Section 2.2.1

This activity will be performed in the following stages:

Stage 1: Will consist of completing items 1 through 8 and 10 by the end of the Unit 1 Cycle 4 outage for Unit 1.

Stage 2: Will complete items 1 through 8 and 10 by the end of the Unit 2 Cycle 4 outage.

Stage 3: Will implement item 9 for both units by the end of the Unit 2 Cycle 5 outage. This portion of the modification is not part of the NRC commitments for Cycle 4.

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Sheet 7
Safety Evaluation
No. ECN L6186
NEP 6.6

ADDITIONAL INFORMATION

18. Systems, Structures, Components Affected

NM - Neutron Monitoring
PAM - Postaccident Monitoring
CV - Containment Vessel

RPS - Reactor Protection System
SR - Source Range
IR - Intermediate Range
RTP - Rated Thermal Power

ITEM	COMPONENT	SYSTEM AFFECTED	DESCRIPTION OF CHANGE
1	Source/Intermediate Neutron Detector and Cabling Channel I, XE-92-5001	NM, PAM, RPS, Appendix R SR Unit 1	Replace non-1E detector with 1E qualified detector which contains two identical, redundant fission chambers and has a pulse signal output proportional to reactor power
2	Source/Intermediate Neutron Detector and Cabling Channel II, XE-92-5002	NM, PAM, RPS, Appendix R SR Unit 2	Replace non-1E detector with 1E qualified detector which contains two identical, redundant fission chambers and has a pulse signal output proportional to reactor power
3	Unit 2 Primary Containment Penetration No. 23 Channel II	Penetration Feedthroughs	Replace 75 ohms triax with 50 ohms triax
4	Unit 2 Primary Containment Penetration No. 31 Channel I	CV and Penetration	Replace complete non-qualified with 1E qualified penetration
5	Unit 1 Primary Containment Penetration No. 48 Channel II	Penetration Feedthroughs	Replace 75 ohm triax with 50 ohm triax
6	Unit 1 Primary Containment Penetration No. 43 Channel I	CV and Penetration	Replace complete non-qualified with 1E qualified penetration
7	Channel I Signal Amplifier XM-92-5001A - U1 XM-92-5001 - U2	NM, PAM, RPS Appendix R, SR Unit 1	Replace non-qualified with 1E qualified amplifier to mate with new detector
8	Channel II Signal Amplifier XM-92-5002 - U1 XM-92-5002A - U2	NM, PAM, RPS Appendix R, SR Unit 2	Replace non-qualified with 1E qualified amplifier to mate with new detector
9	Shutdown Monitor Channel I, XIS-92-5001	NM, PAM	Add new device to automatically adjust high flux at shutdown alarm down
10	Shutdown Monitor Channel II, XIS-92-5002	NM, PAM	Add new device to automatically adjust high flux at shutdown alarm down

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Sheet 8
Safety Evaluation
No. ECN L6186
NEP 6.6

ADDITIONAL INFORMATION

18. Systems, Structures, Components Affected (Continued)

NM - Neutron Monitoring
PAM - Postaccident Monitoring
CV - Containment Vessel

RPS - Reactor Protection System
SR - Source Range
IR - Intermediate Range
RTP - Rated Thermal Power

ITEM	COMPONENT	SYSTEM AFFECTED	DESCRIPTION OF CHANGE
11	Source Range Drawer Channel I, XX-92-5001	NM, PAM, RPS	Replace Westinghouse with Gamma-Metrics
12	Source Range Drawer Channel II, XX-92-5002	NM, PAM, RPS	Replace Westinghouse with Gamma-Metrics
13	Intermediate Range Drawer Channel I, XX-92-5003	NM, PAM, RPS	Replace Westinghouse with Gamma-Metrics; range and readout change from $10^{-11}/10^{-3}$ amps to 10^{-8} /2 x 10^{2} % RTP
14	Intermediate Range Drawer Channel II, XX-92-5004	NM, PAM, RPS	Replace Westinghouse with Gamma-Metrics; range and readout change from $10^{-11}/10^{-3}$ amps to 10^{-8} /2 x 10^{2} % RTP
15	1E Optical Isolator, Unit 1 Channel I, XM-92-5001B	NM, PAM, Appendix R SR Unit 1	New device to isolate Main Control Room from remote shutdown
16	1E Optical Isolator, Unit 2 Channel II, XM-92-5002B	NM, PAM, Appendix R SR Unit 2	New device to isolate Main Control Room from remote shutdown
17	Appendix R Source Range Channel (detector, pre-amp and drawer) XI-92-5	NM, Remote Shutdown 10CFR50 App. R	Replace Westinghouse source range channel with optically isolated output from 1E-qualified Gamma-Metrics amplifier and new source/power drawer
18	Components 1 through 17	120 VAC Vital Instrument Power Board	Increased Load
19	Components 1 through 17	125 VDC Vital Batteries	Increased Load

ADDITIONAL INFORMATION

19. Safety Function(s) of System(s) Affected

The safety functions of the systems affected by this ECN are described below:

1. POSTACCIDENT MONITORING (PAM)

The safety function of the postaccident monitoring system is to provide information on plant variables required by control room operating personnel during accident situations to:

1. permit the operator to take preplanned manual actions to accomplish safe plant shutdown.
2. determine whether safety systems or systems important to safety are performing their intended functions.
3. determine the potential for causing gross breach of the barriers to radioactivity release and to determine if a gross breach of a barrier has occurred.
4. assess the operation of plant systems to make appropriate decisions as to their use.
5. allow for early indication of release of radioactive materials in order to initiate action necessary to protect the public and estimate the magnitude of any impending threat.

The PAM variable affected by this ECN is neutron monitoring. Neutron monitoring provides information for purposes 1 and 2 above. Additional safety function information is available in References 1 and 5.

2. NEUTRON MONITORING SYSTEM - SOURCE AND INTERMEDIATE RANGE

The source and intermediate range neutron monitoring safety functions are described below:

Intermediate range high neutron flux trip

The intermediate range high neutron flux trip circuit shall trip the reactor when one out of the two intermediate range channels exceed the trip setpoint (25% RTP), Tech Specs Table 2.2-1. This trip, which |R3. provides protection during reactor startup, can be manually blocked if two out of four power range channels are above approximately 10 percent power (P-10). Three out of the four power range channels below this value automatically reinstates the intermediate range high neutron flux trip. The intermediate range channels (including detectors) shall be separate from the power range channels. The intermediate range channels can be individually bypassed at the nuclear instrumentation racks to permit channel testing at any time under prescribed administrative procedures and only under the direction of authorized supervision. This bypass action shall be annunciated on the control board.

ADDITIONAL INFORMATION

19. Safety Function(s) of System(s) Affected (Continued)

Source range high neutron flux trip

The source range high neutron flux trip circuit shall trip the reactor when one of the two source range channels exceeds the trip setpoint (10^5 CPS), Tech Spec Table 2.2-1. This trip, which provides protection during reactor startup and plant shutdown, can be manually bypassed when one of the two intermediate range channels reads above the P-6 setpoint value (source range outputs disabled and intermediate range on scale power level) and shall be automatically reinstated when both intermediate range channels decrease below the P-6 value. This trip shall be automatically bypassed by two out of four logic from the power range permissive (P-10). R3

This trip function shall also be reinstated below P-10 by an administrative action requiring manual actuation of two control board mounted switches. Each switch will reinstate the trip function in one of the two protection logic trains. The source range trip shall be set between the P-6 setpoint and the maximum source range level. The channels can be individually blocked at the nuclear instrumentation racks to permit channel testing at any time under prescribed administrative procedures and only under the direction of authorized supervision. This blocking action shall be annunciated on the control board.

The source and intermediate range neutron monitoring system also comprises a portion of PAM for purposes described above in "Postaccident Monitoring". Additional detailed safety function information is available in References 2, 39, and 40.

Another safety function of the source range neutron flux monitor is to provide an increasing count rate when RCS boron concentration decreases during shutdown, which is a condition II fault described in Chapter 15 of the FSAR. R3

The intermediate range provides a signal to block rod withdrawal (C-1) in the event of high neutron flux, Reference 2. This is a control function only.

The 120 VAC Vital Instrument Power Boards provides an extremely reliable source of instrument and control power for reactor protection circuits and other critical instruments. It is designed with sufficient independence, redundancy, and testability to perform its safety function assuming a single failure. Furthermore, none of the following design basis events shall prevent the vital instrument power system from performing its function: any single equipment or component failure; any single act, event, component failure, or circuit fault that could cause multiple equipment malfunctions; the safe shutdown earthquake; the postulated accident environments; accident generated missiles; accident generated flooding, sprays or jets; fire; fire protection system operation; loss of off-site power, Reference 34. R3

ADDITIONAL INFORMATION

19. Safety Function(s) of System(s) Affected (Continued)

3. PRIMARY CONTAINMENT SYSTEM

The Primary Containment System will be breached during the replacement of electrical containment penetrations 43 (U1) and 31 (U2) and during the replacement of the feedthroughs on penetrations 23 (U2) and 48 (U1). The safety function of the primary containment system is to limit leakage of radioactive material from the containment building under design basis accident conditions. Additional safety function information available in Reference 14. R3

4. REACTOR PROTECTION SYSTEM

The source and intermediate range neutron monitor input to the reactor trip system high neutron flux trip circuits which are described in #2 above, and will trip the reactor at high source or intermediate flux levels, respectively, during reactor start-up. The reactor trip system comprises the Reactor Protection System.

The Reactor Protection System is by definition a primary safety system, due to its requirement to shut down the reactor and maintain it in a safe condition whenever a possible dangerous situation exists.

The functional performance requirements of the Reactor Trip System shall include provisions for automatically initiating a reactor trip:

a. Whenever necessary to prevent fuel damage for an anticipated transient (Condition II).

b. To limit core damage for infrequent faults (Condition III).

c. To keep the energy generated in the core under control to limit fuel damage such that 10CFR 100 dose limits are met and peak clad temperatures are less than 2200°F.

The Reactor Trip System initiates a turbine trip signal whenever reactor trip is initiated to prevent the reactivity insertion that would otherwise result from excessive reactor system cooldown and to avoid unnecessary actuation of the Engineered Safety Features Actuation System. Additional safety function information is available in Reference 6.

5. REMOTE SHUTDOWN INSTRUMENTATION

Source range neutron flux is an instrumentation channel required in the event of a main control room evacuation for safe shutdown. Reference 21. R3

SAFETY EVALUATION

20. Effects on Safety

This modification and the Tech Spec change will not affect the safety functions of the systems listed in number 19 above or any other systems important to the safety for the following reasons:

1. This ECN upgrades the source and intermediate range neutron monitoring system in order to meet the qualification guidelines of Reg. Guide 1.97 R2. The source and intermediate neutron monitors will provide a primary safety function by providing the control room operator information to take preplanned manual actions to accomplish safety shutdown of the plant during accident conditions. As discussed in Reference 6, Table 3.1.2-1, credit is not taken for source and intermediate range high flux reactor trips in the FSAR Chapter 15 safety analysis since they are in addition to power range trips. Also, the interlocks described in number 19, item 2 are not changed or altered, hence the function of the Reactor Protection System is not changed.

2. During implementation of this modification, the backup source range neutron monitor which presently outputs to the auxiliary control room will also temporarily output to the main control room. Tech Spec 3.3.3.5 only requires that the backup source range neutron monitor be operable only in modes 1, 2, or 3. Since this modification will be performed in modes 5 and 6, L.C.O. 3.3.3.5 will not be entered.

   The existing Westinghouse backup source range electronics has a built-in isolated output which will be used for the main control room readout. Westinghouse testing shows that no credible fault could damage the backup source range electronics with this temporary cable installed. References 20 and 41. This will allow the modification to proceed during Mode 6 with one permanent source range neutron flux channel inoperable without invoking a limiting condition for operation as described in Tech Spec 3/4.9.2 and maintain Appendix R compliance. R3

   The temporary cable from the auxiliary control room to the main control room to connect the backup source range neutron monitor to the main control room (as described in item 2 above) shall be routed such that the qualification values for the isolation amplifier will not be exceeded. Temporary breaches of fire barriers will be administratively controlled as required by Reference 19. This ensures that the consequences of a fire are not increased during implementation of this ECN.

SAFETY EVALUATION

20. Effects on Safety (Continued)

3. Electrically, the source and intermediate neutron range monitors affect output only to PAM and RPS as discussed above. No other system receives input from any portion of the source or intermediate neutron monitoring system. Power consumption of the new Gamma-Metrics units is documented in calculation SQN-CPS-021 (Reference 37). The effect of this increased load has been evaluated in calculation SQN-CPS-022 (Reference 46) for both the vital AC and DC power systems. The addition of the 1E optical isolator assembly, References 35 and 36, will allow the use of one of the Main Control Room detector signals yet maintain the separation required for Appendix R. ECN L6186 will, therefore, have no effect electrically on any other system important to safety except for the 120 VAC vital power boards and 125V vital DC power systems. R3

4. The upgrade of the source and intermediate range neutron monitoring system to Reg Guide 1.97, Rev. 2, Category I qualifications assures that the system will be able to withstand seismic or environmental stresses and remain functional to provide the primary safety function described above.

