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

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<u>Unit 2</u>

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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*	≥ 1x10^{-5}% of RATED THERMAL POWER ~~≥ 1 x 10^{-10} amps~~	≥ 6x10^{-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

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* The trip setpoi [illegible] 10^{-10} amps and the allowable value is 6x10^{-11} amps until replacement of the interm [illegible] monitor during the U1C4 refueling outage.

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

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 *	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						R45
Four Loop Operation	4	2	3	1, 2	$6^{\#}$	R45
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	$7^{\#}$	

The voltage to the detector may be de-energized above the P-6 setpoint until replacement of the source range monitor during the U1C4 refueling outage.

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.

[##] ~~High voltage to detector may be 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	$\geq$ 18% of narrow range instrument span-each steam generator	$\geq$ 17% of narrow range instrument span-each steam generator
14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	$\leq$ 40% of full steam flow at RATED THERMAL POWER coincident with steam generator water level $\geq$ 25% of narrow range instrument span--each steam generator	$\leq$ 42.5% of full steam flow at RATED THERMAL POWER coincident with steam generator water level $\geq$ 24% of narrow range instrument span--each steam generator
15. Undervoltage-Reactor Coolant Pumps	$\geq$ 5022 volts-each bus	$\geq$ 4739 volts-each bus
16. Underfrequency-Reactor Coolant Pumps	$\geq$ 56 Hz - each bus	$\geq$ 55.9 Hz - each bus
17. Turbine Trip		
A. Low Trip System Pressure	$\geq$ 45 psig	$\geq$ 43 psig
B. Turbine Stop Valve Closure	$\geq$ 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	$\geq 1 \times 10^{-5}$% of RATED THERMAL POWER ~~$\geq 1 \times 10^{-10}$ amps~~	$\geq 6 \times 10^{-6}$% of RATED THERMAL POWER ~~$\geq 6 \times 10^{-11}$ amps~~
20. Power Range Neutron Flux (not P-10) Input to Low Power Reactor Trips Block P-7	< 10% of RATED THERMAL POWER	$\leq$ 11% of RATED THERMAL POWER

R7

R76

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

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 *	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						R33
Four Loop Operation	4	2	3	1, 2	6#	R33
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#	

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 may be de-energized 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.

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.

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 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 applicability table 3.3-1 is being revised to delete an unnecessary note involving high-voltage deenergization. The new IR monitor uses a signal that is in units of relative power. Consequently, the trip setpoint and allowable value are being changed in table 2.2-1. The bases to section 2.2 are also being revised to delete references to IR detector current signals that are proportional to power levels. The changes to unit 1 also have appropriate footnotes added to indicate that the changes become effective for unit 1 after installation of the new detectors during the unit 1 cycle 4 refueling outage. The unit 2 detectors will be installed during the unit 2 cycle 3 refueling outage, and the change will be effective at the time of startup following the outage.

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 detector design is compatible with the current system; however, it includes two improvements over the present detector design. First, the electronic equipment automatically adjusts the high flux at shutdown alarm. Currently, this function is performed manually as described in the Final Safety Analysis Report, section 15.2.4.2. Second, the new SR 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. However, the SR/IR detector assemblies remain energized during the full range of power operation. As a result of this feature, the table notation in table 3.3-1 regarding high-voltage deenergization can be deleted because it is not applicable to the new design.

The new IR detector design is compatible with the current system except that the output signal is in units of relative power rather than amperes (amps). The P-6 setpoint and allowable value listed in table 2.2-1 are currently listed in units of amps. 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 startup 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 calibration. 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.

Footnotes have been added to tables 2.2-1 and 3.3-1 of the unit 1 technical specifications that indicate that the proposed changes do not become effective until installation of the SR/IR detector assemblies during the unit 1 cycle 4 refueling outage scheduled for mid-1990. This method of handling the unit 1 change will allow a single review of the issue and avoid separate technical specification change requests for each unit.

In summary, three administrative changes are proposed to support the installation of the Gamma Metrics SR/IR assembly. The first involves the deletion of a table note that is not 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 third involves the addition of certain footnotes to enable the review and approval of the unit 1 changes to proceed independently of the unit 1 installation schedule.

