TENNESSEE VALLEY AUTHORITY

CHATTANOOGA, TENNESSEE 37401

5N 157B Lookout Place

APR 11 1990

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555

Gentlemen:

In the Matter of	)	Docket Nos. 50-327
Tennessee Valley Authority	)	50-328

SEQUOYAH NUCLEAR PLANT (SQN) - EAGLE 21 UNREVIEWED SAFETY QUESTION (USQ)

Reference: TVA letter to NRC dated March 1, 1990, "Sequoyah Nuclear Plant (SQN) - Eagle 21 Upgrade to SQN Reactor Protection System (RPS)"

Enclosed for your information is a copy of the 10 CFR 50.59 evaluation performed in support of the Eagle 21 upgrade modification to the SQN reactor protection system. As detailed in the evaluation, portions of the modification are considered to involve a USQ. Specifically, the possibility of a common mode failure is created by the application of identical software algorithms in more than one process protection channel. The evaluation continues, however, to state that the probability of failure is not increased because of the variety and scope of testing, qualification, and verification. The satisfactory completion of the Eagle 21 design, verification, and validation plan will provide the necessary assurance supporting the conclusion that failure probabilities are not increased. The Eagle 21 design, verification, and validation plan was transmitted by the referenced letter. Documentation of the satisfactory implementation of the plan for the SQN Eagle 21 upgrade will be provided by separate correspondence.

Please direct questions concerning this issue to Russell R. Thompson at (615) 843-7470.

Very truly yours,

TENNESSEE VALLEY AUTHORITY

WJ Ray f.

E. G. Wallace, Manager
Nuclear Licensing and
Regulatory Affairs

Enclosure
cc: See page 2

D030
1/1

U.S. Nuclear Regulatory Commission

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cc (Enclosure):

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Sequoyah Nuclear Plant
2600 Igou Ferry Road
Soddy Daisy, Tennessee 37379

ENCLOSURE

10 CFR 50.59 EVALUATION

(B37 900321 809)

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QA Record

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ATTACHMENT II

SAFETY ASSESSMENT/EVALUATION
COVER SHEET

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B3790 0321 809

Safety Assessment/Evaluation No. M01443A — Revision No. 1	RIMS Accession No.
If Not RO, Previous Revision's RIMS No. B37900310800	

Project and Affected Unit(s) SEQUOYAH, UNIT 1

Activity	Number (Include Revision No.)
✓ Design Change	PMP or DCN No. M01443A/F02878A
___ Temporary Alteration	TACF No. ___
___ Special Test/Experiment	Special Test No. ___
___ Temporary Shielding Request	TSRF No. ___
___ Procedure Change	Procedure No. ___
___ New Procedure	Procedure No. ___

___ Other (Identify) ___

Special Requirements? Yes ✓ No___

Comments: NRC APPROVAL MUST BE RECEIVED PRIOR TO ANY PORTION OF THIS MODIFICATION BEING DECLARED TECH SPEC OPERABLE. THE RPS PANEL WORK MUST BE SEQUENCED DURING THE MODIFICATION TO MAINTAIN TECH. SPEC. REQUIREMENTS.

Distribution (Safety Evaluation Only)

cc: RIMS, ET SLE 26P-K
NE Program Manager (50.59), DSC-D, Sequoyah
Engineering Support Group (Annual Report), O&PS-3, Sequoyah
NSRB Chairman, [illegible] 1N 77B-C
Site Licensing, O&PS-4 Sequoyah
Document Control and Records Management, O&PS-2, Sequoyah - Place copy in open active file of 50.59 library.
Preparer - Return original to originating document

Revision 1 of this SA/SE does not require PORC review since the revision is only to add a reference to a Westinghouse Field Change Notice (FCN). The FCN does not change the scope or conclusions of the origional SA/SE. This was coordinated with LIC and QA.

SAFETY ASSESSMENT

Document No. (i.e., ECN No., procedure No. and revision, special test No., etc.) : DCN M01443A

The documentation necessary to address and provide justification for these questions should be attached to this form. This documentation should provide enough detail for an independent reviewer to concur with only the information provided.

A. **Description**

1. Provide a detailed description of the change, special activity, or condition including the systems, structures, and components affected. Include the number of the activity proposed (e.g., ECN/DCN No., procedure No.).
2. Include a list of references used in completing TVA form 40139. The list should be specific and include revision levels of procedures and drawings.

B. **Impact on Safety** - Use checklist B-1 to determine what safety concerns could be impacted by the proposed change. For each item checked "Yes", provide a discussion of how the change will impact the ability of the plant or plant staff to perform the required functions. This evaluation should address any impacts on functions required in the Safety Analysis Report that are not explicitly included in the checklists. This evaluation should conclude with an answer to the following question:

"Is the change acceptable from a nuclear safety standpoint?"

Yes ✓ No ___

Justification: See attachment 1

If the question in part B is answered in the negative, it is an unsafe change which will require either revision to make it safe or cancellation.

Procedure/Modification Checklist (B-1)

The following items should be considered in evaluating the effects (direct or indirect) of the proposed modification (temporary or permanent) or procedure on nuclear safety. Check those that are considered applicable to the proposed change.

	YES	NO	
1.	✓		Fire Protection (Appendix R)
2.	✓		Internal Flooding Protection (MELB)
3.	✓		Pipe Breaks
4.	✓		Pipe Whip
5.		✓	Modification to Non-Seismic Areas in CB/AB
6.		✓	Radioactive Release Pathways or Quantities (ALARA)
7.	✓		Jet Impingement Effects
8.	✓		Seismic/Dead Weight
9.		✓	Internal/External Missiles
10.	✓		Heavy Load Lifts or Safe Load Paths (NUREG-0612)
11.		✓	Toxic Gases

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12.	___	✓	Hazardous Material
13.	✓	___	Human Factors
14.	✓	___	Electrical Separation/Isolation
15.	___	✓	Primary Containment Integrity/Isolation
16.	___	✓	Secondary Containment Integrity/Isolation
17.	✓	___	Equipment Reliability
18.	___	✓	Materials Compatibility
19.	✓	___	Single Failure Criteria
20.	___	✓	Control Room Habitability
21.	✓	___	Environmental Qualification Category
22.	✓	___	Equipment Failure Modes
23.	___	✓	Tornado or External Flood Protection
24.	___	✓	Protective Coatings Inside Containment
25.	✓	___	Water Spray/Condensation
26.	✓	___	System Design Parameters
27.	✓	___	FSAR/Text Affect
28.	✓	___	Test and Retest Scoping Document (Post Modification Test)
29.	✓	___	CAQ
30.	___	✓	Chemistry Changes or Chemical Release Pathways
31.	✓	___	Equipment Redundancy
32.	✓	___	Equipment Diversity
33.	✓	___	Physical Separation
34.	✓	___	Electrical Loads
35.	✓	___	Response Time of Emergency Safeguards Equipment
36.	✓	___	Safety Injection/Core Cooling Capability
37.	___	✓	Decay Heat Removal Capability
38.	✓	___	Reactor Coolant Pressure Boundary
39.	✓	___	Reactor Core Parameters
40.	___	✓	Pipe Vibration
41.	___	✓	Security System
42.	✓	___	Scaffolding
43.	___	✓	Electrical Breaker Alignment Changes
44.	✓	___	I-TABS, Protection Relay Settings
45.	___	✓	Compensatory Measure

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46. ___ ✓ Environmental Impact Statement
47. ✓ ___ Design Basis Document
48. ___ ✓ Radwaste System Changes
49. ___ ✓ Valve Alignment Changes
50. ___ ✓ Shield Building Integrity (SQN/WBN)
51. ___ ✓ New Radioactive Effluent (Liquid or Gaseous) Release Pathways
52. ___ ✓ Temporary Shielding
53. ✓ ___ Instrument Setpoints
54. ___ ✓ ASME Section XI

C. Potential Safety Analysis Impact

Does the proposed activity affect (directly or indirectly) any information presented in the SAR or deviate from the description given in the SAR:

Yes ✓ No ___ By changing the system design or functional requirements?

Yes ✓ No ___ By changing the text, tables, graphs, or figures?

Justification: ATTACHMENT 1

Does the proposed change involve new procedures or instructions or revisions thereof that:

Yes ✓ No ___ Affect system operation characteristics from that described in the SAR?

Yes ✓ No ___ Affect methods of ensuring compliance with Technical Specifications?

Yes ✓ No ___ Affect a process or procedure outlined, summarized, or described in the SAR?

Justification: ATTACHMENT 1

If the questions in part C are answered in the negative, there is no safety analysis impact and no USQ exists. If any of the answers to the questions in part C is "Yes", or the proposed activity is a change to the radioactive waste system or a special test or experiment, a 10 CFR 50.59 safety evaluation is required.

D. Potential Technical Specification (T/S) Impact

Yes ✓ No ___ Is a change to the T/S required for conducting or implementing the change, test, or experiment?

Justification: SEE ATTACHMENT 1

If the answer to the question in part D is "Yes", a Technical Specification change is required or the activity needs to be revised or cancelled.

E. Review and Approvals

Preparer JERRY W. WEBB / Jerry W. Webb Date: 3-21-90
Level I - Name / Signature

Approver M.R. SEDLACIK / M.R. SEDLACIK by [signature] Date: 3-21-90
Line Mgr - Name / Signature

Reviewer J.E. STAUB / J.E. Staub Date: 3-21-90
Level II - Name / Signature

SAG Consistency: Ira M. Heatherly Ira M. Heatherly 3/21/90

A. DESCRIPTION

See the attached Westinghouse Safety Evaluations:
SECL-89-863, letter TVA-90-630
SECL-85-405, letter TVA-86-525
letter TVA-84-101

The Foxboro equipment used a fuse to isolate between IE power and the non-divisional rod control and annunciator functions. The new equipment will have isolated outputs and non-divisional 120V AC will be supplied for the control and annunciator output circuits.

The logic changes in the Solid State Protection System (SSPS) cabinets have deleted sixteen computer points, redefined sixteen points for Steam Line Low Pressure, and redefined one point for Safety Injection and Steam Line Isolation Block.

Also included in the SSPS modifications are two unimplemented FCN's (Ref 2). FCN TVA0-40523 will eliminate a potentially undetectable failure in the present SSPS output relay test panel configuration. The wiring change will assure that the SSPS output relay test switch failures will be detected during periodic surveillance tests. See attached Westinghouse Safety Evaluation letter number TVA-84-076. FCN TVA0-40556 will add the "Loss of AC power to the SSPS output relays" as a trouble condition to the General Warning Alarm Reactor Trip System. The wiring change will provide immediate automatic detection of output relay 120V AC power failure. See attached Westinghouse Safety Evaluation (SECL-85-405) letter number TVA-86-525.

The Steam Generator Wide Range Level function has been upgraded from the control cabinets to the protection system cabinets as required by PAM DCN M01500 (Ref 18). These channelized instrument loops will be installed in panels 1-R-2, -6, -10, and -13.

The ECN's and DCN's listed below must also be worked concurrently with this modification:

ECN L6189	Incore Thermocouple Upgrade
ECN L6435	Containment Sump Level Cable Modification
ECN L6450	Saturation Margin Monitor
ECN L6492	Third RCS Wide Range Pressure Channel
DCN M1343	Cntmt Sump Level Transmitter Replacement
DCN M1444	RTD Bypass Elimination
DCN M1496	Control Room Indicators
DCN M1499	PAM Cable Reroute
DCN M1500	PAM Wide Range SG Level Modification
DCN M1538	UHI Removal
DCN M2217	Appendix R Modification (CAQR 890523)

There are no predecessor ECN/DCN'S.