5. Each source and intermediate range neutron monitoring channel will be fully redundant and separate from the other in accordance with the requirements of Reference 10. This assures that the primary safety function of the source and intermediate range neutron flux monitors is not compromised by single failure.

6. The upgrade of the source and intermediate range neutron monitoring system to 1E requires the components of the system to be seismically mounted. Therefore, components or equipment of other systems important to safety will not be subjected to seismically induced missile damage. R3

7. This Item moved to Item 2 this question for clarity.

8. The effects of pipe rupture (Reference 27) on the new cable routing, junction boxes, and other hardware have been evaluated in accordance with SQEP-51. The evaluation concluded that the components will not be affected by pipe rupture, assuring the reliability of the primary safety function of the source and intermediate neutron monitors. (See Special Requirements No. 2). R3 R3

9. Containment electrical penetrations 43 and 31 will be seismically and environmentally qualified, and leak tested in accordance with Tech Spec surveillance requirements 4.6.1.2. This ensures that containment leak integrity will be within the margin of safety defined in the bases of Tech Specs 3/4.6.1.1 and 3/4.6.1.2 and that the containment will be capable of providing a radioactivity barrier.

SAFETY EVALUATION

20. Effects on Safety (Continued)

10. All components affected by this ECN located in harsh or essentially mild environments will be qualified for these environments in accordance with the requirements of Reference 12, ensuring reliability of instrumentation during Chapter 15 condition III or IV faults. (See Special Requirements No. 5).

11. When the ECN is completed, one channel of the source and intermediate neutron monitoring system will provide a signal through an optical isolator to the auxiliary control room. The amplifier will receive power directly from an Auxiliary building vital instrument board. This ensures that the effects of an Appendix R fire in the control building or an undesirable habitability condition existing in the main control room will not affect the backup source and intermediate range neutron monitoring system's primary safety function. R3

12. One channel of the source and intermediate range monitors will be routed above Auxiliary Building floor elevation 734 and the other will be routed above floor elevation 714. The Auxiliary Building floor will provide a fire barrier, ensuring that fire in the Auxiliary Building will not affect both neutron monitoring channels and protects one channel from the Design Basis Flood, Reference 33.

13. The new shutdown monitors identified in block 17, Item 7, will automatically adjust the high flux at shutdown alarm setpoint downward during plant shutdown as the count rate decreases. Presently, this function is manually performed and addressed in FSAR Section 15.2.4.2. When the count rate achieves a steady value and then eventually increases, the alarm setpoint remains at its lowest value. An alarm will occur when the count rate reaches a value equal to the alarm setpoint which is set at 3 times the average count rate. The alarm setpoint can be increased only by depressing the alarm setpoint reset at which time a new alarm setpoint will be computed from the current count rate value (Reference 22). There will be one shutdown monitor for each neutron monitoring channel. Each will be electrically separate and fully redundant in accordance with the requirements of Reference 10. Also, visual verification of the setpoint and count rate can be performed by the operator any time below 10^4 counts per second, (Reference 22), when the shutdown monitor is in service to ensure the monitors are performing their function. The high flux at shutdown alarm will continue to perform its function as before, with the alarm setpoint being adjusted automatically by the shutdown monitors. The reliability, redundancy, and shutdown monitor tracking verification feature ensures that this high flux at shutdown alarm function will not be affected, hence the consequences of boron dilution, a condition II fault, will not be increased. R3

SAFETY EVALUATION

20. Effects on Safety (Continued)

14. The new Gamma Metric source range neutron monitors are equivalent to the Westinghouse BF-3 detectors with regard to instrument sensitivity. Indicated response to neutron flux is not changed significantly by this modification and will not affect reactor trip interlock setpoints or alarm setpoints with regard to the power at which they occur.

Technical Instruction (TI-) 81, NIS calibration for restart following core loads, Reference 32, is normally used to provide recalibration information to instrumentation for power and intermediate range detectors prior to restart following refueling. TI-81 ratios the new core design to the last core design multiplied by the detector process output measured during the last cycle at full power to obtain the expected process output at full power for the new cycle.

In the past, the detector process output was in amps which equated to rated thermal power. The new Gamma-Metrics detector process output will be in rated thermal power. TI-81 will have to be revised to use these revised engineering units. For the first startup after the installation of the new Gamma-Metrics detectors, an initial expected full power calibration factor will be supplied by the vendor.

A NIS calibration procedure will be prepared which will be a part of PMT-62, and include instructions for the initial startup after the installation of the new detectors. In order to help assure that the new source/intermediate range detectors do not contribute to any overpower condition or rate of change, the 25 percent intermediate range reactor trip setpoint will be lowered to 12 percent and the 20 percent rod stop will be lowered to 9 percent for this initial startup only. Before power is increased above 5 percent, an evaluation of the intermediate range detector response will be made and the detector electronics recalibrated if necessary. Once an acceptable calibration has been verified, the trip and rod stop setpoints may be reset to the 25 and 20 percent values and power increase continued. R3

Precise measurements of reactor power at several plateaus during the first startup after refueling are standard practice. If necessary, NIS may be recalibrated as a result of any of these measurements. In addition, TI-81 may be used in the 5 to 25 percent power range to obtain recalibration factors when low leakage loading patterns result in erroneous detector responses.

SAFETY EVALUATION

20. Effects on Safety (Continued)

15. A fire hazard analysis evaluation , SQN-26-D053-EPM-MHS-022289 has been performed for Unit 2 to ensure that a fire in one location will not affect both instrument channels. See Special Requirement No. 1 for the Unit 1 limitations until this analysis has been performed.

16. There will be minor additional heat loads added to the Reactor, Auxiliary, and Control Buildings, which is documented in References 42, 43, 44, and 45. This additional heat load is small enough that it will not adversely affect these areas.

17. FSAR Section 7.5 in to-to will be revised by one 10 CFR 50.59 evaluation to address the new plant configuration following the modifications necessary to satisfy NRC commitments in the PAM licensing submittal (RIMS L44 881228 808). Therefore, this safety evaluation will only refer to the CRFSAR for PAM implementation. See Special Requirement No. 6. R3

21. Would the proposed activity increase the probability of an accident previously evaluated in the SAR?

/_/ Yes /X/ No

Justification:

The source and intermediate range neutron monitoring system does not provide a function to reduce the probability of Condition III or IV faults. However, the source range monitor provides an alarm for RCS boron dilution during shutdown (a Condition II fault). The new shutdown monitor described in Block 17 (Item 7) will automatically adjust the neutron flux alarm setpoint down during shutdown. This feature will be verified operable by surveillance testing and eliminate the need for an adjustment to the setpoint during shutdown as described in FSAR Section 15.2.4.2. This will reduce the human involvement in this setpoint, ensure that the setpoint is correct at all times for all background conditions, and subsequently reduce the probability of boron dilution going unnoticed during shutdown.

As discussed in Number 20 "Effects on Safety", the lowered intermediate range reactor trip and rod stop setpoint during initial startup after core reload will ensure conservatism in these setpoints. R3

Also, the modification is designed so that it will not indirectly affect (by seismically induced missiles, etc.) any other component, equipment, or system necessary for reducing the probability of an accident.

SAFETY EVALUATION

22. Would the proposed activity increase the consequences of an accident previously evaluated in the SAR?

/_/ Yes /X/ No

Justification:

The source and intermediate range neutron monitor's function in the Reactor Protection System is not credited for mitigating the consequences of a Chapter 15 accident, according to Reference 6. However, the source and intermediate range neutron monitors do provide the operators reactor power level information after a Condition II, III, Or IV fault in order to take preplanned manual actions to accomplish safe shutdown. As discussed in #20 "Effects on Safety", the new source and intermediate range neutron monitor components and the new shutdown monitors are procured, designed, and will be installed to ensure reliability after being subjected to seismic and environmental stresses. Also, the system is designed so that the neutron monitors will not be rendered inoperable by single failure. R3
Based on the discussion above, the source and intermediate neutron PAM parameter will be available to the operator during and after Condition II, III, or IV faults so he may accomplish safe shutdown and mitigate the consequences of such faults.

The temporary breach of containment to replace the penetration can only be performed during Mode 5, cold shutdown, due to Tech Spec 3.6.1.1 and 3.9.4 which require containment integrity during Modes 1, 2, 3, 4, and 6. Having containment breached during Mode 5 has already been analyzed and consequences of an accident cannot be increased. R3

23. Would the proposed activity increase the probability of a malfunction of equipment important to safety previously evaluated in the SAR?

/_/ Yes /X/ No

Justification:

As discussed in Number 20 "Effects on Safety", the source and intermediate range neutron monitors are upgraded to safety Class 1E and designed in accordance with the requirements of References 1, 10, and 12. They will not be susceptible to seismic or environmental stresses, nor will they be susceptible to single failure. The probability of failure of the neutron monitors is not increased.

Also, as discussed in Number 20 "Effects on Safety", the source and intermediate range neutron monitors cannot become seismically induced missiles and contribute to the probability of malfunction of other equipment important to safety. R3

Safety Evaluation

23. (Continued)

As discussed in Number 20 "Effects on Safety", the addition of the redundant shutdown monitors will enhance the capability of the high flux at shutdown alarms since they will be automatically adjusted downward as the background neutron flux level reduces. This will reduce the human element involved. The alarm setpoint is the lowest previous value of the product of the alarm ratio and the average neutron rate of the last 1200 counts, Reference 22. Surveillance testing will ensure that these alarm setpoints are operable. As discussed in Reference 20, the operator does not depend entirely on this alarm setpoint but has audible indication of increasing neutron flux from the audible count rate drawer of the NIS system and visual indication from counts per second meters for each channel on the main control board and source range drawer. R3

24. Would the proposed activity increase the consequences of a malfunction of equipment important to safety previously evaluated in the SAR?

/ / Yes /X/ No

Justification:

As discussed in the Question 23 Justification, the probability of failure of the neutron monitors is not increased. Hence, the consequences of a reactor power excursion at low-power operation will not be increased since the neutron monitors will be available to initiate a reactor trip.

Also, after a Condition II, III, Or IV fault, the operator will be able to rely on the neutron flux PAM parameter to determine whether certain equipment (such as reactor rods or safety injection, etc.) responded to the fault as required. This will allow the operator to take the necessary action to mitigate the consequences if that equipment did not respond as required.

25. Would the proposed activity create a possibility for an accident of a different type than any evaluated previously in the SAR?

/ / Yes /X/ No

Safety Evaluation

Justification:

The impacted instrumentation will perform the identical function following the modification as prior to the modification with higher availability and reliability. The intermediate range high neutron flux trip and the source range high neutron flux trip will continue to function as described in FSAR Sections 7.2.1.1.2b and c, respectively. PAM instrumentation will be installed as described in #20 "Effects on Safety" and will not be susceptible to single failure. As a result, the operator will have access to reactor power level information and will be able to make decisions based on that information to avoid any possibility of a type of accident not previously evaluated in the FSAR.

Also, since the safety functions of other systems and structures are not affected, no new accident can be created by this modification.

In the event of a failure of the shutdown monitor alarm, the audible count rate and visual indication is still available to the operators. The neutron flux signal to the shutdown monitor is through a pulse buffer whose input is optically isolated from its output, Reference 22. R3

26. Would the proposed activity create a possibility for a malfunction of equipment of a different type than any evaluated in the SAR?

/_/ Yes /X/ No

Justification:

Implementing the proposed activity is necessary to comply with the USNRC's Reg Guide 1.97, Rev. 2. Following this modification, the affected instrumentation will perform its safety function and comply with the design requirements as described in Reference 1, 2, 5, 6, 7, 8, 10, 12, 14, 15, and 16*. Therefore, the proposed activity will not create a possibility for a malfunction of equipment of a different type than any evaluated previously in the FSAR. Additional information for each component affected is provided in the following table: R3

R3

*PMT 62 will be successfully completed prior to declaring the new instrumentation operable.