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 Part 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

(B25 881117 808)

TVA 10657 (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.
APPLICABLE DESIGN DOCUMENT(S) SQN - SC - V - 27 . 8 22	
SAR SECTION(S) TABLE 14.1-2	UNID SYSTEM(S) 92

Rev	(for RIMS' use)	RIMS accession number
R0	881117B0001	825 881110 803
R 1	INFO	825 881117 808
R _	ONLY	
R _		

Revision 0	R1	R2	R3
ECN No. (or indicate Not Applicable) L6186	N/A ~~L6186~~ 11/17/88		
Prepared W.A. MUELLER Willa A. Mueller	W.A. Mueller W.A. MUELLER		
Checked Chandran Balk CHANDRAN BALK	Chandran Balk CHANDRAN BALK		
Reviewed F.M. Riva F.M. Riva	C. Giegitis C. GIEGITIS		
Approved L.W. Sebley for 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 L.W. Sebley for 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

Title: INTERMEDIATE RANGE NEUTRON FLUX P-6 SETPOINT

REVISION LOG

1,2-XE-92-1

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

TVA 10534 (EN DES 4-78)

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 9 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) Steve Stark
Wednesday
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. Beddington
Design Verifier
(Independent Reviewer)

11-16-88
Date

NFP-3.1
Attachment 5
Page 1 of 1

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

Blai
Design Verifier
(Independent Reviewer)

11/16/55
Date

CALCULATION DESIGN VERIFICATION (INDEPENDENT REVIEW) FORM

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

Revision: 0

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

Design Verifier
(Independent Reviewer)

Date: [illegible]

GENERAL COMPUTATION SHEET

DISCIPLINE

United Engineers & Constructors
A Raytheon Company

NAME OF COMPANY TVA - SQN UNITS 1 & 2

SUBJECT INTERMEDIATE RANGE NEUTRON FLUX P-6 SET POINT

CALC IDENT		REV	COMP BY	CHECKED BY
PRELIM		0	DATE 11/4/88	Cb DATE [illegible]
FINAL	1,2-XE-92-1			
VOID				
SHEET 1 OF 14		1	DATE 11/16/88	Cb DATE 11/16/88
JO 8806.129				

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	FSAR COMPLIANCE REVIEW	4
1.0	PURPOSE	5
2.0	TECHNICAL REQUIREMENTS	5
3.0	SOURCES OF DESIGN INPUT INFORMATION	6
4.0	DESIGN INPUT DATA	7
5.0	COMPUTATIONS / ANALYSIS	8 THRU 10
6.0	SUMMARY OF RESULTS	10
7.0	CONCLUSIONS	5
8.0	ATTACHMENTS	11 THRU 14 E |R1

Form 3604 Rev 12 '6 83

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 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 20 14 LATEST REVISION OF CALC. SET 1 0 DATE NOTED 11/9/88

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

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

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

Form 5605 Rev 12/16/83

United Engineers & Constructors
A Raytheon Company

CALCULATION REVISION CONTROL SHEET

SHEET 2A OF 14

PROJECT TITLE SQN DISCIPLINE DNE - EEB

CALCULATION SET NO.

PRELIMINARY	
FINAL	✓
VOID	

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 William C. Mueller (W.A. MUELLER) DATE 11/16/88
Signature

REVISION AUTHORIZED BY F.M. Riva 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 SHEET NOS. REVISED: 1, 4, 8, 9, 10 ADDED: (2) R.V. REHS, 2A, 14A THRU 14E
Quantity List Sheets Revised Added Deleted

REVISION FINISHED BY William C. Mueller (W.A. MUELLER) DATE 11/16/88
Signature

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

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

United Engineers & Constructors
A Raytheon Company

CALCULATION SUMMARY & REFERENCE SHEET

SHEET 3 OF 14

PROJECT TITLE SQN
DISCIPLINE ONE - EEB

CALCULATION SET NO 1,2-NE-KG-1	
PRELIMINARY	
FINAL	✓
VOID	
SHEET 3 OF 14	
10 3806.129	

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

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

(DISCIPLINE)

GENERAL COMPUTATION SHEET

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		0	DATE 11/4/88	DATE [illegible]
FINAL	1,2-XG-02-1			
VOID				
SHEET	4 OF 14	1	DATE 11/16/88	CMB 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		0	Cr	[illegible]
FINAL	1,2-XG-92-1		DATE [illegible]	DATE [illegible]
VOID				
SHEET	5 OF 14		DATE	DATE
JO	8806.129			

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)

GENERAL COMPUTATION SHEET

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		0		
FINAL	1,2-XE-92-1 ✓		DATE 11/9/88	DATE 11/9/88
VOID				
SHEET 6 OF 14				
JO 8806129			DATE	DATE

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)

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	CHKD BY
PRELIM		0	[illegible]	[illegible]
FINAL	SQ-XE-92-1		DATE 11/9/88	DATE [illegible]
VOID				
SHEET 7 OF 14				
JO 8806-129			DATE	DATE

4.0 DATA TAKEN FROM SU-8.5.1 (2SF R2)

ACTUAL VALUE (AMP.) AVERAGE VALUE (AMPS)