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SYSTEM(S), STRUCTURE(S) and COMPONENT(S) AFFECTED

System 1 - Main Steam System

System 3 - Main and Aux Feedwater System

System 30 - Containment Ventilation System

System 47 - Turbogenerator Control System

System 55 - Annunciator System

System 63 - Safety Injection System

System 68 - Reactor Coolant System

System 85 - Control Rod Drive System

System 99 - Reactor Trip System

Components:

The Foxboro electronics is removed and Eagle 21 electronics is installed in the following panels:

1-R-1, -2, -3, -4	Reactor Protection Set I
1-R-5, -6, -7, -8	Reactor Protection Set II
1-R-9, -10, -11	Reactor Protection Set III
1-R-12, -13	Reactor Protection Set IV

The Median Signal Selector circuits are added to the following panels:

1-R-15	Instrumentation Control Group I
1-R-19	Instrumentation Control Group II
1-R-21	Instrumentation Control Group III
1-R-23	Instrumentation Control Group IV

The following relays located in NSSS Auxiliary Relay Panel 1-R-58 will be rewired to receive non divisional 120 V AC, which is avilable in 1-R-58, as the Eagle 21 no longer provides this non divisional power.

1TB411DX	Overtemp. Hi Delta T Turbine Runback & Rod Stop
1TB421DX	Overtemp. Hi Delta T Turbine Runback & Rod Stop
1TB431DX	Overtemp. Hi Delta T Turbine Runback & Rod Stop
1TB441DX	Overtemp. Hi Delta T Turbine Runback & Rod Stop
1TB411HX	Overtemp. Hi Delta T Turbine Runback & Rod Stop
1TB421HX	Overtemp. Hi Delta T Turbine Runback & Rod Stop
1TB431HX	Overtemp. Hi Delta T Turbine Runback & Rod Stop
1TB441HX	Overtemp. Hi Delta T Turbine Runback & Rod Stop
1PB505CX	Lo-turbine Impulse Chamber Pressure C-5

New annunciator windows will be added to panel 1-XA-55-5A for:

Narrow Range RTD Failure Loop 1
Narrow Range RTD Failure Loop 2
Narrow Range RTD Failure Loop 3
Narrow Range RTD Failure Loop 4

To make room for these windows the following windows are moved:

Channel II Pressurizer Press High	to window 3
Aux Bldg Fl & Equip Dr Sump Lo	to window 11
Shutdown Bd Rm Air Cond Sys A-A Abn	to 1-XA-55-5C window 14
Shutdown Bd Rm Air Cond Sys B-B Abn	to 1-XA-55-5C window 21

New annunciator windows will be added to panel 1-XA-55-3C for:

Steam Generator Loop 1 Lo Lo Level
Steam Generator Loop 2 Lo Lo Level
Steam Generator Loop 3 Lo Lo Level
Steam Generator Loop 4 Lo Lo Level
Adverse Containment Pressure

To make room for these windows the following windows are moved:

Cond Stg Tank Hdr to Aux Fwps Press Low	to window 5
Amsac Power Loss	to window 6
No. 1 Fw Htr Pressure Hi	to window 15
Low Diff Press MFP 1A Seal Water	to window 22
Low Diff Press MFP 1B Seal Water	to window 29
Seal Stm Supply Drain Level High	to 1-XA-55-2C window 32

Existing annunciator windows 7, 14, 21, and 28 on panel 1-XA-55-4A are being redefined for each channel.

Old - Spray Actuation Bypass Channel I Test
New - Protection Set I Bypass

Existing annunciator window 3 on panel 1-XA-55-6A for Process Channel Spray Actuation Test Sequence Violated is deleted and a window 17 is redefined.

Old - Process Protect Racks Channel Test Sequence Violated
New - Protection Set Channel Set Failure

Status lights 19, 39, 59, and 79 on panel 1-XX-55-5 are being redefined for each channel.

Old - Prot 1 Test Pnl Open
New - Prot Set 1 Trouble

The new steam line break logic has changed windows 2, 9, 16, 23, on panel 1-XA-55-6B for each loop.

Old - Stm Gen Loop 1 Hi Steamline Flow
New - Low Steamline Pressure Loop 1
Windows 3, 10, 17, 24 for each loop.
Old - Stm Gen Loop 1 Steamline delta P High
New - High Negative Rate Steamline Pressure Loop 1
Window 31.
Old - Stm Gen Loops Steamline Isol Pressure Low
New - Steamline Isol Hi Negative Rate Steamline Pressure.

The changes in trip logic has changed windows on the reactor first out annunciator 1-XA-55-4D. The following eight windows 2, 7, 12, 17, 27, 32, 33, 37 have been deleted:

Stm Gen Loop Feedwater Flow Lo Reactor Trip (four)
Stm Gen Loop Steamline delta P Hi Safety Inj Trip (three)
SAF Hi Stm Flow with Lo-Lo T avg or Lo Stm Press Reactor Trip
Window 22 is redefined.
Old - Stm Gen Loop 1 Steamline delta P Hi Safety Inj Trip
New - Reactor Trip Low Steamline Pressure Safety Injection.

Status lights on panel 1-XX-55-6B are being redefined.

Windows 1, 2, 3, 4, 25, 26, 27, 28, 50, 51, 73, 76.
Old - delta pressure
New - low steamline pressure
Windows 5, 6, 7, 8, 29, 30, 31, 32.
Old - high steam flow
New - high negative rate steamline pressure
Windows 54, 55, 77, 80 are added.
New - high negative rate steamline pressure
Windows 57, 58, 81, 82 are deleted.

The following changes shall be made to support the additional electrical load of the new electronics.

Panel No.	Circuit Breaker Old	Circuit Breaker New	Cable	120V Vital AC Power Board
1-R-1 & -2	15A	25A	Rerouted (M02217)	1-I
1-R-3 & -4	N/A	25A	New	1-I
1-R-5 & -6	15A	25A	Rerouted (M02217)	1-II
1-R-7 & -8	N/A	25A	New	1-II
1-R-9	15A	N/A	No change	1-III
1-R-10 & -11	N/A	25A	New	1-III
1-R-12 & -13	15A	25A	No change	1-IV

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REFERENCES

1. DCR 3094, DCN M01443, RPS Replacement and Upgrade

2. DCR 3372, SSPS wiring upgrade

3. Westinghouse Field Changes Notices:

TVA0-40568	RTD Bypass Elimination
TVA0-40570	MSS elimination of low feedwater flow reactor trip
TVA0-40571	SSPS steam line protection
TVA0-40572	MSS recording of steam generator level
TVA0-40573	Foxboro removal and Eagle 21 installation
TVA0-40523	SSPS modify output relay test panel to insure detection of a potentially undetectable test circuit failure.
TVA0-40556	SSPS add loss of output relay voltage detection.
TVAC-40580	Eagle 21 Process Protection EPROM Installation R1

4. Westinghouse WCAP-8587, Supplement 1, EQDP-ESE-69 "Equipment Qualification Data Package".

5. Westinghouse WCAP-8587 "Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety Related Electrical Equipment".

6. Westinghouse WCAP-11733, Noise, Fault, Surge Withstand Capability, and Radio Interference (RFI) test program.

7. Westinghouse WCAP-12417, Median Signal Selector (MSS) for Foxboro Series Process Instrumentation.

8. IEEE Std. 279-1971 "Criteria for Protection Systems for Nuclear Power Generating Stations".

9. CAQR 890532 Rev 1, Appendix R interaction along the Q line wall on elevation 714' of the auxiliary building.

10. DCR 3415, DCN M02217, Reroute Appendix R cables and wrap conduits.

11. Design Criteria SQN-DC-V-13.10 Rev. 2 "Seismically Qualifying Conduit Supports"

12. SQN-CPS-027 Rev 0, Power Analysis for Eagle 21 modification.

13. AI-17 Rev. 13, "Drilling, Cutting, Chipping, and Excavating"

14. Design Criteria SQN-DC-V-26.2 Rev 2 "Environmental Qualification to 10CFR 50.49"

15. Westinghouse WCAP-11239 Rev 4, Setpoint Methodology

16. Westinghouse Precautions, Limitations, and Setpoint Document

17. DCN M01444, RTD Bypass Removal

18. DCN M01500, PAM Steam Generator Level and Pressure

19. SQN-DC-V-12.2 Rev 6, Separation of electric equipment and wiring.

20. SQN-SQA4-127 Rev 10, Equipment required for safe shutdown per 10CFR50 Appendix R.

21. SQN-CLS-008 Rev 2, Mini Calc for Appendix R.

22. AI-56 Rev 1, Criteria for the erection of scaffolds including those in seismically qualified structures.

23. CEB Report 72-22 Rev 4, Evaluation of the effects of postulated pipe failures outside of containment Sequoyah Nuclear Plants Unit 1 and 2.

24. M&AI-6 Rev 12, Installation of conduit and junction boxes.

25. Westinghouse WCAP-12374 "Topical Report - Eagle 21 Microprocessor Based Process Protection System"

26. SQN-410-D053-EPM-JEI-082187, Calculation on Control Building Flooding due to condenser circulating water pipe breaks in the Turbine Building, (RIM'S B87890930079)

B. IMPACT ON SAFETY

Per Modification Checklist 40139 -

1 - The new power feeds are Appendix R cables. These cables are routed in channelized conduits from the 120 Volt AC Power Boards. DCN 2217 (Ref 10) will reroute the power feeds for 1-R-1 and 1-R-5 with the new power feeds added by this modification. These conduits are routed per separation requirements of SQN-DC-V-12.2 (Ref 19). Also the Appendix R interaction identified by CAQR 890532 (Ref 9) has caused the design of the conduit in the Appendix R (Ref 20) area to be installed with fire barrier(wrap) on the conduits (Ref 21). With the implementation of the above measures this is an acceptable change.

2 - The water level in the control building could be 2.5 inches per calculation SQN-410-D053-EPM-JEI-082187 (Ref 26). The equipment in the auxiliary instrument room panels is located more than six inches above the floor, this is acceptable .

3 - The proposed conduit routing for DCN M01443A is acceptable. There are no potential unacceptable pipe rupture interactions (due to pipe whip or jet impingement) with the proposed conduit because there are no high energy or high pressure/low energy piping in the Control Bay or in the vicinity of the conduit in the Auxiliary building (CEB Report 72-22, Ref 23).

4 - See comment in number 3 above.

7 - See comment in number 3 above.

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ATTACHMENT 1

8 - New conduits containing IE cables are seismic category 1. Conduits containing non-IE cables are installed seismic 1(L). Conduit are seismically qualified in accordance with reference 11 and shall be installed in accordance with references 13 and 24.
The installation of the new electronics in panels 1-R-1 through 1-R-13 have been analyzed by Westinghouse WCAP 8687 (Ref 4) and are seismically qualified and is acceptable.

10- The new power feed cables for the reactor protection system will be routed through the auxiliary building elevation 714. The conduit and cable are routed along the Q line wall in the auxiliary building and is above elevation 723'6", therefore it is not in the NUREG 0612 area and is acceptable.

13- The annunciator window, status light, and computer point changes have had Human Factor Engineering (HFE) Review per the Electrical Engineering Instruction EE-I-22.1-6 Rev 0 and are acceptable.

14- The new cables for power supplies are channelized IE, the annunciation cables are non-divisional as are the cables for the median signal selector circuits. (See attached Westinghouse Safety Evaluation SECL-89-863 page 22 for discussion of the median signal selector). These cables shall be routed per separation requirements of SQN-DC-V-12.2 R6 (Ref 19).
Eagle 21 provides Class IE isolated outputs. The analog output module incorporates surge protection and electrical isolation to prevent destructive transients from propagating from the field conductors back through to the Loop Processor Subsystem. The partial trip output module and the contact output module provides Class IE isolation between the Loop Processor Subsystem and the final output. Filters are incorporated on the I/O Subsystem inputs to provide surge protection, provide for electrical noise rejection and to remove process and A/D quantization noise. This installation is acceptable.