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Safety Evaluation

26. (Continued)

ITEM	COMPONENT	DESCRIPTION OF CHANGE	HAS A NEW MALFUNCTION BEEN CREATED?
1	Source/Intermediate Neutron Detector and Cabling Channel I, XE-92-5001	Replace non-1E qualified with detector which has output proportional to reactor power	No, New component is a sealed, pressure tight, qualified device and contains two identical, redundant fission chambers
2	Source/Intermediate Neutron Detector and Cabling Channel II, XE-92-5002	Replace non-1E qualified with detector which has output proportional to reactor power	No, New component is a sealed, pressure tight, qualified device and contains two identical, redundant fission chambers
3	Unit 2 Primary Containment Penetration No. 23 Channel II	Replace 75 ohms triax with 50 ohms triax to match new detector cable impedance	No, a qualified 50 ohms feedthrough shall replace the existing 75 ohm
4	Unit 2 Primary Containment Penetration No. 31 Channel I	Replace complete non-qualified with 1E qualified penetration	No, an electrically qualified one shall replace the existing non-qualified
5	Unit 1 Primary Containment Penetration No. 48 Channel II	Replace 75 ohm triax with 50 ohms triax to match new detector cable impedance.	No, a qualified 50 ohms feedthrough shall replace the existing 75 ohm
6	Unit 1 Primary Containment Penetration No. 43 Channel I	Replace complete non-qualified with 1E qualified penetration	No, an electrically qualified one shall replace the existing non-qualified
7	Channel I Signal Amplifier XM-92-5001A - U1 XM-92-5001 - U2	Replace non-qualified with 1E qualified amplifier to mate with new detector	No, the function is the same as the existing design and new design is qualified
8	Channel II Signal Amplifier XM-92-5002 - U1 XM-92-5002A - U2	Replace non-qualified with 1E qualified amplifier to mate with new detector	No, the function is the same as the existing design and new design is qualified

ECN NO. L6186 R3

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Safety Evaluation

26. (Continued)

ITEM	COMPONENT	DESCRIPTION OF CHANGE	HAS A NEW MALFUNCTION BEEN CREATED?
9	Shutdown Monitor Channel I XIS-92-5001	Add new device to automatically adjust high flux at shutdown alarm down	No, as discussed in No. 20, even if this new device were to fail totally, the increasing audible count rate would still be available to alert the oper. to the event
10	Shutdown Monitor Channel II XIS-92-5002	Add new device to automatically adjust high flux at shutdown alarm down	No, as discussed in No. 20, even if this new device were to fail totally the increasing audible count rate would still be available to alert the oper. to the event
11	Source Range Drawer Channel I, XX-92-5001	Replace Westinghouse with Gamma-Metrics	No, new qualified drawer has the same output and function, except high voltage does not have to be deenergized
12	Source Range Drawer Channel II, XX-92-5002	Replace Westinghouse with Gamma-Metrics	No, new qualified drawer has the same output and function, except high voltage does not have to be deenergized
13	Intermediate Range Drawer Channel I, XX-92--5003	Replace Westinghouse with Gamma-Metrics	No, new qualified drawer has same output and functions
14	Intermediate Range Drawer Channel II, XX-92-5004	Replace Westinghouse with Gamma-Metrics	No, new qualified drawer has same output and functions
15	1E Optical Isolator, Unit 1 Channel I, XM-92-5001B	New device to isolate Main Control Room remote shutdown	No, any malfunction of this will result in loss in detector signal but there is an associated redundant channel

R

Safety Evaluation

26. (Continued)

ITEM	COMPONENT	DESCRIPTION OF CHANGE	HAS A NEW MALFUNCTION BEEN CREATED?
16	1E Optical Isolator, Unit 2 Channel II, XM-92-5002B	New device to isolate Main Control Room remote shutdown	No, any malfunction of this will result in loss in detector signal but there is an associated redundant channel
17	Appendix R Source Range Channel (detector, pre-amp and drawer), XI-92-5	Replace Westinghouse source range channel with optically isolated output from 1E-qualified Gamma-Metrics amplifier and new source/ intermediate drawer	No, new source/power drawer is a qualified 1E device even though its signal and power cabling is not designed 1E. The power source is the same as previous design vital instr. power. The increased reliability of the new fission chamber detectors and the routing of all cables and elec- tronic mounted above the Design Basis flood level, Ref. 33 will offset the deletion of the backup source range detector and will not increase the susceptibility of Tech. Spec. 3.3.3.5.
18	Components 1 through 17	Increased load to the 120 VAC Vital Instrument Power Board	No, prior to energizing the equip the capacity calc. for the Vital AC shall be performed
19	Components 1 through 17	Increased load to the 125 VDC Vital Batteries	No, prior to energizing the equip the capacity calc. for the Vital AC shall be performed

R

Safety Evaluation

27. Would the proposed activity reduce any margin of safety as defined in the basis for any technical specification?

/_/ Yes /X/ No

Justification:

1. The containment integrity and penetration operability requirements are addressed in Tech Spec Section 3/4.6.1. Implementing the proposed activity (Stages 2 and 3) will require upgrading the two electrical penetrations identified in "Systems, Structures, or Components Affected" (#18). Testable penetrations Surveillance Instruction (SI-157) will be implemented following the modification to assure the above Tech Spec requirements are met.

2. The note in Table 3.3-1 of the Tech Spec stating "High Voltage To Detector May Be Deenergized Above the P-6 (block of source range reactor trip) Setpoint" is no longer required. This was only required when the previous design source range detector was being deenergized. The new detector and amplifier design will be full range and will not be deenergized. However, in order to reduce operator and site procedures impact, all source range outputs can be disabled above the P-6 setpoint. Therefore, Tech Spec table notation for Table 3.3-1 will be revised as shown on Sheets 26 and 27 of this safety evaluation. Sheets 28 and 29 will not be impacted since table notation is still required. Detector replacement will occur in Stages 1 and 2 described in Block 17. R3

3. Tech Spec Tables 3.3-10 and 4.3-7 "Accident Monitoring Instrumentation" will be revised as shown on Sheets 30 through 33 of this safety evaluation to show the new PAM instrumentation identified in "Systems, Structures or Components Affected". The monitoring equipment identified will be installed in Stages 1 and 2. R3

4. The intermediate range neutron flux P-6 permissive engineering units will change from amps to percent power to provide the plant operators more meaningful information in the main control room. This will result in a revision to Tech Spec Table 2.2-1 as shown on Sheets 34 and 35 of the safety evaluation. Reference 13 justifies this scale change. The scale change will be performed during Stages 1 and 2. R3

Safety Evaluation

27. (Continued)

5. A demonstrated loop accuracy analysis (Reference 26) will be performed to prove that the allowable values for these reactor trips specified in Table 2.2-1 are acceptable. The new monitors will be installed during Stages 1 and 2 (see Special Requirements No. 3).

   Technical Specification Section 2.2.1, "Limiting Safety System Settings" (Bases), intermediate and source range nuclear flux states "no credit was taken for operation of the trips associated with either the intermediate or source range channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System." R3

6. The margin of safety as specified in the bases of Tech Spec 3/4.9.2 will not be reduced. As mentioned in Block #20 "Effects on Safety", two neutron monitoring channels will be available at all times during refueling while this modification is implemented, ensuring changes in the reactivity condition of the core that may occur will be detected. This will be ensured during Stages 1 and 2.

7. Contrary to Revision 2 of this safety evaluation, Tech Spec 3/4.3.3.5 Table 3.3-9 and 4.3-6 Sheets 39 through 42 will not be revised. Source range instrumentation is still available for remote shutdown. Intermediate range and decades per minute indication will be available, however, is not required for remote shutdown. Therefore, a Tech Spec revision is not warranted. R3

8. Tech Spec Section 2.2.1 "Limiting Safety System Settings, Bases", the reference to intermediate range current level, shall be revised as shown on pages 43 and 44, since the intermediate range will read in percent power.

   Based on the discussion above, implementing ECN L6186 will not challenge or degrade the margin of safety as defined by any of the Tech Spec bases.

28a. Special Requirements

1. The Fire Hazard evaluation for the Unit 1 conduit installation for this ECN has to be completed prior to installation beginning in the outstanding locations. The outstanding locations are inside the Unit 1 annulus and containment vessel. All elevations in the Auxiliary Building have been evaluated. R3

Safety Evaluation

28a. (Continued)

2. The unverified assumption in pipe rupture calculation associated with each individual channel, SQN CEB-SCG-4E00168, will have to be resolved before that channel can be declared operable.

3. Prior to the Unit 1 and 2 Gamma-Metrics equipment being installed, the Instrument Accuracy calculation (SQN-EEB-PS-TI-28-0001) shall be completed.

4. Equipment/channel cannot be considered operable until voltage drop calculation SQN-VD-VAC-016 can be verified with as-constructed cable lengths for that channel. R3

5. Installation of detectors, detector cabling, and electrical penetrations cannot begin until EQ Binders SQN EQ INM001 and SQN EQ PENE005 are issued for this ECN.

6. Prior to declaring any of the system operable, this safety evaluation will be revised to reference the CRFSAR for the PAM implementation (FSAR Section 7.5 in to-to).

28b and 29

Based on the safety evaluation provided above, it is concluded that no unreviewed safety question exists as a result of implementing ECN L6186.

NEP 6.6 Sheet 26 | R3
Safety Evaluation
No. ECN L6186

TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the reactor trip system breakers in the closed position and the control rod drive system capable of rod withdrawal, and fuel in the reactor vessel.

**The channel(s) associated with the protective functions derived from the out of service Reactor Coolant Loop shall be placed in the tripped condition.

#The provisions of Specification 3.0.4 are not applicable.

##Source Range Outputs ~~High voltage to detector~~ may be disabled ~~de-energized~~ above the P-6 (Block of Source Range Reactor Trip) setpoint. R3

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours and/or open the reactor trip breakers.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours. R51

b. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1.1. R51

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL and the Power Range, Neutron Flux high trip reduced to less than or equal to 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours.

d. The QUADRANT POWER TILT RATIO, as indicated by the remaining three detectors is verified consistent with the normalized symmetric power distribution obtained by using the movable incore detectors in the four pairs of symmetric thimble locations at least once per 12 hours when THERMAL POWER is greater than 75% of RATED THERMAL POWER.

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NEP 6.6 sheet 27 | R3
Safety Evaluation
No. ECN L6186

TABLE 3.3-1 (Continued)

TABLE NOTATION

*With the reactor trip system breakers in the closed position, the control rod drive system capable of rod withdrawal, and fuel in the reactor vessel.

**The channel(s) associated with the protective functions derived from the out of service Reactor Coolant Loop shall be placed in the tripped condition.

#The provisions of Specification 3.0.4 are not applicable.

##Source Range outputs ~~High voltage to detector~~ may be disabled ~~de-energized~~ above the P-6 (Block of Source Range Reactor Trip) setpoint. | R

ACTION STATEMENTS

ACTION 1 - With the number of OPERABLE channels one less than required by the Minimum Channels OPERABLE requirement, restore the inoperable channel to OPERABLE status within 48 hours or be in HOT STANDBY within the next 6 hours and/or open the reactor trip breakers.

ACTION 2 - With the number of OPERABLE channels one less than the Total Number of Channels, STARTUP and/or POWER OPERATION may proceed provided the following conditions are satisfied:

a. The inoperable channel is placed in the tripped condition within 6 hours. | R39

b. The Minimum Channels OPERABLE requirement is met; however, one additional channel may be bypassed for up to 4 hours for surveillance testing per Specification 4.3.1.1.1. | R39

c. Either, THERMAL POWER is restricted to less than or equal to 75% of RATED THERMAL POWER and the Power Range, Neutron Flux trip setpoint is reduced to less than or equal to 85% of RATED THERMAL POWER within 4 hours; or, the QUADRANT POWER TILT RATIO is monitored at least once per 12 hours.

d. The QUADRANT POWER TILT RATIO, as indicated by the remaining three detectors, is verified consistent with the normalized symmetric power distribution obtained by using the movable incore detectors in the four pairs of symmetric thimble locations at least once per 12 hours when THERMAL POWER is greater than 75% of RATED THERMAL POWER.