CHAN N35 3.6×10^{-5}
CHAN N36 3.4×10^{-5} @ 10% POWER 3.5×10^{-5}

CHAN N35 1.09×10^{-4}
CHAN N36 1.15×10^{-4} @ 29% POWER 1.12×10^{-4}

CHAN N35 1.66×10^{-4}
CHAN N36 1.73×10^{-4} @ 50% POWER 1.695×10^{-4}

CHAN N35 2.549×10^{-4}
CHAN N36 2.65×10^{-4} @ 75% POWER 2.6×10^{-4}

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		0		
FINAL	1,2-XE-42-1 ✓		DATE 11/9/88	DATE 11/9/88
VOID				
SHEET 8 OF 14		1		CO
0 8806-129			DATE 11/16/88	DATE 11/18/88

SECTION 5.0

POWER PERMISSIVE P-6 SETPOINT

10% POWER

AVERAGE = 3.5 X 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~~

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. IDENT. NO.		REV	COMP BY	CHK'D BY
PRELIM				
FINAL	62-XE-92-1	0	DATE 11/7/88	DATE 11/10/88
VOID				
SHEET 9 OF 14		1	DATE 11/16/88	Ch DATE 11/16/88
JO 8806.129				

SECTION 5.0 (CONT'D)

ALLOWABLE VALUE

INTERPOLATION MAY BE USED BECAUSE IR CURRENT AND POWER ARE LINEAR ~~LOGRITHMIC~~ 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:

GIVEN 1×10^{-5} % RTP = 10^{-10} 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

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		0	DATE 11/9/88	DATE 11/10/88
FINAL	1,2-XE-92-1			
VOID				
SHEET 10 OF 14		1	DATE 11/16/88	DATE 11/16/88
JO 8306.129				

SECTION 6.0 (CONT'D)

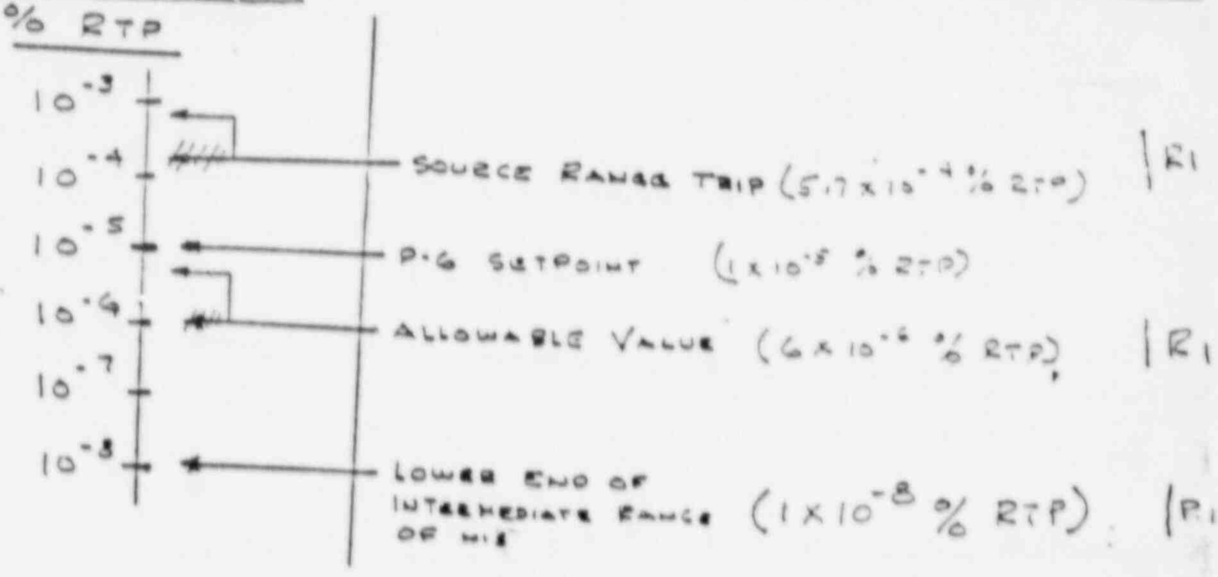
TO BE ABLE TO SHOW THESE BLOCK & TRIP SETPOINTS ON THE SAME SCALE FOR COMPARASON PURPOSES WE CALCULATE THE ACTUAL SOURCE RANGE TRIP IN % RTP BY PERFORMING THE FOLLOWING: | R

SOURCE RANGE TRIP CALCULATION

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

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

6.0 SUMMARY OF RESULTS



ATTACHMENT 1

1,2-KE-HZ-1

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

: 10/3/80 Time 0510 c Unit 1

MEDIATE RANGE

Channel N35

Main Control Board	2.9 x10-5 amps
NI Drawer	3.4 x10-5 amps

Channel N36

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

R RANGE

Channel N41

Main Control Board	11. %
NI Drawer	9.6 %

Channel N42

Main Control Board	9.5 %
NI Drawer	9.3 %

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. F. Sup Ed , 10/3/80

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

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

Date 10/31/80 Power 29% Unit 1

10/31/80

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}	µA	100
4	N-41 Det. B	Back of meter	.1472	.1415	.1421	.144	$10^{-3}V$ =	1.44×10^{-4}	µA	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}	µA	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	~~2.492~~ 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