17- The reliability of the new equipment has been verified by the Westinghouse WCAP's (Ref 4, 5, and 6) and is acceptable.

19- See comment in attached Westinghouse Safety Evaluation SECL-89-863 page 28 section A. Single Failure Criteria, and is acceptable.

21- The cables are qualified for the environment for which they are exposed in accordance with reference 14. Therefore, the degradation of cables due to environmental effects will not occur to the extent that plant safety will be affected during an accident condition for the remainder of the plant life.
The new equipment installed in panels located in the auxiliary instrument room is located in a mild environment and is qualified per Westinghouse WCAP's (Ref 4, 5, 6, and 7) and is acceptable.

22- See attached Westinghouse Safety Evaluation SECL-89-863 page 33 Conclusion, upon completion of Validation and Verification program this will be acceptable. The Verification and Validation is not complete and therefore must be considered a USQ.

25- The new IE cables will be in a fire suppression area in the auxiliary building elevation 714. These cables are routed in conduit and will not be effected by spray and is acceptable.
The new electronics are located in the control building auxiliary instrument room which is not subject to water spray and is acceptable. This installation is acceptable.

26- The setpoint values are justified in the Westinghouse Setpoint Methodolgy WCAP (Ref 15) and are acceptable.
See the attached Westinghouse Safety Evaluation SECL-89-863:
Median Signal Selector elimination of low feedwater flow reactor trip on page 22.
The new steamline break protection system rewiring of the SSPS on page 24.
Hardware modification Foxboro removal and Eagle installation on page 26.
Trip Time Delay on page 15 and 18.
Environmental Allowance Modifier page 18.

27- The FSAR is revised to address the following changes:
The new steamline break protection system rewiring of the SSPS
Median Signal Selector elimination of low feedwater flow reactor trip
Removal of the Foxboro H-line analog system and installation of the Eagle 21 digital system
Environmental Allowance Modifier (EAM)
Trip Time Delay (TTD).

28- Eagle 21 Post Modification Test (PMT) number 125 will be run to verify the equipment is able to provide the designed safety function.

29- CAQR 890523 identified an Appendix R interaction in the auxiliary building with the power supply cables for protection set I and II. This modification installed new conduits with 1 hour fire wrap for cables to 1-R-3 and 1-R-7. DCN M02217 rerouted the cables for 1-R-1 with the new cable for 1-R-3 (Protection Set I) and rerouted the cable for 1-R-5 with the new cable for 1-R-7 (Protection Set II). This has resolved the Appendix R interaction problem.

31- See attached Westinghouse Safety Evaluation SECL-89-863 page 28 sections, A. Single Failure Criteria and 3. Channel Independence and is acceptable.

32- See attached Westinghouse Safety Evaluation SECL-89-863 page 33. The required testing will demonstrate no common mode failure exists and is acceptable.

33- Physical separation of cable/conduit is designed and field routed per separation requirements of SQN-DC-V-12.2 (Ref 19), SQN-SQS4-127 (Ref 20) and mini calculation for Appendix R (Ref 21). The conduit on elevation 714' of the auxiliary building will have a 1 hour rated fire barrier (wrap) to meet the Appendix R requirements. The four reactor protection channels are located in panels which are physically separate and electrically separate and is acceptable.

34- Electrical loads have increased and have been analyzed by calculation SQN-CPS-027 (Ref 12) and will have no adverse effect on the power supply.

35- See attached Westinghouse Nuclear Safety Evaluation SECL-89-863 page 3 paragraph on RTD bypass system and page 6 paragraph on the environmental allowance modifier, and is acceptable.

36- See attached Westinghouse Safety Evaluation SECL-89-863 page 24 New Steamline Break Protection System, and is acceptable.

38- The change associated with reactor coolant pressure boundary are discussed in DCN M01444 and attached Westinghouse Safety Evaluation SECL-89-863 page 20, and is acceptable.

39- See attached Westinghouse Safety Evaluation SECL-89-863 pages 7 through 14 , 22 and 24, and is acceptable.

42- Temporary scaffolding may be required during implementation of this modification. The amount of scaffolding required is minimal and will not impede operation personnel from performing essential plant functions, nor will it impose any threat to safety related systems. The scaffolding shall be installed in accordance with AI-56 (Ref 22) and is acceptable.

44- Instrument Tabulations for systems 1, 3, 30, 63, and 68 are being revised to add new equipment in the RPS electronics and the revised setpoints (Ref 15 and 16). Also a new instrument tabulation is being added for system 99 which identifies system cards and hardware in the reactor protection system racks. These changes for the I-tab's are acceptable.

47- The design basis document is changed. This modification will add system 99 functional logic diagrams.

53- Instrument setpoints have been analyzed by the Westinghouse Setpoint Methodolgy WCAP (Ref 15) and are acceptable.

The scope is acceptable from a nuclear safety standpoint.

C. POTENTIAL SAFETY ANALYSIS IMPACT

The following sections will be revised for the digital electronics and reactor trip changes.

Sections: 3.1, 5.6, 6.2, 6.3, 7.1, 7.2, 7.3, 7.7, 8.3, 9.4 (figures only), 10.3 (figures only) RRM 3-9-90, 10.4, 15.1, 15.2.2, 15.2.4, 15.2.7, 15.2.8, 15.2.9, 15.2.12, 15.4.2.2

Surveillance and maintenance instructions for the existing Process Protection System must be revised to reflect the change from Foxboro analog equipment to the new Eagle 21 digital equipment. Operating instructions currently in use must be revised according to the functional changes being implemented with the Eagle 21 modification as described in the Westinghouse safety evaluation.

D. POTENTIAL TECHNICAL SPECIFICATION (T/S) IMPACT

The following sections and tables will be revised by this modification.

Sections: 1.6, 2.2.1

Tables: 2.2-1, 3.3-1,-2,-3,-4,-5, 4.3-1,-2

See attached letter, Subject Eagle 21 Installation, (RIM's S10900302831).

Letter TVA-SQN-TS-89-27, RIM's No. L44900124802, SQN-Technical Specification Change 89-27 has been submitted to the U.S. Nuclear Requlartory Commission.

SAFETY EVALUATION

Safety Evaluation No: ______________________

Document No. (i.e., ECN No., procedure No. and revision, special test No., etc.): DCN M01443A

The documentation necessary to address and provide justification for these questions should be attached to this form. This documentation should provide enough detail for an independent reviewer to concur with only the information provided.

The results of Part A (Description) of the Safety Assessment (TVA form 40139) should be included.

A. Introduction - Discussion of Change

1. Describe the expected effects of change being evaluated.
2. Identify the parameters and systems affected by the change.
3. Identify credible failure modes of the change.
4. Provide references to location of information used for the safety evaluation.

B. Effect on the Accidents Evaluated as the Design Basis

1. Identify the design basis accidents in the SAR to be reviewed for potential impact by the change.
2. Discuss how the parameters and systems, affected by the change, affect the radiological consequences of the accidents identified in part B.1. The affected parameters and systems are identified in part A.2.
3. Identify the design basis accidents, if any, for which failure modes associated with the change can be an initiating event.
4. Discuss the effect of the change on the probability of occurrence of the design basis accidents identified in part B.3 and identify the safety systems and systems important to safety affected by the change.
5. Discuss the effect of the change and/or the failure modes associated with the change on the probability of failure of the systems identified.
6. Discuss the effect of the change on the performance of the safety systems.

Yes ______ No ✓ Based on the results of part B.2, does the change increase the radiological consequences of a design basis accident?

Yes ______ No ✓ Based on the results of part B.3 and B.4, does the change increase the probability of a design basis accident?

Yes ______ No ✓ Based on the results of part B.5, does the change increase the probability of a failure of a safety system?

Yes ______ No ✓ Based on the results of part B.6, does the change degrade the performance of a safety system below that assumed in the design basis analysis?

If any of the above four questions is answered "Yes", the change involves an unreviewed safety question.

C. Potential for Creation of a New Type of Unanalyzed Event

Based upon part B, assess the impact of the change and/or failure modes associated with the change, to determine if the impact has modified the plant response to the point where it can be considered a new type of accident. Discuss the basis for this determination including whether:

- The failure modes of equipment important to safety associated with the change represent a new unanalyzed type of malfunction.

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SAFETY EVALUATION

The change or a failure mode associated with the change, increases the probability of an accident to the point where it should be considered within the design basis.

Yes ✓ No ______ Based on the results of part C, does the change create the potential for a new type of unanalyzed accident or a new type of malfunction?

If the answer is "Yes", the change represents an unreviewed safety question.

D. Impact on the Margin of Safety

1. Identify the acceptance limits, from the licensing basis for the Technical Specifications (i.e. the accident analysis and other design basis), that could be affected by the change.

2. Discuss the impact of the change on the acceptance limits which were identified.

Yes ______ No ✓ Based on the results of part D, does the change decrease the margin of safety as defined in the basis for any Technical Specification?

If the answer is "Yes", the change represents an unreviewed safety question.

E. Unreviewed Safety Question Determination Conclusion

Based on the results of part B, C, and D, the change:

______ Does not involve an unreviewed safety question.

✓ Involves an unreviewed safety question and must be revised, cancelled, or reviewed by NRC prior to implementation.

F. Reviews and Approvals

	Name	Signature	Date
Preparer	JERRY W. WEBB (Level I - Name)	Jerry W. Webb	3-21-90
Approver	M.R. SEDLACIK (Line Mgr - Name)	M. R. SEDLACIK by [signature]	3-21-90
Reviewer (a) (Optional)	N/A (Level I - Name)		
Reviewer	J.E. STAUB (Level II - Name)	J. E. Staub	3-21-90
Reviewer (b,c)	N/A (PORC/QR - Name)	N/A	
Approver (b)	D.L. Michlink (Plant Manager or designee - Name)	Douglas L. Michlink	3/21/90

SAG CONSISTENCY: Ira M. Heatherly Ira M. Heatherly 3/21/90

a. If a review is not performed this line may be "N/A" by the Level II.
b. These reviews and approvals are not required for corporate level procedures.
c. As required by Technical Specification.

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A. DISCUSSION OF CHANGE

See the attached Westinghouse Safety Evaluations:
SECL-89-863, letter TVA-90-630
SECL-85-405, letter TVA-86-525
letter TVA-84-101
Also see the description of change in the TVA safety assessment.

CREDIBLE FAILURE MODES

See attached Westinghouse Safety Evaluation SECL-89-863 page 33.

REFERENCES

1. DCR 3094, DCN M01443, RPS Replacement and Upgrade

2. DCR 3372, SSPS wiring upgrade (FCN's TVAO-40523 and TVAO-40556)

3. Westinghouse Field Changes Notices:

TVAO-40568	RTD Bypass Elimination
TVAO-40570	MSS elimination of low feedwater flow reactor trip
TVAO-40571	SSPS steam line protection
TVAO-40572	MSS recording of steam generator level
TVAO-40573	Foxboro removal and Eagle 21 installation
TVAO-40523	SSPS modify output relay test panel to insure detection of a potentially undetectable test circuit failure
TVAO-40556	SSPS add loss of output relay voltage detection
TVAO-40580	Eagle 21 Process Protection EPROM Installation | R1

4. Westinghouse WCAP-8587, Supplement 1, EQDP-ESE-69 "Equipment Qualification Data Package".

5. Westinghouse WCAP-8587 "Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety Related Electrical Equipment".

6. Westinghouse WCAP-11733, Noise, Fault, Surge Withstand Capability, and Radio Interference (RFI) test program.

7. Westinghouse WCAP-12417, Median Signal Selector (MSS) for Foxboro Series Process Instrumentation.

8. IEEE Std. 279-1971 "Criteria for Protection Systems for Nuclear Power Generating Stations".

9. FSAR Sections 3.1, 5.6, 6.2, 6.3, 7.1, 7.2, 7.3 ,8.3, 10.4, 15.1, 15.2.2, 15.2.4, 15.2.7, 15.2.8, 15.2.9, 15.2.12, 15.4.2.2; Figures 5.1-2,-2,-3,-4,-5,-8,-9,-10,-11, 6.3.2-2,-3,-4, 7.2.1-1, 8.3.1-29,-30,-31,-32, 9.4.8-1, 10.3.2-2,-3,-4, 10.4.7-5,-6; Table 7.2.1-1, 8.3.1-11,-12,-13,-14

10. Design Criteria DIM-SQN-DC-V-27.9-8, Reactor Protection System

11. Westinghouse Nuclear Safety Evaluation Checklist SECL-89-863, EAGLE-21 Process Protection System Upgrade and RTD Bypass Elimination

12. Westinghouse WCAP-12374 "Topical Report - Eagle 21 Microprocessor Based Process Protection System"

B. <u>EFFECT ON THE ACCIDENTS EVALUATED AS THE DESIGN BASIS</u>

Evaluation of Non-LOCA transients is discussed in the Westinghouse Safety Evaluation SECL-89-863 beginning on page 5.