This change is No longer required per discussion in safety evaluation Section 27.2 | R3

NEP 6.6 Sheet 28 R3
Safety Evaluation
No. ECN L6186

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2, and *	1
2. Power Range, Neutron Flux	4	2	3	1, 2	2#
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2#
4. Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2	2#
5. Intermediate Range, Neutron Flux	2	1	2	1, 2, and *	3
6. Source Range, Neutron Flux					
A. Startup	2	1	2	2##, and * DELETE	4
B. Shutdown	2	0	1	3, 4 and 5	5
7. Overtemperature Delta T Four Loop Operation	4	2	3	1, 2	6#
8. Overpower Delta T Four Loop Operation	4	2	3	1, 2	6# R45
9. Pressurizer Pressure-Low	4	2	3	1, 2	6# R45
10. Pressurizer Pressure--High	4	2	3	1, 2	6#
11. Pressurizer Water Level--High	3	2	2	1, 2	7#

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This change is NO longer required per disicussion in safety evaluation 27.2 | R3

NEP 6.6 Sheet 29 | R3
Safety Evaluation No. ECN L6186

TABLE 3.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NO. OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1, 2, and *	1
2. Power Range, Neutron Flux	4	2	3	1, 2	2#
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1, 2	2#
4. Power Range, Neutron Flux, High Negative Rate	4	2	3	1, 2	2#
5. Intermediate Range, Neutron Flux	2	1	2	1, 2, and *	3
6. Source Range, Neutron Flux					
A. Startup	2	1	2	2##, and * ← DELETE	4
B. Shutdown	2	0	1	3, 4 and 5	5
7. Overtemperature ΔT Four Loop Operation	4	2	3	1, 2	6#
8. Overpower ΔT Four Loop Operation	4	2	3	1, 2	6#
9. Pressurizer Pressure-Low	4	2	3	1, 2	6#
10. Pressurizer Pressure--High	4	2	3	1, 2	6#
11. Pressurizer Water Level--High	3	2	2	1, 2	7#

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R33

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NEP 6.6 Sheet 30 R3
Safety Evaluation
No. ECN L6186

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Reactor Coolant T_{Hot} (Wide Range)	2	1
2. Reactor Coolant T_{Cold} (Wide Range)	2	1
3. Containment Pressure (Wide Range)	2	1
4. Refueling Water Storage Tank Level	2	1
5. Reactor Coolant Pressure (Wide Range)	2	1
6. Pressurizer Level (Wide Range)	2	1
7. Steam Line Pressure	2	1
8. Steam Generator Level - (Wide Range)	2/steam line	1/steam line
9. Steam Generator Level - (Narrow Range)	1/steam generator	1/steam generator
10. Auxiliary Feedwater Flow Rate	1/steam generator	1/steam generator
11. Reactor Coolant System Subcooling Margin Monitor	1/pump	1/pump
12. Pressurizer PORV Position Indicator*	1	0
13. Pressurizer PORV Block Valve Position Indicator**	2/valve#	1/valve
14. Safety Valve Position Indicator	2/valve	1/valve
15. Containment Water Level (Wide Range)	2/valve#	1/valve
16. In Core Thermocouples	2	1
17. Reactor Vessel Level Instrumentation System***	4/core quadrant	2/core quadrant
18. Source : Intermediate Range Nuclear Instrumentation	2	1
	2	1 ADD

R5
R5
R50

*Not applicable if the associated block valve is in the closed position.
**Not applicable if the block valve is verified in the closed position with power to the valve operator removed.
***This Technical Specification and surveillance requirement will not be implemented until Sequoyah Specific Instructions are developed for the use of this system as committed to in the TVA response to Supplement 1 of NUREG-0737.
#At least one channel shall be the acoustic monitors.

R50

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NEP 6.6 Sheet 31
Safety Evaluation No.
ECN L6186

TABLE 3.3-10

ACCIDENT MONITORING INSTRUMENTATION

INSTRUMENT	REQUIRED NO. OF CHANNELS	MINIMUM CHANNELS OPERABLE
1. Reactor Coolant T_{Hot} (Wide Range)	2	1
2. Reactor Coolant T_{Cold} (Wide Range)	2	1
3. Containment Pressure (Wide Range)	2	1
4. Refueling Water Storage Tank Level	2	1
5. Reactor Coolant Pressure (Wide Range)	2	1
6. Pressurizer Level (Wide Range)	2	1
7. Steam Line Pressure	2/steam line	1/steam line
8. Steam Generator Level - (Wide Range)	1/steam generator	1/steam generator
9. Steam Generator Level - (Narrow Range)	1/steam generator	1/steam generator
10. Auxiliary Feedwater Flow Rate	1/pump	1/pump
11. Reactor Coolant System Subcooling Margin Monitor	1	0
12. Pressurizer PORV Position Indicator*	2/valve#	1/valve
13. Pressurizer PORV Block Valve Position Indicator**	2/valve	1/valve
14. Safety Valve Position Indicator	2/valve#	1/valve
15. Containment Water Level (Wide Range)	2	1
16. In Core Thermocouples	4/core quadrant	2/core quadrant
17. Reactor Vessel Level Instrumentation System***	2	1
18. Source & Intermediate Range Nuclear Instrumentation	2	1

ADD

*Not applicable if the associated block valve is in the closed position.

**Not applicable if the block valve is verified in the closed position with power to the valve operator removed.

***This Technical Specification and surveillance requirement will not be implemented until Sequoyah Specific Instructions are developed for the use of this system as committed to in the TVA response to Supplement 1 of NUREG-0737.

#At least one channel shall be the acoustic monitors.

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NEP 6.6 Sheet 32 | R3
Safety Evaluation
No. ECN L6186.

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Reactor Coolant T_{Hot} (Wide Range)	M	R
2. Reactor Coolant T_{Cold} (Wide Range)	M	R
3. Containment Pressure (Wide Range)	M	R
4. Refueling Water Storage Tank Level	M	R
5. Reactor Coolant Pressure (Wide Range)	M	R
6. Pressurizer Level	M	R
7. Steam Line Pressure	M	R
8. Steam Generator Level - Wide	M	R
9. Steam Generator Level - Narrow	M	R
10. Auxiliary Feedwater Flowrate	M	R
11. Reactor Coolant System Subcooling Margin Monitor	M	R
12. Pressurizer PORV Position Indicator	M	R
13. Pressurizer PORV Block Valve Position Indicator	M	R
14. Safety Valve Position Indicator	M	R
15. Containment Water Level (Wide Range)	M	R
16. In Core Thermocouples	M	R
17. Reactor Vessel Level Instrumentation**	M	R
18. Source & Intermediate Range Nuclear Instrumentation	M	R

ADD

R50
R50
R50
R50

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**This Technical Specification and surveillance requirement will not be implemented until Sequoyah Specific Instructions are developed for the use of this system as committed to in the TVA response to Supplement 1 of NUREG-0737. R50

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Safety Evaluation
No. ECN L6186

TABLE 4.3-7

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	
1. Reactor Coolant T_{Hot} (Wide Range)	M	R	
2. Reactor Coolant T_{Cold} (Wide Range)	M	R	
3. Containment Pressure (Wide Range)	M	R	
4. Refueling Water Storage Tank Level	M	R	R38
5. Reactor Coolant Pressure (Wide Range)	M	R	
6. Pressurizer Level	M	R	R38
7. Steam Line Pressure	M	R	
8. Steam Generator Level - (Wide)	M	R	
9. Steam Generator Level - (Narrow)	M	R	
10. Auxiliary Feedwater Flowrate	M	R	
11. Reactor Coolant System Subcooling Margin Monitor	M	R	
12. Pressurizer PORV Position Indicator	M	R	
13. Pressurizer PORV Block Valve Position Indicator	M	R	
14. Safety Valve Position Indicator	M	R	
15. Containment Water Level (Wide Range)	M	R	
16. In Core Thermocouples	M	R	R38
17. Reactor Vessel Level Instrumentation System*	M	R	
18. Source : Intermediate Range Nuclear Instrumentation	M	R	R38

ADD

ECN NO. L6186 E3
BD OF

*This Technical Specification and surveillance requirement will not be implemented until Sequoyah Specific Instructions are developed for the use of this system as committed to in the TVA response to Supplement 1 of NUREG-0737. R38

NEP 6.6 Sheet 34 | R3
Safety Evaluation
No. ECN L6186

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES	
13. Steam Generator Water Level--Low-Low	≥ 18% of narrow range instrument span-each steam generator	≥ 17% of narrow range instrument span-each steam generator	R20
14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	< 40% of full steam flow at RATED THERMAL POWER coincident with steam generator water level ≥ 25% of narrow range instrument span--each steam generator	< 42.5% of full steam flow at RATED THERMAL POWER coincident with steam generator water level ≥ 24.0% of narrow range instrument span--each steam generator	
15. Undervoltage-Reactor Coolant Pumps	≥ 5022 volts-each bus	≥ 4739 volts-each bus	R89
16. Underfrequency-Reactor Coolant Pumps	≥ 56.0 Hz - each bus	≥ 55.9 Hz - each bus	
17. Turbine Trip			
A. Low Trip System Pressure	≥ 45 psig	≥ 43 psig	
B. Turbine Stop Valve Closure	≥ 1% open	≥ 1% open	
18. Safety Injection Input from ESF	Not Applicable	Not Applicable	
19. Intermediate Range Neutron Flux - (P-6) Enable Block Source Range Reactor Trip	$\geq 1 \times 10^{-5}$ % ~~$\geq 1 \times 10^{-10}$ amps~~ OF RATED THERMAL POWER	$\geq 6 \times 10^{-6}$ % ~~$\geq 6 \times 10^{-11}$ amps~~ OF RATED THERMAL POWER	REVISE
20. Power Range Neutron Flux (not P-10) Input to Low Power Reactor Trips Block P-7	< 10% of RATED THERMAL POWER	< 11% of RATED THERMAL POWER	

R89 R89

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NEP 6.6 Sheet 35|R3
Safety Evaluation
No. ECN L6186

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
13. Steam Generator Water Level--Low-Low	≥ 18% of narrow range instrument span-each steam generator	≥ 17% of narrow range instrument span-each steam generator
14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	≤ 40% of full steam flow at RATED THERMAL POWER coincident with steam generator water level ≥ 25% of narrow range instrument span--each steam generator	≤ 42.5% of full steam flow at RATED THERMAL POWER coincident with steam generator water level ≥ 24% of narrow range instrument span--each steam generator
15. Undervoltage-Reactor Coolant Pumps	≥ 5022 volts-each bus	≥ 4739 volts-each bus
16. Underfrequency-Reactor Coolant Pumps	≥ 56 Hz - each bus	≥ 55.9 Hz - each bus
17. Turbine Trip		
A. Low Trip System Pressure	≥ 45 psig	≥ 43 psig
B. Turbine Stop Valve Closure	≥ 1% open	> 1% open
18. Safety Injection Input from ESF	Not Applicable	Not Applicable
19. Intermediate Range Neutron Flux, P-6, Enable Block Source Range Reactor Trip	$\geq 1 \times 10^{-5}$ % ~~≥ 1 x 10-10 amps~~ OF RATED THERMAL POWER	$\geq 6 \times 10^{-6}$ % ~~≥ 6 x 10-11 amps~~ OF RATED THERMAL POWER
20. Power Range Neutron Flux (not P-10) Input to Low Power Reactor Trips Block P-7	< 10% of RATED THERMAL POWER	≤ 11% of RATED THERMAL POWER

REVISE

R

R76

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UNREVIEWED SAFETY QUESTION DETERMINATION

Sheet 36

TVA 10551 (EN DES-2-81)

SHEET #1
L6186 R.0
IDENTIFIER

ECN NO. L6186 R3
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TO: Sequoyah Nuclear Plant, Daisy, TN

FROM: F16 SNP

MEDS ACCESSION NO.

REV NO.	TOT PP	PREPARED	REVIEWED	APPROVED	DATE APPD	DPM RLSE
0*	3	R.A. Edlund	R.R. Campbell	P.H. Chen/VAB	1/14/85	JPV
1						
2						
3						
4						
5						

*R0 INITIAL ISSUE

R0	SQP '85 0117 503
R1	
R2	
R3	
R4	
R5	

PROJECT SQN AFFECTED UNIT(S) 1 & 2

USQ? YES ___ NO x

ECN NO. L6186 ECN DATE 12-21-84

~~PCR/NCR/~~
~~MCR/~~DCR NO. 1156 DATE ______

	YES/NO	SHEET NO.
SPECIAL REQUIREMENT(S)	NO	
POTENTIAL TECH SPEC CHANGE	yes	3

OTHER ______ DATE ______

REFERENCES: USQD Support Information Sheet
Conversation with Jerry Hatcher, SQEP elec

DESCRIPTION OF CHANGE Neutron Monitoring
Upgrade the source range neutron detection equipment to meet Class 1E and seismic category I requirements per Reg Guide 1.97 (Rev 2). This modification is necessary to provide more reliable postaccident monitoring instrumentation. The equipment will require 120-V Class 1E [illegible] and will draw approximately 1 amp. The equipment will also be environmentally qualified per Jerry Hatcher.

INFORMATION ONLY

REF: EN DES EP's 2.03 & 4.02

CC (ATTACHMENTS): NO - YES
CHIEF, ARCHITECTURAL DESIGN BRANCH, W4C126 C-K
CHIEF, CIVIL ENGINEERING BRANCH, W9D224 C-K
CHIEF, CIVIL EN DES BRANCH, W3C126 C-K
CHIEF, ELECTRICAL ENGINEERING BRANCH, W8C126 C-K
CHIEF, ELECTRICAL EN DES BRANCH, W2D224 C-K
CHIEF, MECHANICAL ENGINEERING BRANCH, W10D225 C-K
MEDS, E4B37 C-K
CHIEF NUCLEAR ENGINEER, W10C126 C-K
CHIEF, MECHANICAL EN DES BRANCH, 1025PT-K
CHIEF, QUALITY ASSURANCE BRANCH, W11C126 C-K
MANAGER OF CONSTRUCTION, E7B24 C-K
CHIEF, COST PLANNING AND CONTROL STAFF, W12C74 C-K
PLANT SUPERINTENDENT
DIRECTOR, NUCLEAR POWER DIVISION, 716 EB-C

UNREVIEWED SAFETY QUESTION DETERMINATION

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Unreviewed Safety Question:

1. Is the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report increased? . . . . Yes ____ No X

Justification: The neutron monitoring system does not perform any function necessary to maintain the unit in a safe condition. The system is being upgraded to ensure its operability following an accident, and thus provide the operators with additional information regarding the condition of the core. Considering that the new equipment will exceed the design requirements of the equipment being replaced, the probability of occurrence or the consequences of an accident previously evaluated in the SAR is not increased.