DVM # 418927 Calibration due date 1/31/81

REMARKS: Det. B scale
N41A: 0.5 µA B: 0.5 µA
N42A: 0.5 µA B: 0.5 µA
N43A: 0.5 µA B: 0.5 µA
N44A: 0.5 µA B: 0.5 µA

Data By Rush Sammons 10/31/80

Checked By Randall 1/1/81

PAGE 12 OF 14

[illegible] Units 1 & 2
Data Sheet 2
Page 1 of 1

Date 11/24/80 Power 75% Unit 1

Item No.	Parameter	Test Point	Volts at [illegible] (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	$10^{(.8V-11)}$ =	2.5491 10	^{4} amps
2	N-36	TP3 to TP4	9.281	9.276	9.280	9.279	$10^{(.8V-11)}$ =	2.6110	^{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	6.303	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

REMARKS:

DVM # 486348 Calibration due date 12/27/80

Data By R.W. Fatenberry 11/24/80

Checked By [illegible] 11/[illegible]

PAGE 14 OF 14

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 [illegible]
[illegible]
DATE [illegible] 6 1984
T.V.A. NUCLEAR ENGR BRANCH
By: CHIEF NUCLEAR ENGINEER
Appr'd by Ltr. # 7601

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

~~APPROVED WITH CORRECTION
AS NOTED
[illegible]
DATE [illegible] 1980
T.V.A. NUCLEAR [illegible]
By: CHIEF NUCLEAR [illegible]
Appr'd by Ltr. # 7211~~

PROJECT SQN DATE MAR 30 1984
CONTRACT 8PCC0-91984 FILE N2M-2-X
DRAWING NO. PLS
SHEET — REV 9 UNIT 1 & 2

PAGE 14A THRU 14E

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. <u>Permissive and Interlock Circuits</u>	is equival to $1X10^{-5}$ A.
A. P-6 (allows manual block of source range high level reactor trip) (NC-35D, NC-36D)	10^{-10} amperes
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 14C THRU 14E

Calc# 1,2-XB-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)	
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