The evaluation of LOCA accidents is discussed in the Westinghouse Safety Evaluation SECL-89-863 beginning on page 17.

C. <u>POTENTIAL FOR CREATION OF A NEW TYPE OF UNANALYZED EVENT</u>

See the attached Westinghouse Safety Evaluation SECL-89-863 page 33. The Westinghouse and TVA Safety Evaluations conclude this is a USQ.

D. <u>IMPACT ON THE MARGIN OF SAFETY</u>

See the attached Westinghouse Safety Evaluation SECL-89-863 page 28.

The modification to the reactor protection system panels 1-R-1 through 1-R-13 will be sequenced to assure having at least one channel of instrumentation operational for the cold over pressurization function and the containment ventilation function per Tech Spec requirements.

E. UNREVIEWED SAFETY QUESTION DETERMINATION CONCLUSION

Based on the above discussions and the conclusion in the attached Westinghouse Safety Evaluation SECL-89-863 page 33, implementation of the proposed activity does result in a USQ.

Installation DCN M01443 may proceed, however, NRC approval must be received and this safety evaluation must be revised prior to any portion of this modification being declared Tech Spec operable.

The demonstrated accuracy calculations listed below contain unverified assumptions which must be resolved prior to returning affected equipment to Tech Spec operable status.

Calculation Number:
SQN-EEB-MS-TI28-0010
SQN-EEB-MS-TI28-0012
SQN-EEB-MS-TI28-0013
SQN-EEB-MS-TI28-0015
SQN-EEB-MS-TI28-0024
SQN-EEB-MS-TI28-0025
SQN-EEB-MS-TI28-0026
SQN-EEB-MS-TI28-0027

UNITED STATES GOVERNMENT

Memorandum

DCN M01443A
Page 65

S10 900302 831

TENNESSEE VALLEY AUTHORITY

TO : P. G. Trudel, Project Engineer, DSC-E, Sequoyah Nuclear Plant

MAR 21 1990

FROM : M. J. Burzynski, Manager, Site Licensing, O&PS-4, Sequoyah Nuclear Plant

DATE : MAR 02 1990

DCN No. F02878
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SUBJECT: SEQUOYAH NUCLEAR PLANT (SQN) - EAGLE 21 INSTALLATION

As noted in Westinghouse Electric Corporation's letter, TVA-90-630, the new Eagle 21 design is considered to represent an unreviewed safety question. Since the protection set provides reactor trip and safety injection actuation functions that are not required in Modes 5 and 6, it is acceptable to proceed with installation of the system. Operation in Modes 1 through 4 cannot occur until NRC approves the Eagle 21 technical specification changes, which effectively constitutes their approval of the unreviewed safety question. It should be recognized that proceeding with installation prior to NRC approval constitutes some risk if NRC cannot, for some unanticipated reason, approve the Eagle 21 design. The results of the NRC review effort to date lead us to believe that the Eagle 21 design will be approved for use.

Mark J. Burzynski
M. J. Burzynski

RRT:SFH
cc: RIMS, MR 4N 35A-C

0002y

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QA Record

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B25 '90 0302 006

Westinghouse Electric Corporation
Energy Systems

Nuclear and Advanced Technology Division

Box 355
Pittsburgh Pennsylvania 15230-0355

Mr. P. G. Trudel
Project Engineer
Tennessee Valley Authority
P. O. Box 2000
Soddy Daisy, TN 37379

TVA-90-630
NS-OPLS-OPL-II-90-169
March 1, 1990
Ref: 1. TVA Contract
89NNP-75380A

Tennessee Valley Authority
Sequoyah Nuclear Plants Units 1 and 2
RTD Bypass Elimination/Eagle-21 RPS Upgrade Program
Safety Evaulation (Integrated)

Dear Mr. Trudel:

Attached is the Safety Evaluation for the subject, which integrates all of the previously supplied separate preliminary safety evaluations.

Please note that this Safety Evaluation is contingent upon completion of Verification and Validation.

If you have any questions, please do not hesitate to contact us.

Very truly yours,

WESTINGHOUSE ELECTRIC CORPORATION

B. J. Garry

B. J. Garry, Manager
TVA Sequoyah Project
Customer Projects Department

LVT/lg

cc: D. M. Lafever, DSC-C20
R. G. Davis

RIMS, ET SLE 26P-K - w/attachments

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Customer Reference No(s).

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Westinghouse Reference No(s).
FCNs TENO-40569,A,-40570,A,-40572,A,
-40574,A,-40567B
TVAO-40568B,-40570,A,-40571,A,
40572,A,-40573,-49574,A

WESTINGHOUSE
NUCLEAR SAFETY EVALUATION CHECK LIST

1) NUCLEAR PLANT(S) SEQUOYAH UNIT 1

2) CHECK LIST APPLICABLE TO: RTD BYPASS ELIMINATION / EAGLE 21 RPS UPGRADE (ANALOG TO DIGITAL) / NEW SLB / MSS-ELIMINATION OF LOW FEEDWATER FLOW REACTOR TRIP / EAM / TTD

3) The safety evaluation of the revised procedure, design change or modification required by 10CFR50.59 has been prepared to the extent required and is attached. If a safety evaluation is not required or is incomplete for any reason, explain on Page 2.

Parts A and B of this Safety Evaluation Check List are to be completed only on the basis of the safety evaluation performed.

CHECK LIST - PART A - 10CFR50.59(a)(1)

(3.1) Yes X No ___ A change to the plant as described in the FSAR?
(3.2) Yes ___ No X A change to procedures as described in the FSAR?
(3.3) Yes ___ No X A test or experiment not described in the FSAR?
(3.4) Yes X No ___ A change to the plant technical specifications (See Note on Page 2)

4) CHECK LIST - PART B - 10CFR50.59(a)(2) (Justification for Part B answers must be included on page 2.)

(4.1) Yes ___ No X Will the probability of an accident previously evaluated in the FSAR be increased?
(4.2) Yes ___ No X Will the consequences of an accident previously evaluated in the FSAR be increased?
(4.3) Yes X No ___ May the possibility of an accident which is different than any already evaluated in the FSAR be created?
(4.4) Yes ___ No X Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?
(4.5) Yes ___ No X Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?
(4.6) Yes X No ___ May the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR be created?
(4.7) Yes ___ No X Will the margin of safety as defined in the bases to any technical specification be reduced?

If the answers to any of the above questions are unknown, indicate under 5) REMARKS and explain below.

If the answer to question (3.4) of Part A or any of the questions in Part B cannot be answered in the negative, based on written safety evaluation, the change review requires an application for license amendment as stated in 10CFR50.59(c) and must be submitted to the NRC pursuant to 10CFR50.90.

5) REMARKS:

The following summarizes the justification upon the written safety evaluation, (*) for answers given in (3.4) of Part A and in Part B.

SEE FOLLOWING PAGES

(*) Reference to document(s) containing written safety evaluation: ______

FOR FSAR UPDATE

Section:______ Page(s):______ Table(s):______ Figure(s):______

Reason for / Description of Change:

THE APPLICABLE UPDATES HAVE BEEN TRANSMITTED UNDER SEPARATE COVER, AS FOLLOWS:

CHAPTER 15 (BASE SCOPE) :	TRANSMITTED VIA LETTER TVA-89-877
(EAM / TTD) :	TRANSMITTED VIA LETTER TVA-89-944
(SEC 15.1) :	TRANSMITTED VIA LETTER TVA-90-601
CHAPTER 7 :	TRANSMITTED VIA LETTER TVA-89-1021
TECH SPECS :	TRANSMITTED VIA LETTER TVA-90-602
SETPOINT WCAP :	TRANSMITTED VIA LETTER TVA-90-598

Prepared by (Nuclear Safety): L.V.Tomasic Date: 3-1-90

Coordinated with Engineer(s): Signatures On File Date: ______

Coordinating Group Manager(s): Signatures On File Date: ______

Nuclear Safety Group Manager: S.D.Rupprecht Date: 3-1-90

SUMMARY

This safety evaluation addresses the changes for the RTD Bypass Elimination, Eagle 21 Reactor Protection System Upgrade (from analog to digital microprocesser), New Steamline Break, Median Signal Selector (Elimination Of Low Feedwater Flow Reactor Trip), Environmental Allowance Modifier (EAM), and Trip Time Delay (TTD). This safety evaluation also provides the basis for the marked-up Technical Specifications (issued under separate cover as noted on page 2), and is contingent upon completion of Verification & Validation, that would conclude that these changes would not represent an unreviewed safety question pursuant to 10 CFR 50.59, (a), (2) criteria, and would support a no significant hazards consideration pursuant to 10 CFR 50.92, (c) criteria.

This safety evaluation addresses the applicable FSAR Chapters 6 & 15 Accidents and the Hardware Modifications described in Field Change Notices (FCNs) provided for Unit 1, or for Units 1 & 2, as noted for the pressure boundary integrity the the RTD thermowells (FCNs TENO-40567B & TVAO-40568B) (Units 1 & 2), removal of the Foxboro H-Line analog process protection system and installation of the Eagle 21 microprocessor based digital process protection system (FCN TVAO-40573) (Unit 1), elimination of the low feedwater flow reactor trip for the Median Signal Selector (MSS) installation (FCNs TENO-40569, 40569A A & 40570, 40570A , and TVAO-40570, 40570A & 40571, 40571A) (Units 1 & 2), recording of the Steam Generator Level Median Signal for the Median Signal Selector (MSS) installation modification (FCNs TENO-40572, 40572A & TVAO-40572, 40572A) (Units 1 & 2), installation of the New Steamline Break Protection System rewiring of the SSPS (FCNs TVAO-40571, 40571A and TENO-40570, 40570A) (Units 1 &2), and the Steam Generator Wide Range Level PAMS Modification (FCNs TVAO-40574, 40574A & TENO-40574, 40574A) (Units 1 & 2)

Elimination of the RTD Bypass system causes an increase in the response times of the temperature detectors. This increase in response time causes the Overtemperature ΔT and Overpower ΔT reactor trips to be delayed. Previously, Sequoyah was limited by Technical Specifications to an RTD response time of 6.0 seconds for the ΔT reactor trips. This evaluation will justify a response time of 8.0 seconds for ΔT reactor trips.

A discussion of the interaction between the TTD and the Seqyouah AMSAC (ATWS Mitigating Actuation Circuitry) system is also included in this evaluation.

The Environmental Allowance Modifier (EAM) would be disabled during the time-periods that the containment ventilation system purge valves would be open.

Protection System uncertainty calculations were performed and documented in the Setpoint Methodology For Protection Systems, Eagle Version, WCAPs 11239 and 11626, Revision 4 (issued under separate cover as noted on page 2).