2. Is the possibility for an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report created? . . . . Yes ____ No X

Justification: The new equipment will be seismic category I and Class IE. Therefore, the system will exceed the original design requirements, and the possibility for an accident or malfunction of a different type is not created. In addition, an evaluation will be made to ensure that the additional load on the Class IE power system will not have any adverse affects.

3. Is the margin of safety as defined in the basis for any technical specification reduced? . . . . Yes ____ No X

This modification serves to improve the accident monitoring capability of the plant. Thus, the margin of safety as defined in tech spec 3/4.3.3.7 (Accident Monitoring Instrumentation) is not reduced. Per the note on page 3, this tech spec may be revised to include this instrumentation.

UNREVIEWED SAFETY QUESTION DETERMINATION

Sheet # ~~3~~ 38

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Project SQN

Special Requirement(s) or Precaution(s) .. ☐

Safety Evaluation .... ☐

Additional Information .... ☒

(☒, Marks Subject to Which This Page Applies)

Preparer RAE

Reviewer RRC (initial)

Potential Tech Spec Change:

NUC PR should review applicable tech specs on accident monitoring instrumentation to determine if any changes are required.

INFORMATION ONLY

THIS CHANGE IS NO LONGER REQUIRED PER THE DISCUSSION IN SECTION 27.7 OF THIS SAFETY EVALUATION

R3

NEP 6.6 SHEET 39 / R3
SAFETY EVALUATION
NO. ECN L6186

TABLE 3.3-9

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

ADD

ADD

INSTRUMENT	READOUT LOCATION	MEASUREMENT RANGE	MINIMUM CHANNELS OPERABLE
1. Source Range/ INTERMEDIATE Nuclear Flux	NOTE 1	1 to 1 x 10^6 cps / 10^{-8} TO 200% RTP	1 "
2. Reactor Trip Breaker Indication	at trip switchgear	OPEN-CLOSE	1/trip breaker
3. Reactor Coolant Temperature - Hot Leg	NOTE 1	0-650°F	1/loop R80
4. Pressurizer Pressure	NOTE 1	0-3000 psig	1
5. Pressurizer Level	NOTE 1	0-100%	1 R80
6. Steam Generator Pressure	NOTE 1	0-1200 psig	1/steam generator
7. Steam Generator Level	NOTE 2 or near Auxilary F. W. Pump	0-100%	1/steam generator
8. Full Length Control Rod Position Limit Switches	Auxilary Instrument Room: Racks R41-44	On-off	1 insertion limit switch/rod
9. RHR Flow Rate	NOTE 1	0-4500 gpm	1
10. RHR Temperature	NOTE 1	50-400°F	1
11. Auxiliary Feedwater Flow Rate	NOTE 1	0-440 gpm	1/steam generator R80

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THIS CHANGE IS NO LONGER REQUIRED PER THE DISCUSSION IN SECTION 27.7 OF THE SAFETY EVALUATION R3

NEP 6.6 SHEET 40/R3
SAFETY EVALUATION
NS. ECN L6186

TABLE 3.3-9

REMOTE SHUTDOWN MONITORING INSTRUMENTATION

ADD

ADD

INSTRUMENT	READOUT LOCATION	MEASUREMENT RANGE	MINIMUM CHANNELS OPERABLE	
1. ~~Source~~ Intermediate Range Nuclear Flux	NOTE 1	1 to 1 x 10^6 cps 10^{-8} to 200% RTP	1	
2. Reactor Trip Breaker Indication	at trip switchgear	OPEN-CLOSE	1/trip breaker	
3. Reactor Coolant Temperature - Hot Leg	NOTE 1	0-650°F	1/loop	R67
4. Pressurizer Pressure	NOTE 1	0-3000 psig	1	
5. Pressurizer Level	NOTE 1	0-100%	1	
6. Steam Generator Pressure	NOTE 1	0-1200 psig	1/steam generator	R67
7. Steam Generator Level	NOTE 2 or near Auxilary F. W. Pump	0-100%	1/steam generator	
8. Full Length Control Rod Position Limit Switches	Auxilary Instrument Room: Racks R41-44	On-off	1 insertion limit switch/rod	
9. RHR Flow Rate	NOTE 1	0-4500 gpm	1	
10. RHR Temperature	NOTE 1	50-400°F	1	
11. Auxiliary Feedwater Flow Rate	NOTE 1	0-440 gpm	1/steam generator	R67

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THIS CHANGE IS NO LONGER REQUIRED PER THE DISCUSSION IN SECTION 27.7 OF THE SAFETY EVALUATION | R3

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SAFETY EVALUATION
NO. ECN L6186

ADD

INTERMEDIATE

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION
1. Source Range Nuclear Flux	M	R
2. Reactor Trip Breaker Indication	M	N.A.
3. Reactor Coolant Temperature - Hot Leg	M	R
4. Pressurizer Pressure	M	R
5. Pressurizer Level	M	R
6. Steam Generator Pressure	M	R
7. Steam Generator Level	M	R
8. Full Length Control Rod Position Limit Switches	M	R
9. RHR Flow Rate	M	R
10. RHR Temperature	M	R
11. Auxiliary Feedwater Flow Rate	M	R
12. Pressurizer Relief Tank Pressure	M	R
13. Containment Pressure	M	R

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THIS CHANGE IS NO LONGER REQUIRED PER THE DISCUSSION IN SECTION 27.7 OF THE SAFETY EVALUATION. | R3

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SAFETY EVALUATION
NO. ECNL6186

SEQUOYAH - UNIT 2

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION
SURVEILLANCE REQUIREMENTS

ADD

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	
1. Source Range INTERMEDIATE Nuclear Flux	M	R	
2. Reactor Trip Breaker Indication	M	N.A.	
3. Reactor Coolant Temperature - Hot Leg	M	R	
4. Pressurizer Pressure	M	R	
5. Pressurizer Level	M	R	
6. Steam Generator Pressure	M	R	
7. Steam Generator Level	M	R	
8. Full Length Control Rod Position Limit Switches	M	R *	R20
9. RHR Flow Rate	M	R	
10. RHR Temperature	M	R	
11. Auxiliary Feedwater Flow Rate	M	R	
12. Pressurizer Relief Tank Pressure	M	R	
13. Containment Pressure	M	R	

* For cycle 1, this surveillance is to be completed before the next cooldown or by August 5, 1983 whichever is earlier. R20

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Range Channels will initiate a reactor trip at ~~a current level proportional to~~ approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

Overtemperature Delta T

The Overtemperature Delta T trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the High and Low Pressure reactor trips. This setpoint includes corrections for axial power distribution, changes in density and heat capacity of water with temperature and dynamic compensation for piping delays from the core to the loop temperature detectors. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table 2.2-1. R3

Operation with a reactor coolant loop out of service below the 4 loop P-8 setpoint does not require reactor protection system setpoint modification because the P-8 setpoint and associated trip will prevent DNB during 3 loop operation exclusive of the Overtemperature Delta T setpoint. Three loop operation above the 4 loop P-8 setpoint is permissible after resetting the K1, K2 and K3 inputs to the Overtemperature Delta T channels and raising the P-8 setpoint to its 3 loop value. In this mode of operation, the P-8 interlock and trip functions as a High Neutron Flux trip at the reduced power level.

Overpower Delta T

The Overpower Delta T reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature Delta T protection, and provides a backup to the High Neutron Flux trip. The setpoint includes corrections for changes in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. No credit was taken for operation of this trip in the accident

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BASES

Intermediate and Source Range, Nuclear Flux (Continued)

Range Channels will initiate a reactor trip at ~~a current level proportional to~~ approximately 25 percent of RATED THERMAL POWER unless manually blocked when P-10 becomes active. No credit was taken for operation of the trips associated with either the Intermediate or Source Range Channels in the accident analyses; however, their functional capability at the specified trip settings is required by this specification to enhance the overall reliability of the Reactor Protection System.

Overtemperature ΔT

The Overtemperature delta T trip provides core protection to prevent DNB for all combinations of pressure, power, coolant temperature, and axial power distribution, provided that the transient is slow with respect to piping transit delays from the core to the temperature detectors (about 4 seconds), and pressure is within the range between the High and Low Pressure reactor trips. This setpoint includes corrections for axial power distribution, changes in density and heat capacity of water with temperature and dynamic compensation for piping delays from the core to the loop temperature detectors. With normal axial power distribution, this reactor trip limit is always below the core safety limit as shown in Figure 2.1-1. If axial peaks are greater than design, as indicated by the difference between top and bottom power range nuclear detectors, the reactor trip is automatically reduced according to the notations in Table 2.2-1. R3

Operation with a reactor coolant loop out of service below the 4 loop P-8 setpoint does not require reactor protection system set point modification because the P-8 setpoint and associated trip will prevent DNB during 3 loop operation exclusive of the Overtemperature delta T setpoint. Three loop operation above the 4 loop P-8 setpoint is permissible after resetting the K1, K2, and K3 inputs to the Overtemperature delta T channels and raising the P-8 setpoint to its 3 loop value. In this mode of operation, the P-8 interlock and trip functions as a High Neutron Flux trip at the reduced power level.

Overpower ΔT

The Overpower delta T reactor trip provides assurance of fuel integrity, e.g., no melting, under all possible overpower conditions, limits the required range for Overtemperature delta T protection, and provides a backup to the High Neutron Flux trip. The setpoint includes corrections for changes in density and heat capacity of water with temperature, and dynamic compensation for piping delays from the core to the loop temperature detectors. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

TVA

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REVISION LO

Title: Safety Evaluation for ECN L6186

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Revision No.	DESCRIPTION OF REVISION	Date Approv
0	Initial Issue	
1	Revised ECN L6186 USQD (RIMS SQP 850117 503) to incorporate NEP 6.6 requirements. RO of this USQD is attached for information only on sheets ~~34, 35, & 36~~ 36, 37, & 38 | R2	
2	Incorporate plant licensing comments to include boron dilution fault in safety evaluation and correlate question #27 items to implementing stages of ECN L6186. Add new sheets 8 and 14, and revise safety evaluation sheets accordingly.	
3	Incorporated NSRB reviewer and initial calibration comments, source range outputs disabled change and various clarifications. Also added special requirements and Tech Spec Bases change. This revision also identified electrical load changes which resulted in revisions to FSAR chapter 8. Sheets 45 through 62 were added to include the FSAR changes required.	

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As discussed in Subparagraph 7.2.1.1.2, factors included in establishing the Overtemperature ΔT and Overpower ΔT trip setpoints includes the reactor coolant temperature in each loop and the axial distribution of core power through the use of the two section ex-core neutron detectors.

4.4.5.3 Instrumentation to Limit Maximum Power Output

The outputs ~~output~~ of the three ranges (source, intermediate, and power) ~~of detectors, with the electronics~~ of ~~the~~ nuclear instruments, are used to limit the maximum power output of the reactor within their respective ranges.

There are six radial locations. ~~containing a total of eight neutron flux detectors installed around the reactor in the primary shield~~, two fission chamber assemb ~~proportional counters~~ for the source/intermediate range installed on opposite "flat" portions of the core containing the primary startup sources. ~~at an elevation approximately one quarter of the core height. Two compensated ionization chambers for the intermediate range, located in the same instrument wells and detector assemblies as the source range detectors, are positioned at an elevation corresponding to one half of the core height.~~ Four dual section uncompensated ionization chamber assemblies for the power range installed vertically at the four corners of the core and located equidistant from the reactor vessel at all points and, to minimize neutron flux pattern distortions, within one foot of the reactor vessel. Each power range detector provides two signals corresponding to the neutron flux in the upper ~~and in the~~ lower sections of a core quadrant. The three ranges of ~~detectors~~ are used as inputs to monitor neutron flux from a completely shutdown condition to 120 percent of full power with the capability of recording overpower excursions up to 200 percent of full power.

The difference in neutron flux between the upper and lower sections of the power range detectors are used to limit the Overtemperature ΔT and Overpower ΔT trip setpoints and to provide the operator with an indication of the core power axial offset. In addition, the output of the power range channels are used for:

1. the rod speed control function,

2. to alert the operator to an excessive power unbalance between the quadrants,

3. protecting the core against the consequences of rod ejection accidents, and

4. protecting the core against the consequences of adverse power distributions resulting from dropped rods.

Details of the neutron detectors and nuclear instrumentation design and the control and trip logic are given in Chapter 7. The limits on neutron flux operation and trip setpoints are given in the SQN Technical Specifications.

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a. Power range high neutron flux trip.

The power range high neutron flux trip circuit trips the reactor when two of the four power range channels exceed the trip setpoint.