is equivalent to 5.7×10^{-4} % FP

PAGE 14C THRU 14E

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

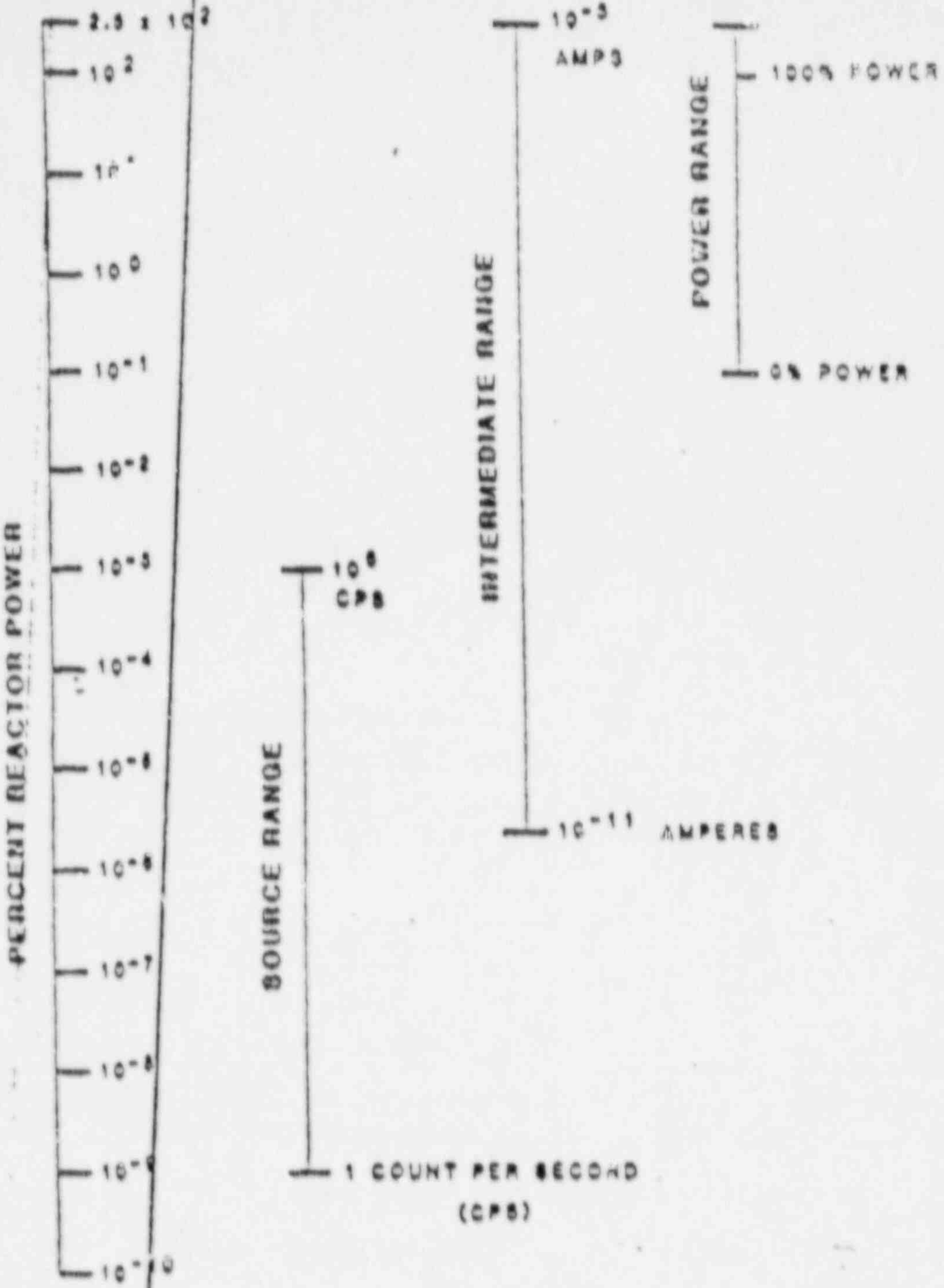


Figure 10.1-1 Neutron Detectors and Ranges of Operation

PAGE 14 D - THRU 14 E

Ⓦ FUNCTIONAL REQUIREMENTS

CALC # 1,2-XE-9

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 this span

1×10^{-5}
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.

Page 14C Thru 14E

QA Record

NEP-6.6
Attachment 3
Page 1 of ~~5~~ ~~33~~ ~~47~~
~~37~~
39 11/17/8

SAFETY EVALUATION FORM

1. To
Sequoyah Nuclear Plant, Daisy, TN

2. From
E16 SNP

3. USQ?
☐ Yes
☑ No

Sheet 1

4. Safety Evaluation Number
ECN L6186

9. RIMS Accession Number

R 0	SQP 850117 503
R 1	B25 881114 53
R 2	B25 881117 57
R 3	
R 4	
R 5	

Rev No.	Tot pp	5. Prepared	6. Reviewed	7. Approved	Date 8. Appd
0	3	R.A. Edlund	RR Campbell	LH Chin/VAB	1/14/85
1	37 ~~33~~	T Fleischer	H Wilder 11/17/88		11-14-88
2	39	T Fleischer			11-17-88
3					
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. 20-14

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.

(continued on sheet 3)

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

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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 QAM 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 "Seperation of Electrical Equipment and Wiring"
11. Tech Specs
12. Design Criteria SQN-DC-V-26.2 "Environmental Qualification to 10CFR 50.49
13. Calculation 1,2-XE-92-1 (EEB Calculation)
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

FSAR chapter 15.0

19. Plant procedure PHYSI-13 "Fire"

ADDITIONAL INFORMATION

17. DESCRIPTION OF PROPOSED ACTIVITY (continued)

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

1. Replace the non-1E Westinghouse source and intermediate range neutron detectors with class 1E GammaMetrics source and intermediate range detectors.

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).

3. Replace the non-1E Westinghouse pre-amplifier to a 1E GammaMetrics amplifier assembly

4. Replace the cable from Unit 1 penetrations 43 & 48 and Unit 2 penetrations 23 & 31 to the new amplifier assemblies described in #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 to with GammaMetrics intermediate and source range drawers. The new GammaMetrics drawers will provide the required electronics to provide qualified signals for compliance with Reference #1

7. Install a shutdown margin monitor on each source range neutron monitoring channel to automatically adjust for flux decay during shutdown and identify and report flux increases that indicate a loss of reactor shutdown margin.

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ADDITIONAL INFORMATION

17. DESCRIPTION OF PROPOSED ACTIVITY (continued)

8. Route a temporary cable from the auxiliary control room L-10 to main control room panel M-4 to provide redundant neutron flux information to the main control room from the Appendix R back-up 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 back-up source range monitor and the operating source and intermediate flux monitor channel will ensure the operator has redundant neutron flux information during implementation of this ECN.
9. Replace the Appendix R Westinghouse back-up source range detector, cabling, and electronics with an optically isolated signal from the new GammaMetrics amplifier to a full range processor and display.
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.

Implementing the proposed activity will require revisions to FSAR Chapters 7.0 and 15.0 and the Tech Specs (see Justification for Question 27). This activity will be performed in the following stages:

STAGE 1 : Will consist of implementing items 1 through 8 above during the Unit 2 cycle 3 outage for Unit 2.