The FSAR does not evaluate accidents which are the result of the process protection system failing to perform its intended function. Common mode failure of more than one process protection channel could lead to the possibility of an accident being created which is different than any already evaluated in the FSAR, the probability increased of a malfunction of equipment important to safety previously evaluated in the FSAR, and the possibility created of a malfunction of equipment important to safety different than any already evaluated in the FSAR.

The following preventative measures have been taken to ensure that the integrity of the process protection system will not be subject to common mode failure:

Seismic and Environmental Qualification

Noise, Fault, Surge Withstand Capability, EMI and RFI Qualification Testing

Process Protection System Reliability Study

Verification and Validation (V&V) Program

Independent Design Specification Verification

Independent Software Verification

Independent System Validation Testing

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NON-LOCA TRANSIENTS

BACKGROUND

The Tennessee Valley Authority intends to install the Eagle 21 Digital Protection System in the Sequoyah Units 1 and 2. The new protection system is designed to support the following changes to the current plant configuration:

a) Resistance Temperature Detector (RTD) Bypass Elimination
b) New Steamline Break Protection
c) Elimination of the Low Feedwater Flow Reactor Trip, accomplished by the use of a Median Signal Selector (MSS).
d) Environmental Allowance Modifier/Trip Time Delay (EAM/TTD)

Elimination of the RTD Bypass system causes an increase in the response times of the temperature detectors. This increase in response time causes the Overtemperature ΔT and Overpower ΔT reactor trips to be delayed. Previously, Sequoyah was limited by Technical Specifications to an RTD response time of 6.0 seconds for the ΔT reactor trips. This evaluation will justify a response time of 8.0 seconds for ΔT reactor trips.

Typically, there are two types of Steamline Break Protection in use at Westinghouse Pressurized Water Reactors. The current configuration of the Sequoyah Nuclear Plant reactor protection system includes safety injection and steamline isolation actuation logic commonly known in Westinghouse plants as Old Steamline Break Protection. With Old Steamline Break protection a safety injection signal will be generated with any of the following logic satisfied:

1. Low steamline pressure coincident with high steamline flow.
2. Low-low average coolant temperature coincident with high steamline flow
3. High steamline differential pressure
4. Low pressurizer pressure
5. High containment pressure

With Old Steamline Break protection a steamline isolation signal will be generated with any of the following logic satisfied:

1. Low steamline pressure coincident with high steamline flow.
2. Low-low average coolant temperature coincident with high steamline flow
3. High-High containment pressure

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With the introduction of Eagle 21 digital electronics, Sequoyah will change to New Steamline Break Protection. New Steamline Break protection actuation of safety injection will result from any of the following:

1. Low Steamline Pressure
2. Low Pressurizer Pressure
3. High Containment Pressure

With New Steamline Break Protection, steamline isolation is actuated on:

1. High-High Containment Pressure
2. High Steamline Pressure Rate (coincident with manual block of the Low Steamline Pressure safety injection/steamline isolation function below the P-11 permissive)
3. Low Steamline Pressure

This evaluation will justify using New Steamline Break Protection at Sequoyah. The non-LOCA transients, during which steamline isolation or safety injection occurs, must be reviewed to insure that any impact due to the differences between the logic is acceptable.

The elimination of the Low Feedwater Flow Reactor Trip and Steam Flow/Feed Flow mismatch logic potentially impacts the non-LOCA safety analyses by eliminating a reactor trip which could have been credited. This evaluation will demonstrate that this reactor trip is not used as a primary trip in any of the non-LOCA analyses and can therefore be removed from the protection system with respect to these analyses.

The Environmental Allowance Modifier and Trip Time Delay are modifications to the Steam Generator Low-Low Water Level Reactor Trip logic intended to reduce the number of unnecessary feedwater related reactor trips. The Environmental Allowance Modifier monitors containment pressure to sense the presence of a feedline break. When a normal containment environment is detected, the Steam Generator Low-Low Water Level reactor trip setpoint is lower due to elimination of the adverse environmental error component of the Technical Specification setpoint. A higher setpoint is activated when an adverse environment is detected. The safety analysis assumption used in the determination of the Steam Generator Low-Low Water Level reactor trip setpoints was 0% of narrow range span (NRS). The Trip Time Delay is a set of curves of Steam Generator Low-Low Water Level reactor trip time delays versus power level. The time delay will be implemented below 50% power and is dependent on the number of steam generators experiencing a steam generator water level transient. This safety evaluation will discuss the impact of implementing the EAM/TTD system on the non-LOCA transients which depend on steam generator low-low level protection.

A discussion of the interaction between the TTD and the Seqyouah AMSAC (ATWS Mitigating Actuation Circuitry) system is also provided.

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In summary, the following protection channels from the current Sequoyah configuration are impacted by the changes:

a) Overpower ΔT
b) Overtemperature ΔT
c) SIS Low Steamline Pressure Coincident with High Steamline Flow (the Low Steamline Pressure signal will be retained w/o the coincidence logic)
d) SIS High Steamline Flow Coincident with Lo-Lo Tavg
e) High Steamline Differential Pressure
f) Low Feedwater Flow Reactor Trip
g) Steam Generator Low-Low Water Level Reactor Trip

The following discussion will examine all non-LOCA events to determine if the reactor trip which has been credited is going to be eliminated or delayed or if the safety injection and steamline isolation actuation signals which are credited will be eliminated.

EVALUATION OF NON-LOCA TRANSIENTS

Uncontrolled RCCA Withdrawal From a Subcritical Condition (FSAR 15.2.1)

An RCCA bank withdrawal accident is defined as an uncontrolled addition of reactivity to the reactor core caused by withdrawal of one or more RCCA banks, resulting in a power excursion. The Rod Withdrawal from Subcritical event is not affected by the changes previously described because it relies on the High Neutron Flux- Low Setting to provide reactor trip. Steamline isolation and safety injection are not modeled in this analysis.

Uncontrolled RCCA Withdrawal at Power (FSAR 15.2.2)

The Rod Withdrawal at Power event is analyzed with a variety of reactivity insertion rates at 10%, 60% and 100% of rated thermal power. Depending on the case analyzed, either the High Neutron Flux or Overtemperature ΔT reactor trip occurs. This event was reanalyzed in support of the RTD Bypass Elimination. The safety analysis criterion of the DNBR limit being met was satisfied with the increased RTD response time.

Safety injection and steamline isolation are not modeled in this analysis.

RCCA Misalignment (FSAR 15.2.3)

Both the dropped rod event and the statically misaligned rod event are not affected by the changes described because it is assumed that they are detected by the Nuclear Instrumentation System, core thermocouples, or rod control position indicators and alarms. In addition, neither steamline isolation or safety injection is assumed to occur during these events.

Uncontrolled Boron Dilution (FSAR 15.2.4)

The Boron Dilution event is analyzed to identify the amount of time available for operator or automatic mitigation of an inadvertent boron dilution prior to complete loss of shutdown margin. This transient is required to be considered for Sequoyah for operational modes 1, 2, and 6. The Mode 2 analysis is unaffected by the functional upgrades because an automatic reactor trip is not assumed. The Mode 6 transient is administratively precluded by the Technical Specifications.

The Mode 1 event is analyzed in two separate cases which assume either that the control rods are in manual mode of operation or automatic. If the control rods are in automatic, the operator would be alerted to the occurrence of a boron dilution by the rod insertion limit alarms. If the rods are in manual, the first indication may be the Overtemperature ΔT reactor trip. The Mode 1-rods in manual analysis is impacted by the RTD

Bypass Elimination because the time of reactor trip on OTΔT is taken from the Uncontrolled RCCA Bank Withdrawal at Power (RWAP). The time of reactor trip is taken from the RWAP case which has a reactivity insertion rate equal to or less than that calculated for the boron dilution event and then subtracted from the amount of time available between start of event and loss of shutdown margin. Acceptable results were obtained when the RWAP results from the previously discussed reanalysis were used to calculate the amount of time for operator action. It was demonstrated that there were over 40 minutes available between the start of the event and the complete loss of shutdown margin.

Partial Loss of Forced Reactor Coolant Flow (FSAR 15.2.5)
Complete Loss of Forced Reactor Coolant Flow 4(FSAR 15.3.4)

The Loss of Flow events are not impacted by the Eagle 21 functional upgrades. In the Partial Loss of Flow event, a Low Reactor Coolant Flow reactor trip signal is reached. In the Complete Loss of Flow event, an Undervoltage reactor trip is generated. Safety injection and steamline isolation are not assumed to occur during these events.

Startup of an Inactive Reactor Coolant Loop (FSAR 15.2.6)

This transient is caused by the starting of an idle reactor coolant pump without bringing the inactive loop hot leg temperature close to the core inlet temperature. This causes both a significant increase in core coolant flow and injection of cold water into the core which results in a rapid core power increase due primarily to moderator reactivity feedback. The Startup of an Inactive Loop analysis credits a reactor trip on High Neutron Flux. Steamline isolation and safety injection are not assumed. Therefore, the event is not affected by the Eagle 21 functional upgrades.

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Loss of External Electrical Load/Turbine Trip (FSAR 15.2.7)

The Loss of Load/Turbine Trip analysis includes beginning of life and end of life cases both with and without pressurizer control. In the current FSAR analysis, Overtemperature ΔT actuates a reactor trip in the BOL and EOL cases which assume pressurizer control. For this reason, the Loss of Load/Turbine Trip event was reanalyzed in support of the RTD Bypass Elimination. The results of the analysis met the applicable safety criteria.

It should be noted that the Steam Generator Low-Low Water Level reactor trip was actuated in the recent analysis for End of Life- with pressurizer control. The assumption for S/G Low-Low Water Level reactor trip was 0% of narrow range span. As was stated previously, this safety analysis value was used in the determination of the S/G Low-Low Water Level reactor trip setpoints.

The 52% rated thermal power cases of the Loss of Load/Turbine Trip event presented in the Sequoyah FSAR are not impacted by the Eagle 21 functional upgrades because a reactor trip is actuated by Low Reactor Coolant Flow for the cases which assume pressurizer pressure control and High Pressurizer Pressure for the cases which do not assume pressurizer pressure control.

Loss of Normal Feedwater (FSAR 15.2.8)

This event is analyzed to demonstrate that the auxiliary feedwater system is of sufficient capacity to remove core decay heat, stored energy, and RCS pump heat following reactor trip. In the Loss of Normal Feedwater event, a reactor trip on Steam Generator Low-Low Water Level occurs. Steamline isolation and safety injection are not assumed for this event.

The Trip Time Delay was developed by analyzing a number of part-power Loss of Normal Feedwater cases to determine the acceptable amount of delay which could be applied to the S/G Low-Low Water Level reactor trip. These cases were run at various power levels below 50% RTP and modeled either one or multiple affected steam generators. These analyses were completed with the LOFTRAN computer code (ref. 1). The present FSAR full power Loss of Normal Feedwater licensing basis analysis was completed using the MARVEL computer code (ref. 2). Since the TTD equations were determined using the LOFTRAN code, the full power Loss of Normal Feedwater event was reanalyzed to form a consistent and comparable licensing basis. A safety analysis assumption of 0% NRS was used for the Steam Generator Low-Low Level Reactor Trip in both the full and part power cases of the Loss of Normal Feedwater analyses. In all cases, it was demonstrated that the auxiliary feedwater capacity ensures that the RCS heatup is controlled such that the coolant expansion does not result in filling the pressurizer with water.

Loss of Offsite Power to the Station Auxiliaries (Station Blackout (FSAR 15.2.9)

This event is analyzed to show that adequate heat removal capability exists via natural circulation flow as aided by the Auxiliary Feedwater System to remove core decay heat and stored energy following reactor trip. The current FSAR analysis was completed using the BLKOUT code (ref. 3). More recently, the event has been modeled as a loss of normal feedwater simultaneous with the station blackout using the LOFTRAN code. A reactor trip on S/G Low-Low Water Level typically occurs. Therefore, this event was analyzed during the Eagle 21 effort to form a consistent and comparable licensing basis. The analysis resulted in a reactor trip on S/G Low-Low Water Level using a safety analysis trip setpoint assumption of 0% NRS.