There are two independent bistables each with their own trip setting (a high and a low setting) per channel (four channels total). The high trip setting provides protection during normal power operation and is always active. The low trip setting, which provides protection during startup, can be manually bypassed when two out of the four power range channels read above approximately 10 percent power (P-10). Three out of the four channels below 10 percent automatically reinstates the trip function. Refer to Table 7.2.1-2 for a listing of all protection system interlocks. 6

b. Intermediate range high neutron flux trip

The intermediate range high neutron flux trip circuit trips the reactor when one out of the two intermediate range channels exceed the trip setpoint. This trip, which provides protection during reactor startup, can be manually blocked if two out of four power range channels are above approximately 10 percent power (P-10). Three out of the four power range channels below this value automatically reinstates the intermediate range high neutron flux trip. The intermediate range channels (including detectors) are separate from the power range channels. The intermediate range channels can be individually bypassed at the nuclear instrumentation racks to permit channel testing at any time under prescribed administrative procedures and only under the direction of authorized supervision. This bypass action is annunciated on the control board.

(within 7.2.1.1.2.)

c. Source range high neutron flux trip

The source range high neutron flux trip circuit trips the reactor when one of the two source range channels exceeds the trip setpoint. This trip, which provides protection during reactor startup and plant shutdown, can be manually bypassed when one of the two intermediate range channels reads above the P-6 setpoint value (source range ~~cutoff~~ power level) and is automatically reinstated when both intermediate range channels decrease below the P-6 value. This trip is also automatically bypassed by two out of four logic from the power range permissive (P-10).

outputs disabled and intermediate range on scale

This trip function can also be reinstated below P-10 by an administrative action requiring manual actuation of two control board mounted switches. Each switch will reinstate the trip function in one of the two protection logic trains. The source range trip is set between the P-6 setpoint and the maximum source range level. The channels can be individually blocked at the nuclear instrumentation racks to permit channel testing at any

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The auto stop oil pressure signal also dumps the stop emergency trip fluid, closing all of the turbine steam stop valves. When all stop valves are closed, a reactor trip signal will be initiated if the reactor is above P-9 setpoint. This trip signal is generated by redundant (two each) limit switches on the stop valves. 6

7. Safety Injection Signal Actuation Trip

A reactor trip occurs when the Safety Injection System is actuated. The means of actuating the Safety Injection System are described in Section 7.3. This trip protects the core against a loss of primary or secondary coolant. 6

Figure 7.2.1-1, Sheet 8, shows the logic for this trip. A detailed functional description of the process equipment associated with this trip function is provided in Reference 1.

8. Manual Trip

The manual trip consists of two switches with two outputs on each switch. One output is used to actuate the train A trip breaker, the other output actuates the train B trip breaker. Operating a manual trip switch removes the voltage from the undervoltage trip coil and energizes the shunt reactor trip breaker trip coil. 6

There are no interlocks which can block this trip. Figure 7.2.1-1, Sheet 3, shows the manual trip logic.

7.2.1.1.3 Reactor Trip System Interlocks

1. Power Escalation Permissives

The overpower protection provided by the out of core nuclear instrumentation consists of three discrete, but overlapping, levels. Continuation of startup operation or power increase requires a permissive signal from the higher range instrumentation channels before the lower range level trips can be manually blocked by the operator.

A one out of two intermediate range permissive signal (P-6) is required prior to source range level trip blocking and ~~detector high~~ source range ~~voltage cutoff~~ outputs disabled. Source range level trips are automatically reactivated and ~~high voltage~~ outputs restored when both intermediate range channels are below the permissive (P-6) level. There is a manual reset switch for administratively reactivating the source range level trip and ~~detector high voltage~~ outputs when between the permissive P-6 and P-10 level, if required. 6

Source range level trip block and ~~high voltage cutoff~~ outputs disabled are always maintained when above the permissive P-10 level.

7.2.1.1.5 Pressurizer Water Level Reference Leg Arrangement

The design of the pressurizer water level instrumentation includes a slight modification of the usual tank level arrangement using differential pressure between an upper and a lower tap. The modification shown in Figure 7.2.1-3, consists of the use of a sealed reference leg instead of the conventional open column of water. Refer to 7.2.2.3.4 for an analysis of this arrangement.

7.2.1.1.6 Analog System

The process analog system is described in Reference 1.

7.2.1.1.7 Solid State Logic Protection System

The solid state logic protection system takes binary inputs (voltage/no voltage) from the process and nuclear instrument channels corresponding to conditions (normal/abnormal) of plant parameters. The system combines these signals in the required logic combination and generates a trip signal (no voltage) to the undervoltage coils of the reactor trip circuit breakers when the necessary combination of signals occur. The system also provides annunciator, status light and computer input signals which indicate the condition of bistable input signals, partial trip and full trip functions and the status of the various blocking, permissive and actuation functions. In addition the system includes means for semi-automatic testing of the logic circuits. A detailed description of this system is given in Reference 3.

7.2.1.1.8 Isolation Amplifiers

In certain applications, Westinghouse considers it advantageous to employ control signals derived from individual protection channels through isolation amplifiers contained in the protection channel, as permitted by IEEE-279.

In all of these cases, analog signals derived from protection channels for non-protective functions are obtained through isolation amplifiers located in the analog protection racks. By definition, non-protective functions include those signals used for control, remote process indication, and computer monitoring.

Isolation amplifier qualification tests are described in References 4, ~~and~~ 5 and 17

7.2.1.1.9 Energy Supply and Environmental Variations

The energy supply for the Reactor Trip System, including the voltage and frequency variations, is described in Section 7.6. The environmental variations, throughout which the system will perform, are given in Section 3.11.

partial trip alarm and channel status light actuation in the control room. Each channel contains those switches, test points, etc. necessary to test the channel. See Reference 1 for additional information.

The power range channels of the Nuclear Instrumentation System are tested by superimposing a test signal on the actual detector signal being received by the channel at the time of testing. The output of the bistables is not placed in a tripped condition prior to testing. Also, since the power range channel logic is two out of four, bypass of this reactor trip function is not required.

To test a power range channel, a "TEST-OPERATE" switch is provided to require deliberate operator action and operation of which will initiate the "CHANNEL TEST" annunciator in the control room. Bistable operation is tested by increasing the test signal level up to its trip setpoint and verifying bistable relay operation by control board annunciator and trip status lights.

It should be noted that a valid trip signal would cause the channel under test to trip at a lower actual reactor power level. A reactor trip would occur when a second bistable trips. No provision has been made in the channel test circuit for reducing the channel signal level below that signal being received from the Nuclear Instrumentation System detector.

A Nuclear Instrumentation System channel which can cause a reactor trip through one of two protection logic (source or intermediate range) is provided with a bypass function which prevents the initiation of a reactor trip from that particular channel during the short period that it is undergoing test. These bypasses initiate an alarm in the control room.

For a detailed description of the Nuclear Instrumentation System see References 2A and 17

The logic trains of the Reactor Trip System are designed to be capable of complete testing at power, except for those trips listed in Subsection 7.2.3. Annunciation is provided in the control room to indicate when a train is in test, when a reactor trip is bypassed and when a reactor trip breaker is bypassed. Details of the logic system testing are given in Reference 3.

The reactor coolant pump breakers cannot be tripped at power without causing a plant upset by loss of power to a coolant pump. However, the reactor coolant pump breaker open trip logic and continuity through the shunt trip coil can be tested at power. Manual trip cannot be tested at power without causing a reactor trip since operation of either manual trip switch actuates both Train A and Train B. Note, however, that manual trip could also be initiated from outside the control room by manually tripping one of the reactor

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(within 7.2.4)

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11. The Institute of Electrical and Electronic Engineers, Inc., "IEEE Trial-Use Standard; General Guide for Qualifying Class I Electric Equipment for Nuclear Power Generating Stations," IEEE Std. 323-1971.

12. The Institute of Electrical and Electronic, Inc., "IEEE Trial-Use Guide for Type Tests of Continuous-Duty Class I Motor Installed Inside the Containment of Nuclear Power Generating Stations," IEEE Std. 334-1971.

13. The Institute of Electrical and Electronic Engineers, Inc., "IEEE Trial-Use Guide for Seismic Qualification of Class I Electric Class I Electric Equipment for Nuclear Power Generating Stations," IEEE Std. 344-1971.

14. "General Design Criteria for Nuclear Power Plants," Appendix A to Title 10 CFR 50, July 7, 1971.

15. E. P. Rahe, "Evaluation of Surveillance Frequencies and Out of Service Times for Reactor Protection System," WCAP 10271 and Supplement 1 (Westinghouse NES Proprietary). 4

16. W. H. Moomau, "Westinghouse Setpoint Methodology for Protection Systems, Sequoyah Units 1 and 2," WCAP 11239, Rev. 3, October 1987 (Westinghouse Proprietary Class 2). 6

17. TVA Environmental Qualication Binder SQN EQ-NM-001, Gamma-Metrics, Neutron Flux Monitoring.

(within section 8.3.1.2.3)

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Electrical Penetration Assemblies

The Westinghouse electrical penetration assemblies have been tested to TVA specification requirements which conform to IEEE-317, 1971, "IEEE Standard for Electrical Penetration Assemblies in Containment Structures for Nuclear Power Generating Stations." The Conax electrical penetration assemblies meet the 1976 version of IEEE-317.
The documentation of successful completion included certified test reports of all tests required and listed in the specifications and quality assurance appendix, and applicable TVA inspector's reports.

Each electrical penetration assembly furnished has been shop inspected by a commissioned representative of the National Board of Boiler and Pressure Vessel Inspectors. Each assembly has been Code stamped, in accordance with the 1971 Edition ASME Boiler and Pressure Vessel Code, Section III.

The dose rate at which TVA has conducted 100 hour tests on materials and equipment is 10^6 Rad/hr dose rate that may occur during the first hour of a LOCA. It is the TVA position that a factor of 5 in dose rate is not significant in this region. There is no mechanism that TVA is aware of that would tend to produce significant increases in degradation in the region between 10^6 and 10^7 Rad/hr. However, radiation-induced oxidation of materials can become an important damage mechanism at lower exposure rates and consequent longer exposure times. Therefore, IEEE 278, "Guide For Classifying Electrical Insulating Materials Exposed to Neutron and Gamma Radiation," recommend using exposure rates above 10^7 Rad/hr. It is the TVA position that 10^6 Rad/hr for 100 hours represents a reasonable and conservative combination of dose rate and exposure time for radiation testing. 6

Cable terminations to low voltage power, control, and indication penetration assemblies are generally made in all metal splice boxes. However, in a number of instances on the outboard side of containment electrical penetrations, field cables were spliced to the penetration pigtails in cable trays. In these cases, a special enclosure was used to act as a qualified fire stop (refer to Figure 8.3.1-37, -38, and -39). These particular splices are located within the last 5-foot section of the cable tray. The trays in the annulus area of containment containing these splices are fitted with solid top and bottom covers in the immediate area of these splices. A qualified fire barrier made of silicone foam and ceraform/kaowool fiberboard was installed on the side of the splice opposite to the penetration as shown on Figure 8.3.1-37, -38, and -39. On the other side of the splice in the tray (end of tray runs toward the electrical penetration), kaowool materials were inserted in the voids between conductors, and all the exposed conductors to the electrical penetration were covered with Flamemastic material. This configuration constitutes a qualified fire barrier which in the unlikely event of a fire in the splice area, will contain and isolate the fire from adjacent trays of electrical equipment. 6

operation are considered in this analysis. Table 15.2-1 contains the time sequence of events for this accident.

Dilution During Refueling

An uncontrolled boron dilution accident cannot occur during refueling. This accident is prevented by administrative controls which isolate the RCS from the potential source of unborated water.

Various valve combinations that are required to be locked closed during refueling operations are specified in technical specification 3.9.1. These valves will block the flow paths which could allow unborated makeup to reach the RCS. Any makeup which is required during refueling will be borated water supplied either from the refueling water storage tank by the low head safety injection pumps or the centrifugal charging pumps, or from the boric acid tanks via a boric acid transfer pump and a centrifugal charging pump. 6

Dilution During Startup

Prior to startup the RCS is filled with borated (approximately 2000 ppm) water from the refueling water storage tank.

Nuclear instrumentation is monitored closely in anticipation of an unplanned reactivity rate of change. Mixing of the reactor coolant is accomplished by operation of the reactor coolant pumps. High source range flux level and all reactor trip alarms are effective. 6

In the analysis, a maximum dilution flow of 300 gpm limited by the capacity of the two primary water makeup pumps is considered. The volume of the reactor coolant is approximately 9967 ft^3, which is the active volume of the Reactor Coolant System excluding the pressurizer.