STAGE 2 : Will implement the entire modification for Unit 1 during the Unit 1 cycle 4 outage.

STAGE 3 : Will implement items 9 and 10 for Unit 2 du[illegible] the Unit 2 cycle 4 outage.

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ADDITIONAL INFORMATION

18. SYSTEMS, STRUCTURES, COMPONENTS AFFECTED

COMPONENT	SYSTEM AFFECTED
XE-92-1 Neutron Detector	Neutron Monitoring (NM) Post Accident Monitoring (PAM) Reactor Protection System (RPS)
XE-92-2 Neutron Detector	NM, PAM, RPS
Unit 2 Primary Containment Penetration #23 Channel II	Containment Vessel and Penetrations
Unit 2 Primary Containment Penetration #31 Channel I	Containment Vessel and Penetrations
Unit 1 Primary Containment Penetration #48 Channel II	Containment Vessel and Penetrations
Unit 1 Primary Containment Penetration #43 Channel I	Containment Vessel and Penetrations
Channel 1 signal amplifier assembly	NM, PAM, RPS
Channel 2 signal amplifier assembly	NM, PAM, RPS
Shutdown Margin Monitor located in panel M-13	NM, PAM
Source Range Drawer N31 (Channel 1) located in panel M-13	NM, PAM, RPS
Intermediate Range Drawer N32 (Channel 1) located in panel M-13	NM, PAM, RPS
Source Range Drawer N35 (Channel 2) located in panel M-13	NM, PAM, RPS
Intermediate Range Drawer N36 (Channel 2) located in panel M-13	NM, PAM, RPS
1E to specialized 1E optical isolator	NM, PAM,
Signal processor and display unit located on panel L-10	NM, PAM
Shutdown Margin Monitor	

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ADDITIONAL INFORMATION

1[illegible] [illegible]nction(s) of System(s) Affected

[illegible]ty functions of the systems affected by this ECN are described

POST ACCIDENT MONITORING (PAM)

The safety function of the post accident 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.

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ADDITIONAL INFORMATION

19. SAFETY FUNCTIONS OF SYSTEMS AFFECTED (continued)

2. Neutron Monitoring System - Source & 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. 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) 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.

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. 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 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).

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 "POST ACCIDENT MONITORING" Additional detailed safety function information is available in Reference 2.

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ADDITIONAL INFORMATION

19. SAFETY FUNCTIONS OF SYSTEMS AFFECTED (Cont'd)

2. Neutron Monitoring System - Source Range (Cont'd)

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

ADDITIONAL INFORMATION

19. SAFETY FUNCTIONS OF SYSTEMS AFFECTED (continued)

3. Primary Containment System

The Primary Containment System will be breached during the replacement of electrical containment penetrations 43 and 31. 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.

4. Reactor Protection System

The source and intermediate range neutron monitors 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:

1. Whenever necessary to prevent fuel damage for an anticipated transient (Condition II).
2. To limit core damage for infrequent faults (Condition III).
3. 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.

SAFETY EVALUATION

20. EFFECTS ON SAFETY

This modification and the Tech Spec change will not affect the safety functions of the systems listed in # 19 above or any other systems important to 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 safe 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 backup to power range trips. Also, the interlocks described in #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 back-up source range neutron monitor which presently outputs to the auxiliary control room will also output to the main control room. 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.

SAFETY EVALUATION

20. EFFECTS ON SAFETY (Continued)

3. Electrically, the source and intermediate neutron range monitors affect only PAM and RPS as discussed above. No other system receives input from any portion of the source or intermediate neutron monitoring system. Also, based on a comparison between Westinghouse and Gammametrics specifications of neutron monitoring equipment, power consumption of the new GammaMetrics units will be less. ECN L6186 will therefore have no effect electrically on any other system important to safety.

4. The upgrade of the source and intermediate range neutron monitoring system to Reg Guide 1.97 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, assuring that components or equipment of other systems important to safety will not be subjected to seismically induced missile damage.

7. The temporary cable routed from the auxiliary control room to the main control room to connect the back-up source range neutron monitor to the main control room (as described in item 2 above) and other 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)

8. The effects of pipe rupture 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.

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.

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 17, ensuring reliability of instrumentation during Chapter 15 condition III or IV faults.

11. One channel of the source and intermediate neutron monitoring system will provide a signal through an optical isolator to the auxiliary control room. 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 source and intermediate range neutron monitoring system's primary safety function.