Excessive Heat Removal Due to Feedwater System Malfunctions (FSAR 15.2.10)

This event is analyzed at both full power and zero power to demonstrate that the reactivity excursion caused by the cooldown does not result in a violation of the DNB design basis. Neither of the cases is impacted by the Eagle 21 upgrades.

The reactivity excursion of the zero power analysis is bounded by the Rod Withdrawal From Subcritical analysis. The transient reactivity response will not change due to any of Eagle 21 upgrades. As was previously demonstrated, the Rod Withdrawal From Subcritical analysis is also unaffected by the Eagle 21 upgrades. Therefore, the zero power feedwater malfunction event is not impacted by the Eagle 21 upgrades.

The reactor trip credited in the full power analysis is the Steam Generator High-High Water Level. This case does not rely on the steamline break protection functions of steamline isolation and safety injection. Therefore, this case is not impacted by the Eagle 21 functional upgrades.

Excessive Load Increase (FSAR 15.2.11)

This event is defined as a rapid increase in the steam flow that causes a power mismatch between the reactor core power and the steam generator load demand. Overtemperature and Overpower Delta-T reactor trips are available in the event that they are needed to protect the reactor against this transient. However, as shown in the FSAR analysis, the event does not even generate a reactor trip. Therefore, RTD Bypass Elimination does not impact the results of this event. The remaining Eagle 21 functional upgrades also do not impact the results because safety injection and steamline isolation are not modeled.

Accidental Depressurization of the Reactor Coolant System (FSAR 15.2.12)

An accidental depressurization of the reactor coolant system could occur as a result of an inadvertent opening of a pressurizer relief or safety valve. The Overtemperature ΔT reactor trip actuates in this event. Therefore, the event was analyzed in support of the RTD Bypass Elimination. The acceptance criteria were met with the increased RTD response time. Safety injection and steamline isolation were not modeled in this analysis.

Accidental Depressurization of the Main Steam System (FSAR 15.2.13)
Rupture of a Main Steam line (FSAR 15.4.2.1)

The increased steam flow caused by the inadvertent opening of a steam generator safety or relief valve (accidental depressurization) or a double ended rupture of the main steam pipe causes an increase in the heat extraction rate from the RCS, resulting in a reduction of primary coolant temperature and pressure. Steamline isolation and safety injection are actuated during these events. These events were recently analyzed in support of the Upper Head Injection (UHI) removal which is scheduled to occur at the same time as Eagle 21 installation. The New Steamline Break protection system was incorporated into that analysis. The remaining Eagle 21 functional upgrades do not impact the results of this event.

Spurious Operation of the Safety Injection System at Power (FSAR 15.2.14)

This event assumes that a safety injection signal inadvertently occurs. The FSAR section describes the actuation signals as those coming from old steamline break protection. The event is not impacted by the switch to new steamline break protection however, because the analysis does not assume one particular safety injection signal. It also does not assume coincident reactor trip or steamline isolation from the spurious signal. Therefore, the Eagle 21 functional upgrades have no impact on this event.

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Major Rupture of a Main Feedwater Pipe (FSAR 15.4.2.2)

This event is analyzed to demonstrate that adequate heat removal capability exists via the auxiliary feedwater system to remove core decay heat, stored energy and RCS pump heat following reactor trip. Typically, safety injection does not occur in feedline break analyses of plants with old steamline break protection because neither the Low Pressurizer Pressure or High Steamline Flow coincident with Lo-Lo Tavg or Low Steamline Pressure setpoints are reached. Although the actuation setpoint would most likely be reached, safety injection on High Staeamline Differential Pressure is not modeled. However, with new steamline break protection the Low Steamline Pressure safety injection actuation setpoint is reached. This event was analyzed to incorporate the effects of new steamline break protection. The assumed Steam Generator Low-Low Water Level reactor trip setpoint was 0% NRS. This is consistent with the calculation of the Steam Generator Low-Low Water Level reactor trip setpoints. All acceptance criteria for this event were met.

The feedline break event was analyzed at 10%, 30%, 40%, and 50% rated thermal power assuming both with and without offsite power to determine the impact of the trip time delay on a part-power feedline break event. The results of each case demonstrated that the acceptance criteria applicable to the feedline break event were met.

Rupture of a Control Rod Drive Mechanism Housing (RCCA Ejection) (FSAR 15.4.6)

This accident is the result of the assumed mechanical failure of a control rod mechanism pressure housing such that the RCS would eject the control rod and drive shaft to the fully withdrawn position. The upgrades for Eagle 21 do not impact the results of this analysis because in each case a reactor trip is actuated on High Neutron Flux. Safety injection and steamline isolation are not modeled for this event.

Steamline Break with Coincident Rod Withdrawal at Power

In September of 1979, IE Information Notice 79-22 was issued by the NRC addressing a potential unreviewed safety question resulting from Control and Protection Systems interaction. One of the postulated scenarios identified was the operation of the rod control system following an inside containment steamline rupture.

This analysis is simulated by modeling a steamline rupture and a coincident withdrawal of control bank D at full power conditions. A spectrum of steamline break sizes was analyzed to determine the limiting condition. The following reactor trip functions may actuate during this transient depending on the break size:

a. Overpower ΔT

b. A reactor trip is generated subsequent to safety injection system and steamline isolation actuation caused by Low Steamline Pressure.

This event was analyzed to support RTD Bypass Elimination due to the use of the Overpower ΔT reactor trip. For the cases which tripped on Overpower ΔT, acceptable results were obtained when the lead/lag compensator on measured ΔT was used with setpoints of 12/3 along with the Overpower ΔT setpoints currently specified in the Technical Specifications. This compensation provides for a faster reactor trip on both Overpower ΔT and Overtemperature ΔT. It should be noted that presently this lead/lag compensation is not included in the protection system at Sequoyah. The introduction of the Eagle 21 electronics incorporates this lead/lag into the Sequoyah protection system.

The change from Old to New Steamline Break protection was not a cause for reanalysis because Low Steamline Pressure is used for actuation of safety injection and steamline isolation in both steamline break protection systems.

Steamline Break Mass/Energy Release Outside Containment (WCAP-10961)

Steamline ruptures occurring outside the reactor containment structure may result in significant releases of high energy fluid to the structures surrounding the steam systems. The interrelationship between many of the factors which influence steamline break mass and energy releases makes determination of a single worst case difficult. For this reason, a number of cases were initially analyzed. Several cases (full power, upstram breaks ranging form 0.1 to 1.2 square feet, and downstream breaks ranging from 0.4 to 1.4 square feet) of this event credit a reactor trip on Overpower ΔT. In addition, old steamline break protection functions were used to actuate safety injection and steamline isolation. A representative sampling of the initial cases were analyzed to support both new steamline break protection and the RTD Bypass Elimination. The results of this analysis showed that these changes had an insignificant impact on the conclusions of the original analysis.

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Steamline Break Mass/Energy Release Inside Containment (FSAR 6.2)

In 1981, a complete reanalysis of this accident in order to justify removal of the Boron Injection Tank (BIT) was performed. Mass and energy releases from a steamline break inside containment are calculated for a break size of 1.4 square and 4.6 square feet at 0 % and 102 % of rated thermal power. In addition, the mass and energy relaeases from a variety of smaller split breaks are analyzed at 30 % and 100 % of rated thermal rated power.

The Steamline Break Mass/Energy Release Inside Containment analysis which was completed to support removal of the Boron Injection Tank was evaluated to insure that the protection system logic assumed in each of the cases is still available with the New Steamline Break protection system. In several cases, steamline isolation and safety injection were actuated by Low Steamline Pressure with Coincident High Steamline Flow.

The Low Steamline Pressure setpoint is contained in the New Steamline Break Protection without the coincidence logic. In the previous analyses, Low Steamline Pressure was the second signal to complete the coincidence logic and actuate safety injection and steamline isolation. Therefore, the signal would be generated with new steamline break protection at the same times it was generated with old steamline break protection. Neither the setpoint pressure or the lead/lag compensation on the signal will change as a result of the Eagle 21 digital protection system design. In the remaining cases, safety injection and steamline isolation were actuated on the High and High-High Containment pressure signals, respectively. Both of these signals are retained in the switch from old to new steamline break protection. This evaluation demonstrates that the change from Old to New Steamline Break protection as well as the remaining Eagle 21 functional upgrades does not impact this event.

CONCLUSIONS

Based on the previous discussion, the proposed protection system modifications are acceptable with respect to the non-LOCA safety analyses. A number of transients were analyzed specifically to support the functional upgrades which are being incorporated as part of the Eagle 21 digital protection system. The results of those analyses demonstrated that the acceptance criteria particular to each event were met. The remaining analyses were evaluated and it was demonstrated that the Eagle 21 functional upgrades have no impact on the current analyses.

TRIP TIME DELAY (TTD)/ATWS MITIGATING SYSTEM ACTUATION CIRCUITRY (AMSAC) INTERACTION

BACKGROUND

The Trip Time Delay (TTD) is a system of pre-determined programmed steam generator low-low level reactor trip and auxiliary feedwater delay times that are based upon (1) the prevailing power level at the time a low-low level trip setpoint is reached, and by (2) the number of steam generators that are affected. In the Sequoyah TTD design, the trip delay times are determined from two delay curves which are functions of power (below 50% rated thermal power). One trip delay curve applies to situations where a single steam generator is low and the other trip delay curve is for multiple steam generators with low inventories.

The design of the ATWS Mitigating System Actuation Circuitry (AMSAC), as described in WCAP-10858-P-A, provides an independent backup to the existing reactor protection system to initiate a turbine trip and actuates auxiliary feedwater flow in the event of an anticipated transient without a reactor trip while the power level is above 40 percent RTP. AMSAC is required by 10CFR50.62. In the Sequoyah units, AMSAC will trip the turbine and start the auxiliary feedwater systems if the water level in three or four steam generators drops below the AMSAC setpoint (which is set at 5% NRS below the steam generator low-low level reactor trip setpoint) and the power level is 40% RTP or greater. The AMSAC functions are delayed by 25 seconds.

In the 40% RTP to 50% RTP power range, if the water levels in three or four steam generators drop below the AMSAC setpoint, the AMSAC and TTD systems will both be actuated. Since, in this power range, the AMSAC delay time is shorter than the TTD delay time, the turbine will be tripped and the auxiliary feedwater system will be started by AMSAC before the reactor can be tripped by the TTD. Furthermore, since the Sequoyah units are equipped with the P-9 permissive, the turbine trip will not cause a reactor trip in this power range. Under these circumstances, if a trip is not demanded by another part of the Reactor Protection System (e.g., high pressurizer pressure or level), then the steam generator water low-low level trip will not occur until the TTD trip delay expires.

DISCUSSION

The intent of the AMSAC design was to develop a system which would not interfere with the reactor protection functions. However, it is acceptable from the non-LOCA safety analysis standpoint for the turbine trip to occur before a reactor trip.

If an AMSAC-initiated turbine trip should occur at a power level between 40% RTP and 50% RTP, then it can be assumed that the low-low level trip setpoint has already been reached and has (1) commenced the appropriate TTD trip delay or (2) has failed to activate the appropriate TTD and reactor trip. In the first possibility, the reactor will eventually be tripped and the resulting transient will meet Condition II acceptance criteria, since events similar to this one have been used to determine the TTD trip delay curves. In the second instance, there is an ATWS event in progress, and it is known, from generic analyses, that the AMSAC will mitigate the consequence such that overpressure limits will not be exceeded.