Dilution Following Reactor Shutdown

Following reactor shutdown, when in hot standby, hot shutdown, and subsequent cold shutdown condition, and once below the P-6 interlock setpoint, ~~in accordance with Sequoyah Nuclear Plant Units 1 and 2~~ and 10^4 counts per second ~~Surveillance Instruction CO3, (High Flux Adjustment After Shutdown)~~, the high flux at shutdown alarm setting will be automatically adjusted downward to no higher than ~~1/2 decade above~~ 3 times the background count rate as ~~30 minutes after plant shutdown~~ reduces. ~~The alarm setpoint must be set or verified every 30 minutes for the first 2 hours following plant trip, every 2 hours for the next 6 hours, and once per shift thereafter until the flux level has stabilized.~~ Surveillance testing will ensure that the alarm setpoint is operable

Dilution at Power

With the unit at power and the RCS at pressure, the dilution rate is limited by the capacity of the primary water makeup pumps which supply the charging pump. A conservatively high value for the expected boron concentration (1575 ppm) at power and a conservatively high dilution flow rate of 300 gpm was used. 1 6

The operator does not depend entirely on this alarm setpoint but has audible indication of increasing neutron flux from the audible count rate drawer and visual indication from counts per second meters for each channel on the main control board and source range drawer.

15.2.4.3 Conclusions

For dilution during refueling:

Dilution during refueling cannot occur due to administrative controls (see Section 15.2.4.2).

The operator has prompt and definite indication of any boron dilution from the audible count rate instrumentation. At all times during refueling the source range audibl ~~High~~ count rate is ~~alarmed~~ on the refuel floor of ~~in the~~ reactor containment and the main control room. In addition, a high source range flux level is alarmed in the control room And in the reactor containment. The count rate increase is proportional to the subcritical multiplication factor.

For dilution during startup:

For dilution during startup, there is adequate time (~52 minutes) from transient initiation for the operator to recognize the high count rate signal and terminate manually the source of dilution flow. 6

The operator is alerted to the uncontrolled reactivity insertion during startup via the increasing count rate on the Source Range Nuclear Instrumentation. Recorders on the control board continuously provide a time history of the nuclear flux level. This increase in flux level is very slow, based on the reactivity insertion rate for the startup case, it takes approximately 42 minutes for the flux level to increase by a factor of 2. This is adequate time for the operator to recognize from the recorders the need for action. Thus there would still be 48 minutes for the operator to ascertain and isolate the source of the reactivity insertion. Also on the source range channel is the high flux at shutdown alarm. The setpoint for this alarm is normally placed at 5^3 times the source level. Even assuming that the operator doesn't recognize the increasing count rate, the alarm will occur at approximately 70 minutes into the transient. Thus there would still be 21 minutes for the operator to stop the dilution. 1

For dilution during full power operation:

1. With the reactor in automatic control, the power and temperature increase from boron dilution results in insertion of the rod cluster control assemblies and a decrease in the shutdown margin.

   The operator is alerted to an uncontrolled reactivity insertion by the rod insertion limit alarms. Two insertion limit alarms are available: The first occurs when the rods are 10 steps above the insertion limit (Lo Insertion Limit Alarm) and the second occurs at the insertion limit (Lo-Lo Insertion Limit Alarm). The analysis assumed that the operator is alerted to the need for action by the Lo-Lo Alarm although action would be taken when the first alarm occurs. Thus the analysis already assumes a 10 step allowance for 1

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TABLE 15.2.4-1

SEQUENCE OF EVENTS

Equilibrium Xe Case	Time (sec)
Reactor Trip	0
Reactor Power = 7.5% of nominal 80 sec reactor period Intermediate NIS reads ∫ ~~10^{-8} amps~~ 10^{-6} % POWER	10
Source Range NIS Available	930
Source Range NIS no longer decreasing (without dilution event, flux would stabilize at this point - an 18 day half life decay of flux would be normal*).	1,250
~~Operator set~~ High Flux ~~to~~ at Shutdown Alarm ~~0.5 decade above~~ 3 times stabilized flux level. For this example, the value is ∫ 700 cps.	1,250
High pressurizer Level Trip and Alarm	1,800
Source Range High Flux at Shutdown Alarm	7,400
K_{eff} = 1.0	12,960

*Source Range Count Rate would change from 200 cps to 197 cps

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TABLE 8.3.1-11 (Sheet 1)

120V AC VITAL INSTRUMENT POWER BOARD 1-I
(Battery Board I)

BKR NO	LOAD	SAFETY RELATED	CONNECTED LOAD-VA
1	SSPS (A) Ch I Input Relays (Pnl 1-R-48)	Yes	1,080
2	SSPS (B) Ch I Input Relays (Pnl 1-R-49)	Yes	600
3	NIS Instr Power Ch I (JB 3398)	Yes	~~239~~ 163
4	NIS Control Power Ch I (Pnl 1-M-13)	Yes	~~440~~ 457
5	Process Protection Set I (Pnl 1-R-1)	Yes	782
6	UHI Accumulator Ch I Isolation Valve (Pnl 1-M-23A)	Yes	62
7	Cnmt. Bldg. Rad. Monitor (1-RE-90-106 & 0-RE-90-133)	Yes	949
8	Instrumentation Bus A (Pnl 0-M-27B)	Yes	50
9	SSPS Aux Relays (Pnl 1-R-73)	Yes	110
10	Reactor Building Isolation Valve (JB2670)	Yes	109
11	Aux Compressor A Aux Bldg Isol Valve (FCV-32-82)	Yes	52
12	Radiation Rate Meters (PNL 0-M-12)	Yes	1,406
13	Radiation Monitors (0-RE-90-125)	Yes	360
14	Instruments (125V Vital Battery Board I)	Yes	20
15	Incore TC Monitoring (PNL 1-R-59)	Yes	308
16	Chlorine Detector (Pnl 0-L-450)	Yes	24
17	PASF Solenoid Valves (PNL 1-L-572/C)	Yes	619
18	Aux Relays PCO-65-81 & PCO-65-86	Yes	84
19	Toilet & Locker & Spread Room Isol Dampers	Yes	168
20	BOP Process Instr Control Rack	Yes	488
21	Aux Bldg Instr Bus A (PNP 1-L-26)	Yes	150
22	Aux Bldg Stm Isol Vlv FCV-12-82 (Pnl 1-M-9)	Yes	62
23	Containment Purge Air Exhaust Rad Monitor	Yes	360
24	Reac. Vessel Hd. Vent Throttle Valve FCV-68-397	Yes	60
25	NSSS Aux Relay Rack A Bus (Pnl 1-R-54)	Yes	462
26	Sep & Aux Relays (Pnl 1-R-73)	Yes	532
27	A Relay Bus (Pnl 1-L-11A)	Yes	266
28	A Instrument Bus (Pnl 1-L-11A)	Yes	140
29	RVLIS (Pnl 1-R-148)	Yes	748
30	Aux Dryer Train A	Yes	981
31	Sep & Aux Relays (Pnl 1-R-74)	Yes	238
32	Borid Acid Tank A Htr A-A Control (1-L-303)	Yes	31
33	Aux Bldg Gas Treat Fan A-A Mod Dmpr (0-L-429)	Yes	142
34	Boric Acid Tank C Hrt A-A Control (0-L-306)	Yes	31
35	Radiation Monitor (0-RE-90-205)	Yes	360
36	RCP1 UV & UF Relays	Yes	30
37	Process Control Group I (Pnl 1-R-14)	No	715
38	Instrument Bus 1 (Pnl 0-M-27B)	No	56
39	Plugmold Instrument Bus 1 (Pnl 1-M-5)	No	225
40	Plugmold Instrument Bus 1 (Pnl 1-M-6)	No	280
41	Instrument Bus 1 (Pnl 1-M-4)	No	60

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TABLE 8.3.1-11 (Sheet 2)
(Continued)

120V AC VITAL INSTRUMENT POWER BOARD 1-I
(Battery Board I)

BKR NO	LOAD	SAFETY RELATED	CONNECTED LOAD-VA
42	Fire Pump 2A-A Sep Relays	No	84
43	Aux Bldg Gen Exhaust Fan 1A Flow Control (0-L-426)	No	49
44	~~******* Spare *******~~ NIS Insts (XM-92-5001A and 5001B) Ch I	Yes	262
45	UHI Instrument Bus 1	Yes	32
46	PR-30-310 (Installation on Hold)	Yes	30
47	5th Vital Battery Instrumentation	Yes	26
48	******* Spare *******	-	-
	TOTAL		14,100

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NOTES

1. Each inverter is capable of supplying 15 kva continuously.

2. No automatic load stripping or load sequencing is employed.

3. Power Boards 1-I, 1-II, 1-III, and 1-IV also supply common plant loads and are more heavily loaded than respective unit 2 boards.

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TABLE 8.3.1-12 (Sheet 1)

120V AC VITAL INSTRUMENT POWER BOARD 1-II
(Battery Board II)

BKR NO	LOAD	SAFETY RELATED	CONNECTED LOAD-VA
1	SSPS (A) Ch II Relays (Pnl 1-R-46)	Yes	600
2	SSPS (B) Ch II Input Relays (Pnl 1-R-51)	Yes	1,080
3	NIS Instr Power Ch II (JB 3399)	Yes	~~239~~ 340
4	NIS Control Power Ch II (Pnl 1-M-13)	Yes	~~440~~ 457
5	Process Protection Set II (Pnl 1-R-5)	Yes	1,111
6	UHI Accumulator Ch II Isolation Valve (Pnl 1-M-23B)	Yes	62
7	ERCW & Containment Rad. Monitor (Pnl 1-RE-90-112)	Yes	949
8	Instrumentation Bus B (Pnl 0-M-27B)	Yes	30
9	SSPS Aux Relays (Pnl 1-R-78)	Yes	103
10	Reactor Building Isolation Valve (FCV-32-102A)	Yes	62
11	Aux Compressor B Aux Bldg Isol Valve (FCV-32-85)	Yes	48
12	Radiation Rate Meters (PNL 0-M-12)	Yes	1,054
13	Radiation Monitors (0-RE-90-126)	Yes	360
14	Instruments (125V Vital Battery Board II)	Yes	26
15	Incore TC Monitoring (PNL 1-R-60)	Yes	308
16	Chlorine Detector (Pnl 0-L-451)	Yes	24
17	PASF Solenoid Valves (PNL 1-M-10)	Yes	619
18	Aux Relays PCO-65-83 & PCO-65-87	Yes	84
19	Toilet & Locker & Spread Room Isol Dampers (Pnl 1-R-78)	Yes	168
20	BOP Process Instr Control Rack (Pnl 1-R-131)	Yes	203
21	Aux Bldg Instr Bus B (PNP 1-L-26)	Yes	150
22	Aux Boiler Stm Isol Vlv FCV-12-79 (Pnl 1-M-9)	Yes	62
23	Containment Purge Air Exhaust Rad Monitor (1-RE-90-131)	Yes	360
24	Reac. Vessel Hd. Vent Throttle Valve FCV-68-396	Yes	60
25	NSSS Aux Relay Rack B Bus (Pnl 1-R-55)	Yes	308
26	Aux Rly Rack Sep & Aux Relays (Pnl 1-R-78)	Yes	546
27	Aux Cont Pnl B Relay Bus (1-L-11B)	Yes	168
28	Aux Cont Pnl B Instrument Bus (1-L-11B)	Yes	184
29	RVLIS (Pnl 1-R-148)	Yes	768
30	Aux Dryer Train B	Yes	981
31	Aux Rly Rack Sep & Aux Relays (Pnl 1-R-77)	Yes	210
32	Borid Acid Tank A Htr B-B Control (Pnl 1-L-304)	Yes	31
33	Aux Bldg Gas Treat Fan B-B Mod Dmpr (0-L-428)	Yes	142
34	Boric Acid Tank C Hrt B-B Control (Pnl 0-L-305)	Yes	31
35	Radiation Monitor (0-RE-90-206)	Yes	360
36	RCP2 UV & UF Relays	Yes	30
37	Process Control Group 2 (Pnl 1-R-17)	No	743
38	Instrument Bus 2 (Pnl 0-M-27B)	No	38
39	Plugmold Instrument Bus 2 (Pnl 1-M-3)	No	600
40	Plugmold Instrument Bus 2 (Pnl 1-M-6)	No	166
41	Accoustic Flow Monitor (Pnl 1-M-27A)	No	117

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TABLE 8.3.1-13 (Sheet 1)

120V AC VITAL INSTRUMENT POWER BOARD 1-III
(Battery Board III)