SAFETY EVALUATION

20. EFFECTS ON SAFETY (continued)

12. One channel of the source and intermediate range monitors will be routed above Aux 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

13. The new shutdown margin monitors identified in block 17, item 7, will automatically adjust the high flux alarm setpoint during plant shutdown. Presently, this function is manually performed as described in FSAR Section 15.2.4.2. There will be one shutdown margin monitor for each neutron monitoring channel. Each will be electrically separate and fully redundant in accordance with the requirements of Reference 10. Also, verification of the setpoint and count rate can be performed by the operator any time below the P-6 interlock setpoint (described in block 17 item 2) when the shutdown margin monitor is in service to ensure the monitors are performing their function. The high flux alarm will perform its function as before, with the alarm setpoint being adjusted automatically by the shutdown margin monitors. The reliability, redundancy, and shutdown monitor tracking verification feature ensure that the high flux alarm function during shutdown will not be affected, hence the consequences of boron dilution, a condition II fault, will not be increased.

SAFETY EVALUATION

14. The new GammaMetric 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 by this modification and will not affect reactor trip interlock setpoints or alarm setpoints. R2

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SAFETY EVALUATION

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

☐ Yes ☑ 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 margin monitor described in block 17 (item 7) will automatically adjust the neutron flux alarm setpoint during shutdown. This feature will eliminate the need for an operator to adjust the setpoint during shutdown as described in FSAR section 15.2.4.2, and will reduce the probability of human error and subsequently reduce the probability of boron dilution during shutdown. R2

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.

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

☐ Yes ☑ 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 Chapter 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 margin 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. R2

SAFETY EVALUATION

22. (Continued)

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.

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

☐ Yes ☑ No

Justification:

As discussed in # 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 susceptable to seismic or environmental stresses, nor will they be susceptable to single failure. The probability of failure of the neutron monitors is not increased.

Also, as discussed in # 20 "Effects on Safety," the source and intermediate range neutron monitors will be seismically mounted, ensuring that seismically induced missiles cannot occur and contribute to the probability of malfunction of other equipment important to safety.

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SAFETY EVALUATION

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

☐ Yes ☑ 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.

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SAFETY EVALUATION

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

☐ Yes ☑ No

Justification:

The impacted instrumentation will perform the identical function following the modification as prior the modification with a higher available 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 7.2.1.1.2 b and c, respectively. PAM instrumentation will be installed as described in #20 "Effects on safety" and will not be susceptable 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.

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SAFETY EVALUATION

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

☐ Yes ☑ No

Justification:

Implementing the proposed activity is necessary to comply with the USNRC's Reg Guide 1.97. Following this modification, the affected instrumentation will perform its safety function and comply with the design requirements as described in References 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.

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

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SAFETY EVALUATION

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

☐ Yes ☑ 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. | R

2. The note in Table 3.3-1 of the Tech Spec stating "High voltage to detector may be de-energized above the P-6 (block of source range reactor trip) setpoint" is no longer required. This note was only required when the previous design source range detector was being de energized. The new design will be full range and will not be de-energized. Therefore, Tech Spec Table 3.3-1 will be revised as shown on sheets 22, 23, 24, and 25 of this safety evaluation. Detector replacement will occur in stages 1 and 2 described in block 17. | R2

3. Tech Spec Table 3.3-10 "Accident Monitoring Instrumentation" will be revised as shown on sheets 24 and 25 of the safety evaluation to show the new PAM instrumentation identified in "Systems, Structures or Components Affected." | R2
The monitoring equipment identified will be installed in stages ~~1~~ 2 and ~~2~~ 3. | R2
* 11/11/86

SAFETY EVALUATION

27. (Continued)

4. The intermediate range neutron flux P-6 permisive 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 30 and 31 of the USQD. Reference 13 justifies this scale change. The scale change will be performed during stages 1 and 2. R

5. A demonstrated loop accuracy analysis has been performed to prove that the new GammaMetrics source and intermediate neutron flux monitors will be capable of providing reactor trips below the limits specified in Table 2.2-1 (other than identified for P-6 above). The new monitors will be installed during stages 1 and 2. R

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. Intermediate source range neutron flux instrumentation will be added to aux control board L-10 and requires revision to Tech Spec 3/4.3.3.5 Tables 3.3-9 and 4.3-6. The additional instrumentation supplements the capability of permitting shutdown and maintenance of hot standby of the facility and potential subsequent cold shutdown from locations outside of the control room in the event that control room habitability is lost. The configuration after implementation of ECN L6186 remains consistent with the requirements of GDC 19 of 10CFR50. Revisions to Tables 3.3-9 and 4.3-6 are shown on sheets 32, 33, 34, & 35. This portion of ECN L6186 will performed during stages 2 and 3 R2

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.

28 & 29 Based on the safety evaluation provided above, it is concluded that no unreviewed safety question exists as a result of implementing ECN L6186.

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TABLE 3.3-1 (Continued)

TABLE NOTATION

tor trip system breakers in the closed position and the control
tem capable of rod withdrawal, and fuel in the reactor vessel.