Under the conditions previously discussed, the TTD analysis basis could be affected by the AMSAC, since the AMSAC functions will be executed before the TTD delay can be completed. The AMSAC functions will be executed only when the water level drops below the AMSAC setpoint in three or four steam generators. The Loss of Normal Feedwater event was analyzed to determine the acceptable Trip Time Delays at power levels of interest (40%-50% RTP). These analyses did not assume a turbine trip actuated by the AMSAC steam generator low-low level setpoint. However, the analyses would not be impacted by the AMSAC-initiated turbine trip since, at these power levels, there is adequate steam relief capability via the steam generator safety valves, steam generator PORVs, and steam dumps to accomodate the load rejection. Therefore, there is no significant impact on the Trip Time Delay analyses due to the AMSAC-initiated turbine trip.

In addition, it is possible that the AMSAC, by tripping the turbine and starting auxiliary feedwater, can restore the water level in all steam generators before the TTD trip delay is completed, and thereby cause the trip signal to be cleared. In this sense, it can be said that the AMSAC effect upon the TTD may be to improve the performance of the TTD in reducing unnecessary feedwater-related reactor trips.

CONCLUSIONS

The potential exists for the AMSAC system to generate a turbine trip before the reactor protection system produces a reactor trip with the introduction of the TTD. The above discussion demonstrates that this interaction does not pose a safety concern. The results of the FSAR Chapter 15 non-LOCA safety analyses which have been completed in support of the Eagle 21 Digital Protection system and associated upgrades remain valid. The analyses used in the development of the TTD are also not impacted. Since the AMSAC delay time is shorter than the TTD delay time, the AMSAC design basis is not affected by the TTD. AMSAC will continue to provide protection against loss of feedwater ATWS events regardless of presence (or absence) of a TTD system.

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LOCA ACCIDENTS

BACKGROUND

As part of the elimination of the RTD (Resistance Temperature Detector) bypass piping, various other modifications are also being made to Sequoyah Units 1 and 2. These modifications are as follow:

1. elimination of the RTD (Resistance Temperature Detector) bypass piping
2. EAGLE-21 process protection system upgrade
3 new steamline break protection
4 elimination of low feedwater flow reactor trip with addition of MSS (median signal selector)
5 EAM (Environmental Allowance Modifier) installation
6. TTD (trip time delay)

This safety evaluation examines the following LOCA-related FSAR accident analyses of record for potential effects due to the proposed modifications: large break LOCA, small break LOCA, hot leg switchover to prevent potential boron precipitation, blowdown reactor vessel and loop forces, post-LOCA longterm core cooling subcriticality requirement, and rod ejection mass and energy release for dose calculations.

EVALUATION

RTD Bypass Elimination

The RTD bypass line is being removed from Sequoyah and is being replaced by three RTD's mounted in thermowells 120^{o} apart at the same location in the hot leg. Two RTD's will be placed in the cold leg at the RCP discharge.

The RTD's generate a delta-T and a Tave in each loop which are used for overpower and overtemperature delta-T protection, Lo Tave feedwater isolation, Lo-Lo Tave SI and steamline isolation, rod control, steam dump control, pressurizer level control, and RCS flow measurement. Of these functions, the only effect on the LOCA analyses would be RCS flow and Tave determination. However, the uncertainties associated with the RTD's ($\pm 1.2^{o}F$) are within current limits. As a result, the RCS inlet and outlet temperatures, thermal design flow rate, and steam generator performance data as used in the LOCA analyses will not be affected. Therefore, the LOCA-related accident analyses are unaffected by this modification.

EAGLE-21 Process Protection System

EAGLE-21 is a digital process protection system which replaces the current analog system. System response and setpoints will not be affected. Therefore, the LOCA-related accident analyses are unaffected by this modification.

New Steamline Break Protection

The new steamline break protection removes the high steamline flow, low-low average coolant temperature, and high steamline differential pressure protective functions for SI actuation. LOCA analyses assume reactor trip and SI action based on low pressurizer pressure or high containment pressure. Therefore, the LOCA-related accident analyses are unaffected by this modification.

Median Signal Detector (MSS)

Installation of an MSS for each steam generator will eliminate the need for the low feedwater flow reactor trip. LOCA analyses assume reactor trip and SI action based on low pressurizer pressure or high containment pressure. Therefore, the LOCA-related accident analyses are unaffected by this modification.

Environmental Allowance Modifier (EAM)

The EAM enables a higher adverse environment steam generator low-low level trip set point when an adverse containment condition (high containment pressure) is sensed. The elevated setpoint accounts for instrument uncertainties related to harsh environments. LOCA analyses assume reactor trip and SI action based on low pressurizer pressure or high containment pressure. Therefore, the LOCA-related accident analyses are unaffected by this modification.

Trip Time Delay (TTD)

The TTD system consists of programmed trip delay times as a function of power level at the time the low-low level steam generator level is reached. LOCA analyses assume reactor trip and SI action based on low pressurizer pressure or high containment pressure. Therefore, the LOCA-related accident analyses are unaffected by this modification.

CONCLUSIONS

The effect of the modifications described above have been evaluated with respect to the following LOCA-related accident analyses: large break LOCA, small break LOCA, hot leg switchover to prevent potential boron precipitation, blowdown reactor vessel and loop forces, post-LOCA longterm core cooling subcriticality requirement, and rod ejection mass and energy release for dose calculations. In all cases, it was shown that the proposed modifications did not result in exceeding any design or regulatory limit.

STEAM GENERATOR TUBE RUPTURE (SGTR)

EVALUATION

The FSAR analysis for a steam generator tube rupture (SGTR) accident for Sequoyah is described in Section 15.4.3 of the FSAR. The FSAR analysis was performed to demonstrate that the radiological consequences resulting from a SGTR accident are below the 10CFR100 limits. The major factors that affect the radiological doses of an SGTR event are the amount of fuel failure, the amount of primary coolant transferred to the secondary side of the ruptured steam generator through the ruptured tube and the amount of steam released to the atmosphere from the ruptured steam generator. The impact of the Eagle 21 Process Equipment Upgrade System (Eagle 21 upgrade from analog to digital, NSLB, MSS, EAM & TTD) and a two (2) second increase in RTD response time due to RTD Bypass Elimination with respect to these major factors has been determined.

The amount of fuel failure assumed for the SGTR event is independent of the aforementioned changes and will not change. The Eagle 21 Process Equipment Upgrade System will have no impact on the assumptions or parameters which affect the primary to secondary break flow and the atmospheric steam release. However, the RTD Bypass Elimination could effect the primary to secondary break flow and atmospheric steam release via a change to the reactor trip time. For a SGTR, the loss of reactor coolant inventory due to the primary to secondary break flow results in a decrease in the pressurizer pressure and a subsequent automatic reactor trip signal. The two (2) second increase in RTD response time is via the lag compensation for the measured T-avg and Delta-T ; additionally, reactor trip on overtemperature delta-T or overpower delta-T for a SGTR event is primarily due to the depressurization transient while temperature response is a secondary effect. Due to these factors, the two (2) second increase in RTD response time will result in an insignificant change in the time of reactor trip signal, and therefore an insignificant change in the primary to secondary break flow and the atmospheric steam release via the ruptured steam generator.

Based on the above, it is concluded that RTD Bypass Elimination and the Eagle 21 Process Equipment Upgrade System will not change the conclusions reported for the Sequoyah FSAR SGTR analysis.

CONTAINMENT INTEGRITY ACCIDENTS

Containment Integrity Response To MSLB Outside Containment

The mass / energy releases resulting from the RTDBE and NSLB modifications showed an insignificant impact on the conclusions of the accident analysis. The modifications involved for the Eagle Upgrade of analog to digital, MSS, EAM and TTD, do not adversely impact the mass / enrgy releases of this accident.

Containment Integrity Response To LOCA And MSLB Inside Containment

The modifications involved for the RTDBE, Eagle Upgrade of analog to digital, NSLB, MSS, EAM and TTD, do not adversely impact the mass / enrgy releases of these accidents.

HARDWARE SYSTEM EVALUATIONS (BASED UPON APPLICABLE FCNs)

RTD BYPASS ELIMINATION (RTD HARDWARE MODIFICATION
(FCNs TENO-40567B and TVAO-40568B)
(Previously SECL-89-863-P-6, TVA-89-945)

Background

The presently installed RTD bypass system is to be replaced with fast acting narrow range RTD thermowells. This change requires modifications to the hot leg scoops, the crossover leg bypass return piping, the cold leg piping and the cold leg bypass manifold connection. Each of these modifications is evaluated below.

Discussion

The RTD bypass piping that connects to the original three scoops in each hot leg, which feed the bypass manifold, and the bypass manifold must be removed and the scoops modified to accept three fast response RTD thermowells. An appropriately sized hole will be machined through the tip of the scoop to provide the proper flow path. A thermowell design will be used such that the tip of the thermowell will be positioned to provide an average temperature reading. The thermowell will be fabricated in accordance with Section III Class 1 of the ASME Code. The installation of the thermowell into the scoop will be performed using a GTAW root pass and a GTAW or SMAW finish pass. The welding will be examined by penetrant test (PT) per the ASME Code Section III. Prior to welding, the surface of the scoop onto which welding will be performed will also be examined by PT per Section III.

The cold leg RTD bypass nozzle must also be modified to accept a fast response thermowell and the bypass line removed. Additionally, a spare fast response thermowell will be added to the cold leg in the length between the reactor coolant pump discharge and the accumulator nozzle. This necessitates the creation of a new penetration into the piping. The boss for the new connection will be root welded by GTAW. Finish welding can be either GTAW or SMAW. Weld inspection by PT will be performed after the root pass and the final pass. The thermowells will extend into the flow stream.

The root weld joining the thermowells to the modified cold leg or bosses will be deposited with GTAW and the remainder of the weld may be deposited with GTAW or SMAW. Penetrants test will be performed in accordance with the ASME Code Section III. The thermowells and installation bosses will be fabricated in accordance with ASME Section III Class 1.

These two thermowells will be installed in the upper half of the piping. With the three thermowells in the hot leg and the two thermowells in the cold leg, a total of 20 thermowells will be utilized at each of the four-loop Sequoyah units and they will perform the same function as the original bypass T_{hot} and T_{cold} signals.

The crossover leg bypass return piping connection must be severed and the remaining pipe stub capped. The field machined surfaces are given a liquid penetrant exam per ASME Section III requirements. The cap design, including materials, will meet the pressure boundary criteria and ASME Section III Class 1. The cap will be root welded to the pipe stub by GTAW and fill welded by either GTAW or SMAW. Radiographs and penetrant tests will be performed on the completed weld per ASME Section III. Machining of the bypass return piping, as well as any machining performed during modification of the penetrations in the hot and cold legs, shall be performed such as to minimize debris escaping into the reactor coolant system.

The integrity of the reactor coolant piping as a pressure boundary component, is maintained by adhering to the applicable ASME Code Sections and Nuclear Regulatory Commission General Design Criteria. The pressure retaining capability and fracture prevention characteristics of the piping is not compromised by these modifications. Therefore, no unresolved safety issue is involved as defined in 10 CFR 50.59.