BKR NO	LOAD	SAFETY RELATED	CONNECTED LOAD-VA
1	SSPS (A) Ch III Input Relays (Pnl 1-R-46)	Yes	600
2	SSPS (B) Ch III Input Relays (Pnl 1-R-49)	Yes	600
3	NIS Instr Power Ch III (JB 3400)	Yes	88
4	NIS Control Power Ch III (Pnl 1-M-13)	Yes	240
5	Process Protection Set III (Pnl 1-R-9)	Yes	456
6	UHI Accumulator Ch III Isolation Valve (Pnl 1-M-23A)	Yes	65
7	RCP 3 UV & UF Relays	Yes	30
8	Aux Feed Turb Controller (Pnl 1-L-381)	Yes	222
9	Turb Dr Aux FW PMP St Gen (Pnl 1-L-361)	Yes	62
10	Turb Dr Aux FW PMP St Gen (Pnl 1-L-11A)	Yes	80
11	******* Spare *******	-	-
12	******* Spare *******	-	-
13	******* Spare *******	-	-
14	Instrument Bus & Xfmr Pwr (Pnl 1-M-3)	No	30
15	Aux Cont Pnl A Inst Bus (PNL 1-L-11A)	No	475
16	Process Cont Group 3 (Pnl 1-R-20)	No	282
17	BOP Process Instr Control Rack (Pnl 1-R-126)	No	3,170
18	Control Room Doors Security Lock	No	100
19	Emergency Gas Treat Filter Train A (Pnl 0-L-25)	No	20
20	BOP Process Instr Control Rack (Pnl 1-R-128)	No	845
21	******* Spare *******	-	-
22	Aux Bldg Inst A Bus 1 (Pnl 1-L-57)	No	20
23	Aux Relay Rack A Bus (Pnl 1-R-76)	No	364
24	Aux Relay Rack C Bus (Pnl 1-R-76)	No	196
25	NSSS Aux Relay Rack A Bus (Pnl 1-R-58)	No	84
26	Aux Control Panel A Bus (Pnl 1-L-10)	No	168
27	Aux Relay Rack A Bus (Pnl 1-R-75)	No	434
28	SSPS Control Room Demod (Pnl 1-M-22)	No	480
29	Aux Control Panel C Relay Bus (Pnl 1-L-10)	No	42
30	Aux Control Panel A Instr Bus (Pnl 1-L-10)	No	~~184~~ 332
31	Control Air Hdr A Moisture Alm (JB281)	No	13
32	Aux Relay Rack A Bus (Pnl 1-R-72)	No	182
33	******* Spare *******	-	-
34	Post Accident Monitoring (Pnl 1-M-5)	No	48
35	******* Spare *******	-	-
36	LOCA H2 Cntmnt Flow Monitor (Pnl 1-M-10)	No	106
37	CO2 Fire Protection Computer Room	No	-
38	CO2 Fire Protect Diesel Gen & Lube Oil Rm	No	-
39	Feed To Bkrs 37 & 38	No	1,500
40	Loose Parts Monitor Equipment Panel (Pnl 0-R-139)	No	240
41	Reactor Vessel Level Instr System	No	110

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TABLE 8.3.1-15 (Sheet 1)

120V AC VITAL INSTRUMENT POWER BOARD 2-I
(Battery Board I)

BKR NO	LOAD	SAFETY RELATED	CONNECTED LOAD-VA
1	SSPS (A) Ch I Input Relays (Pnl 2-R-46)	Yes	600
2	SSPS (B) Ch I Input Relays (Pnl 2-R-49)	Yes	600
3	NIS Instr Power Ch I	Yes	~~220~~ 340
4	NIS Control Power Ch I (Pnl 2-M-13)	Yes	~~440~~ 457
5	Process Protection Set I (Pnl 2-R-1)	Yes	1,076
6	UHI Accumulator Ch I Isolation Valve (Pnl 2-M-283)	Yes	62
7	RCP 1UV & MF Relays	Yes	30
8	Aux Feed Pump Turb Flow Cont	Yes	222
9	Turb Dr Aux FW PMP St 3&4 Gen (Pnl 2-L-381)	Yes	75
10	Turb Dr Aux FW PMP St 3&4 Gen (Pnl 2-L-11A)	Yes	80
11	RVLIS (Pnl 2-R-148)	Yes	745
12	Incore TC Monitoring (Pnl 2-R-60)	Yes	308
13	******* Spare *******	-	-
14	Instr Bus (Pnl 2-M-4)	No	60
15	Plugmold Inst Bus 1 (PNL 2-M-5)	No	194
16	Plugmold Inst Bus 1 (Pnl 2-M-6)	No	280
17	Process Cont. Group (Pnl 2-R-14)	No	749
18	BOP Process Instr Cont Rack (Pnl 2-R-126)	No	3,100
19	Aux Cont Pnl A Instr Bus (Pnl 2-L-11A)	No	455
20	BOP Process Instr Cont Rack (Pnl 2-R-128)	No	600
21	******* Spare *******	-	-
22	Aux Bldg Inst A Bus (Pnl 2-L-57)	No	50
23	Aux Relay Rack A Bus (Pnl 2-R-76)	No	182
24	Aux Relay Rack C Bus (Pnl 2-R-76)	No	140
25	NSSS Aux Relay Rack A Bus (Pnl 2-R-58)	No	84
26	Aux Cont Pnl A Relay Bus (Pnl 2-L-10)	No	154
27	Aux Relay Rack A Bus (Pnl 2-R-75)	No	266
28	SSPS Cont Rm Demod (Pnl 2-M-22)	No	480
29	Aux Cont Pnl A Inst Bus (Pnl 2-L-10)	No	210
30	******* Spare *******	-	-
31	******* Spare *******	-	-
32	Aux Relay Rack A Bus (Pnl 2-R-32)	No	56
33	******* Spare *******	-	-
34	Post Accident Monitoring 1 (Pnl 2-M-5)	No	48
35	******* Spare *******	-	-
36	LOCA H2 Cntmnt Flow Monitor (Pnl 2-M-10)	No	106
37	******* Spare *******	-	-
38	******* Spare *******	-	-
39	******* Spare *******	-	-
40	******* Spare *******	-	-
41	******* Spare *******	-	-

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TABLE 8.3.1-15 (Sheet 2)
(Continued)

120V AC VITAL INSTRUMENT POWER BOARD ~~1-IV~~ 2-I
(Battery Board I)

BKR NO.	LOAD	SAFETY RELATED	CONNECTED LOAD-VA
42	******* Spare *******	-	-
43	******* Spare *******	-	-
44	******* Spare *******	-	-
45	UHI Instrument Bus 1 (Pnl 2-M-23A)	No	32
46	******* Spare *******	-	-
47	******* Spare *******	-	-
48	******* Spare *******	-	-
	TOTAL		11,723

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NOTES

1. Each inverter is capable of supplying 15 kva continuously.
2. No automatic load stripping or load sequencing is employed.
3. Power Boards 1-I, 1-II, 1-III, and 1-IV also supply common plant loads and are more heavily loaded than respective unit 2 boards.

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TABLE 8.3.1-16 (Sheet 1)

120V AC VITAL INSTRUMENT POWER BOARD 2-II
(Battery Board II)

BKR NO	LOAD	SAFETY RELATED	CONNECTED LOAD-VA
1	SSPS (A) Ch II Input Relays (Pnl 2-R-46)	Yes	600
2	SSPS (B) Ch II Input Relays (Pnl 2-R-49)	Yes	600
3	NIS Instr Power Ch II (JB 3403)	Yes	~~239~~ 163
4	NIS Control Power Ch II (Pnl 2-M-13)	Yes	~~440~~ 457
5	Process Protection Set II (Pnl 2-R-5)	Yes	1,111
6	UHI Accumulator Ch II Isolation Valve (Pnl 2-M-23B)	Yes	62
7	RCP 2UV & UF Relays	Yes	30
8	Aux Feed Pump Turb Flow Cont (2-L-381)	Yes	222
9	Turb Dr Aux FW PMP St Gen 1&2 LIC-3-173,174	Yes	60
10	Turb Dr Aux FW PMP St Gen 1&2 Instr Loop	Yes	244
11	RVLIS (Pnl 2-R-148)	Yes	745
12	Incore TC Monitoring (Pnl 2-R-60)	Yes	308
13	******* Spare *******	-	-
14	Process Cont Group 2 (Pnl 2-R-17)	No	743
15	Plugmold Inst Bus 2 (PNL 2-M-3)	No	509
16	Aux Cont Pnl B Inst Bus (Pnl 2-L-11B)	No	535
17	Plugmold Instr Bus 2 (Pnl 2-M-6)	No	136
18	BOP Process Instr Cont Rack (Pnl 2-R-122)	No	629
19	******* Spare *******	-	-
20	BOP Process Instr Cont Rack (Pnl 2-R-130)	No	454
21	******* Spare *******	-	-
22	Aux Bldg Inst B Bus (Pnl 2-L-299)	No	40
23	Aux Relay Rack B Bus (Pnl 2-R-76)	No	168
24	NSSS Aux Relay Rack C Bus (Pnl 2-R-58)	No	322
25	NSSS Aux Relay Rack B Bus (Pnl 2-R-58)	No	84
26	Aux Relay Rack B Bus (Pnl 2-R-75)	No	294
27	Aux Relay Rack C Bus (Pnl 2-R-75)	No	168
28	Aux Cont Pnl B Relay Bus (Pnl 2-L-10)	No	84
29	Aux Cont Pnl C Relay Bus (Pnl 2-L-10)	No	28
30	Aux Cont Pnl B Instr Bus (Pnl 2-L-10)	No	~~150~~ 321
31	Emerg VHF Radio (Pnl G) -- Equipment Removed --	-	-
32	Aux Relay Rack B Bus (Pnl 2-R-72)	No	112
33	Aux Relay Rack C Bus (Pnl 2-R-72)	No	56
34	Post Accident Monitoring 2 (Pnl 2-M-4)	No	64
35	******* Spare *******	-	-
36	LOCA H2 Cntmnt Flow Monitor (Pnl 2-M-10)	No	106
37	RVLIS (Pnl 2-R-148)	Yes	729
38	******* Spare *******	-	-
39	******* Spare *******	-	-
40	******* Spare *******	-	-
41	******* Spare *******	-	-

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TABLE 8.3.1-16 (Sheet 2)
(Continued)

120V AC VITAL INSTRUMENT POWER BOARD 2-II
(Battery Board II)

BKR NO	LOAD	SAFETY RELATED	CONNECTED LOAD-VA
42	******* Spare *******	-	-
43	******* Spare *******	-	-
44	Inplant VHF Radio Repeater F1	No	1,068
45	UHI Instrument Bus 1 (Pnl 2-M-23B)	No	32
46	On Site Paging Radio	No	1,068
47	******* Spare *******	-	-
48	~~******* Spare *******~~ NIS INSTS (XM-92-5002A and 5002B) Ch II	Yes	262
	TOTAL		12,240

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NOTES

1. Each inverter is capable of supplying 15 kva continuously.

2. No automatic load stripping or load sequencing is employed.

3. Power Boards 1-I, 1-II, 1-III, and 1-IV also supply common plant loads and are more heavily loaded than respective unit 2 boards.

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ENCLOSURE 3

PROPOSED TECHNICAL SPECIFICATION CHANGE

SEQUOYAH NUCLEAR PLANT UNITS 1 AND 2

DOCKET NOS. 50-327 AND 50-328

(TVA-SQN-TS-88-42)

DETERMINATION OF NO SIGNIFICANT HAZARDS CONSIDERATIONS

ENCLOSURE 3

Significant Hazards Evaluation

TVA has evaluated the proposed TS change and has determined that it does not represent a significant hazards consideration based on criteria established in 10 CFR 50.92(c). Operation of SQN in accordance with the proposed amendment will not:

(1) Involve a significant increase in the probability or consequence of an accident previously evaluated. Two administrative changes are proposed to support the installation of the new Gamma Metrics SR and IR detector assemblies. The first involves a revision to the notation contained in TS Table 3.3-1 regarding the high-voltage deenergization that will no longer occur for the new SR detectors. The second involves a change in engineering units for the P-6 setpoint that results from the difference in output signals from the IR detectors. The new SR/IR detectors are Class-1E equipment that is seismically and environmentally qualified and compatible with the present design requirements. Because the new hardware is compatible with the present design requirements and the proposed TS changes are administrative in nature, the proposed amendment will not involve a significant increase in the probability or consequences of an accident previously evaluated.

(2) Create the possibility of a new or different kind of accident from any previously analyzed. Two administrative changes are proposed to support the installation of the Gamma Metrics SR and IR detector assemblies. The first involves a revision to the notation contained in TS Table 3.3-1 that is no longer applicable to the design of the new SR detectors. The second involves a change in engineering units for the P-6 setpoint that results from the difference in output signals from the IR detectors. The new SR/IR detectors are Class-1E equipment that is seismically and environmentally qualified and compatible with the present design requirements. Because the new hardware is compatible with the present design requirements and the proposed TS changes are administrative in nature, the proposed amendment will not create the possibility of a new or different kind of accident from any previously analyzed.

(3) Involve a significant reduction in a margin of safety. Two administrative changes are proposed to support the installation of the Gamma Metrics SR and IR detector assemblies. The first involves a revision to the notation contained in TS Table 3.3-1 that is no longer applicable to the design of the new SR detectors. The second involves a change in engineering units for the P-6 setpoint that results from the difference in output signals from the IR detectors. The new SR/IR detectors are Class-1E equipment that is seismically and environmentally qualified and compatible with the present design requirements. Because the new hardware is compatible with the present design requirements and the proposed TS changes are administrative in nature, the proposed amendment will not involve a significant reduction in the margin of safety.