) associated with the protective functions derived from the out
actor Coolant Loop shall be placed in the tripped condition.

s of Specification 3.0.4 are not applicable.

to detector may be de-energized above the P-6 (Block of Source
Trip) setpoint.

ACTION STATEMENTS

ith the number of channels OPERABLE one less than required by
he Minimum Channels OPERABLE requirement, restore the inoperable
hannel to OPERABLE status within 48 hours or be in HOT STANDBY
ithin the next 6 hours and/or open the reactor trip breakers.

th the number of OPERABLE channels one less than the Total
mber of Channels, STARTUP and POWER OPERATION may proceed
ovided the following conditions are satisfied:

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

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

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.

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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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 may be de-energized above the P-6 (Block of Source Range Reactor Trip) setpoint.

DELETE

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.

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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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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#	R33
9. Pressurizer Pressure-Low	4	2	3	1, 2	6#	R33
10. Pressurizer Pressure--High	4	2	3	1, 2	6#	
11. Pressurizer Water Level--High	3	2	2	1, 2	7#	

NEP 6.6 Sheet 26 | R2
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.

NEP 6.6 Sheet 27
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.

NEP 6.6 Sheet 28 | 22
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

R5
R5
R50
R50

**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

NEP 6.6 Sheet 29
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	R38
4. Refueling Water Storage Tank Level	M	R	
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

P.O.D

---

*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 30 / R2
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	R89
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 x 10^{-5} % ~~≥ 1 x 10^{-10} amps~~ OF RATED THERMAL POWER	≥ 6 x 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

NEP 6.6 Sheet 31|R2
Safety Evaluation
No. ECN 6618 6

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

R76

NEP 6.6 SHEET ~~30~~ 32 |R2
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^{-2} 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

NEP 6.6 SHEET ~~34~~ 33 R2
SAFETY EVALUATION
NO. ECN L6186

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
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

R19

NEP 6.6 SHEET ~~32~~ 34/R
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/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

NEP 6.6 SHEET ~~33~~ 35/R5.I
SAFETY EVALUATION.
NO. ECNL6186

TABLE 4.3-6

REMOTE SHUTDOWN MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

ADD INTERMEDIATE

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 *	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

TVA 10551 (EN DES-2-81)

UNREVIEWED SAFETY QUESTION DETERMINATION

Sheet 36 | R2

SHEET #1
L6186R.0
IDENTIFIER

TO: Sequoyah Nuclear Plant, Daisy, TN
FROM: E16 SNP

REV NO.	TOT PP	PREPARED	REVIEWED	APPROVED	DATE APPD	DPM RLSE
0*	2	R.A. Edlund	RRCampbell	R.H. Chin /VAB	1/14/85	Jpv
1						
2						
3						
4						
5						

*R0 INITIAL ISSUE

	MEDS ACCESSION NO.
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 power 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, 102SPT-
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

Sheet ~~2~~ 37.

UNREVIEWED SAFETY QUESTION DETERMINATION

L6186 R.0
IDENTIFIER

Project SQN

INFORMATION ONLY

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 functions 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.

TVA 10551-B (EN DES-7-80)

UNREVIEWED SAFETY QUESTION DETERMINATION

Sheet # ~~3~~ 38

L6186 R-0

IDENTIFIER

Project SQN

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

Safety Evaluation .... ☐

Additional Information .... ☒

(☒, Marks Subject to Which This Page Applies)

Preparer RAE

Reviewer RRC (initial)

INFORMATION ONLY

Potential Tech Spec Change:

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

TVA

Sheet 39

REVISION LOG

Title: Safety Evaluation for ECN L6186

Revision No.	DESCRIPTION OF REVISION	Date Approved
0	Initial Issue	
1	Revised ECN L6186 USQD (RIMS SQP 850117503) to incorporate NEP 6.6 requirements. R0 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.	

TVA 10534 (EN DES-4-78)

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 technical specification 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. The three administrative changes are proposed to support the installation of the Gamma Metrics source range (SR) and intermediate range (IR) detector assemblies. The first involves the deletion of a table note that is not 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 third involves the addition of a certain footnote to enable the review and approval of the unit 1 change to proceed independently of the unit 1 installation schedule. 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 technical specification 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. The three administrative changes are proposed to support the installation of the Gamma Metrics SR and IR detector assemblies. The first involves the deletion of a table note that is not 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 third involves the addition of a certain footnote to enable the review and approval of the unit 1 change to proceed independently of the unit 1 installation schedule. 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 technical specification 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. The three administrative changes are proposed to support the installation of the Gamma Metrics SR and IR detector assemblies. The first involves the deletion of a table note that is not 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 third involves the addition of a certain footnote to enable the review and approval of the unit 1 change to proceed independently of the unit 1 installation schedule. 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 technical specification changes are administrative in nature, the proposed amendment will not involve a significant reduction in a margin of safety.