ELIMINATION OF LOW FEEDWATER FLOW REACTOR TRIP FOR THE MEDIAN SIGNAL SELECTOR (TENO-40569, 40569A, TENO-40570, 40570A, TVAO-40570, 40570A, and TVAO-40571, 40571A) (Previously SECL-89-863-P-2, TVA-89-902, 965)

Background

Currently there are two reactor trip channels associated with the Low Steam Generator Water Level Protection System. These are the Low-Low Steam Generator Water Level Trip channel and the Low Feedwater Flow Trip channel. Both trip functions were required due to the configuration of the control and protection channels. One of the three water level measurement channels is also utilized for control purposes. Thus, proper operation of the Feedwater Control System is dependent on the integrity of this channel. This characteristic permits certain failures which may occur in the Reactor Protection System to negate a particular channel of a protective function, and simultaneously cause undesirable control system action that requires subsequent protective action from the failed safety function. For such a scenario, IEEE-279, Section 4.7.3 imposes the consideration of an additional random failure in the Reactor Protection System. The logic is that the initial protection system failure is considered the initiating event for the transient, and, therefore, does not constitute the "single failure" IEEE-279 imposes on the protection system. With this additional limiting single failure of one of the remaining steam generator low-low level channels, only one channel is operational. This is insufficient to satisfy the 2/3 logic implemented in the low-low steam generator level reactor trip function. The Low Feedwater Flow Reactor Trip was used to satisfy this IEEE-279 requirement in conjunction with the Low-Low Steam Generator Water Level Trip. The Low Feedwater Flow Trip is not required for IEEE-279 if you do not need to consider a second random failure by functionally isolating the Feedwater Control System from the Reactor Protection System. This is accomplished by implementing a Median Signal Selector in the Feedwater Control System.

Discussion

With the implementation of a Median Signal Selector, the control and protection systems would now be functionally isolated. With a signal selector, all three level measurement channels are input to the control system and compared in the signal selector. The signal selector selects the median signal for transmission to the control system, and control system action is then based on this signal. By rejecting the high and low signals, the control system is prevented from acting on any single, failed protection system instrument channel.

Since no adverse control system action may result from a single, failed protection instrument channel, a second random protection system failure (as would otherwise be required by IEEE-279) need not be considered. Thus, since IEEE Std. 279-1971, control and protection interaction criteria is satisfied with implementation of the MSS and the accident analyses for steam generator low level protection is satisfied through the steam generator low-low level reactor trip, there is no requirement to maintain the low feedwater flow reactor trip. When the Low Feedwater Flow Reactor Trip is deleted, one other possible situation must be addressed. The occurrence of a random failure in the protection channel negating the protective action and propagating down to the Median Signal Selector and limiting the ability of the Median Signal Selector to perform its function is the other possible situation. As per internal Westinghouse letter No. I&CP/PSE(89)-148, dated September 29, 1989, the maximum current that could be seen by the MSS under any fault condition that would affect a protection channel is 50mA and this is within the normal signal range of the MSS. Random failures of Median Signal Selector are addressed in WCAP-12471.

Conclusion

Implementation of the Median Signal Selector prevents a random failure in the Reactor Protection Circuit that affects a particular channel of a protection function from simultaneously causing an undesirable control action, and,therefore, the Low Feedwater Flow Reactor Trip is not needed. The channels in the 2/3 logic for the Low-Low Water Level Reactor Trip will be sufficient to provide protection as required for the Low Water Level Protection System. It is concluded that the removal of the Low Feedwater Flow Reactor Trip will not adversely affect the safety of the plant. All performance conclusions are based on the assumption that all visual checks and point-to-point wiring continuity checks demonstrate that the resultant system performs properly.

MEDIAN SIGNAL SELECTOR (MSS) S/G LEVEL CONTROL - MEDIAN SIGNAL RECORDING
(FCNs TENO-40572, 40572A and TVAO-40572, 40572A)
(Previously SECL-89-863-P-3, TVA-89-914)

Background

The Steam Generator Level Channel presently is recorded from the signal from one protection channel. With the implementation of the Median Signal Selector this recorded channel will be from the median signal of the protection channels.

Discussion

Since the median signal is being used for the control functions of the Steam Generator Level, it should be the signal that is recorded. This change will not alter the signal going to the control system in any way.

Conclusion

Since this FCN does not change the safety functions of any equipment, it is concluded that this change will not adversely affect the safety of the plant.

NEW STEAMLINE BREAK PROTECTION SYSTEM
(FCNs TENO-40570, 40570A and TVAO-40571, 40571A)
(Previously SECL-89-863-P-7, TVA-89-965)

Background

These FCNs and FCNs TENO-40569A and TVAO-40570A provide modification instructions for: 1) deletion of the low feedwater flow reactor trip, 2) implementation of a median signal selector, and 3) implementation of the New Steamline Break Protection System. Deletion of the low feedwater flow reactor trip function only removes wires between the output of the low feedwater flow reactor trip and the input to the undervoltage card. In addition, existing universal card assemblies are removed from the SSPS cabinets. The universal card assemblies are not needed since fewer logic functions are required to implement the New Steamline Break Proterction System.

The Steam Generator Protection System is being upgraded at Sequoyah Units 1 and 2. Both the existing and New Steam Generator Protection Systems have two primary functions: 1) prevent loss of heat sink for the reactor, and 2) protect the turbine and steam piping from excessive moisture carry-over. The loss of heat sink for the reactor is prevented by a low-low steam generator water level reactor trip. The turbine and steam piping are protected from excessive moisture carry-over by the high-high steam generator water level turbine trip and feedwater isolation which limits the water level in the steam generators.

The primary functions of the New Steamline Break Protection System functions are to: 1) isolate non-ruptured steamlines following a secondary high energy line rupture and 2) inject borated water into the reactor coolant system.

The New Steamline Break Protection System involves replacement of the existing steamline break protection system in the Process Protection System with a new system called Eagle-21. In addition, rewiring in the SSPS cabinets is also required. The Eagle-21 Process Potection System being installed at Sequoyah Units 1 and 2 is a modular, microprocessor-based form, fit and function replacement for the existing analog electronics that make up the four redundant channel sets of the Foxboro process protection equipment. The New Steamline Break Protection System reduces the potential for spurious actuation of safety injection at low power due to steamline differential pressure. The New Steamline Break Protection System is currently in use as the standard system for later vintage Westinghouse plants.

Discussion

These FCNs provide detailed instructions to remove existing wiring and install new wiring in the SSPS cabirets. In addition, prior to energization, visual checks and point-to-point wiring continuity checks are recommended. These FCNs also provide that, after energization, performance of applicable functional checks be performed.

The wire modifications in the SSPS are needed in order to implement the New Steamline Protection System and receive the signals provided by the modified Process Protection System (Eagle-21). The Eagle-21 Process Protection System provides bistable output signals to the SSPS. Included in the Process Protection System outputs are bistable outputs to the SSPS for the New Steamline Break Protection System.

For the New Steamline Break Protection System, the high steamline flow, low-low average coolant temperature, and high steamline differential pressure functions are removed. The high steamline differential pressure function was a potential cause of spurious safety injection actuations.

For the New Steamline Break Protection System, initiation of a safety injection signal from the SSPS will result from low steamline pressure in two-out-of-three (2/3) bistable outputs per loop in any loop.

For the New Steamline Break Protection System, a steamline isolation signal from the SSPS is initiated by: 1) high steamline pressure rate in two-out-of-three (2/3) bistable outputs per loop in any loop (interlocked with P-11) and 2) low steamline pressure in two-out-of-three (2/3) bistable outputs per loop in any loop (interlocked with P-11).

These FCNs do not provide any new components in the SSPS, only removal of existing universal board assemblies and rewiring of the existing logic. Therefore, the original qualification for the SSPS provided in Reference 4 is still valid.

Conclusion

Implementation of these FCNs for the New Steamline Protection System in the SSPS is deemed to be acceptable at Sequoyah Units 1 and 2.

These FCNs do not provide any new components in the SSPS, only removal of existing universal board assemblies and rewiring of the existing logic. Therefore, the original qualification for the SSPS provided in Reference 4 is still valid.

STEAM GENERATOR WIDE RANGE LEVEL PAMS MODIFICATION
(FCNs TENO-40574, 40574A and TVAO-40574, 40574A)
(Previously SECL-89-863-P-8, TVA-89-966)

Background

Steam Generator wide range level instrumentation loops L-501, L-502, L-503, and L-504 are being modified with the addition of the Foxboro instumentation PAMS modification as described in these Field Change Notices. The analog signals used for control instumentation are fed from the protection rack equipped with Eagle 21 instrumentation. Computer pads for loops L-502 and L-503 are now installed in rack 26, being moved from rack 27. Computer pads for loops L-501 and L-504 are not changed. The Foxboro power supplies (LQ-501, LQ-502, LQ-503, and LQ-504) are no longer needed as the instrument loops are now being fed from power supplies in the protection racks. This addresses control grade equipment.

Discussion

All of the modifications described in these FCNs and the above are occuring on control grade equipment on the control side of the isolated protection system signal. This change does not affect any safety grade equipment in the protection system.

Conclusion

Since these FCNs do not modify any safety grade equipment or safety related signal, it is concluded that this change will not adversely affect the safety of the plant.

EAGLE 21 SYSTEM - HARDWARE MODIFICATION FOXBORO REMOVAL/EAGLE INSTALLATION
(FCN TVAO-40573)
(Previously SECL-89-863-P-5, TVA-89-953)

Background

The Sequoyah Unit 1 Foxboro H-Line analog process protection system is being upgraded with the Westinghouse Eagle 21 microprocessor based process protection upgrade system. This modification will involve removing all of the analog process modules and installing the Eagle 21 equipment in protection racks 1 through 13. All of the protection channels processed in these racks will be converted from analog to digital implementation. Also, Functional Upgrades for RTD Bypass Elimination, New Steamline Break Protection, Environmental Allowance Modifier, Trip Time Delay, and Elimination of the Low Feedwater Flow Reactor Trip are implemented as part of the Eagle 21 installation. This addresses an Instrumentation and Control system safety evaluation of Field Change Notice TVAO-40573 which provides instructions for the removal of the analog process equipment and installation of the Eagle 21 process protection system.

Discussion

The Eagle 21 process protection upgrade system has been designed and tested to satisfy all regulatory requirements specified by the following applicable criteria:

1. IEEE Std. 279-1971 "Criteria for Protection Systems for Nuclear Power Generating Stations"

2. IEEE Std. 323-1974 "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations"

3. IEEE Std. 338-1971 "IEEE Standard Criteria for the Periodic Testing of Nuclear Power Generating Station Safety Systems"

4. IEEE Std. 344-1975 "IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations"

5. IEEE Std. 352-1975 "IEEE Guide for General Principles of Reliability Analysis of Nuclear Power Generating Station Protection Systems

6. IEEE Std. 379-1977 "IEEE Standard Application of the Single Failure Criterion to Nuclear Power Generating Station Class 1E Systems"

7. IEEE Std. 384-1981 "IEEE Standard Criteria for Independence of Class 1E Equipment and Circuits

8. IEEE Std. 603-1980 "IEEE Standard Criteria for Safety Systems for Nuclear Power Generating Stations"

9. Regulatory Guide 1.153, December, 1985 "Criteria For Power, Instrumentation, and Control Portions of Safety Systems"

   - Regulatory Guide 1.153 endorses the guidance of IEEE Std. 603-1980.

10. ANSI/IEEE-ANS-7-4.3.2 1982 "Application Criteria for Programmable Digital Computer Systems in Safety Systems of Nuclear Power Generating Stations"

    - ANSI/IEEE-ANS-7-4.3.2 1982 - expands and amplifies the requirements of IEEE Std. 603-1980.

11. REGULATORY GUIDE 1.152, November 1985 "Criteria for Programmable Digital Computer System Software in Safety-Related Systems in Nuclear Plants"

    - Regulatory Guide 1.152 endorses the guidance of ANSI/IEEE-ANSI-7-4.3.2

12. REGULATORY GUIDE 1.47, May 1973 "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems"

The most significant Instrumentation and Control system regulatory criteria which were evaluated are summarized here below:

A. Single Failure Criterion

The Eagle-21 Process Protection System is designed to provide three or four instrumentation channels and outputs to two trip logic trains for each protective function. These redundant channels and trains are electrically isolated and physically separated. Thus, any single failure within a channel or train will not prevent a required protective system action. This implementation is consistent with the design of the analog system which is being replaced.

B. Channel Independence

Within the Eagle-21 Process Protection System, there are four separate and independent rack sets. Channels which provide signals for the same protective functions are each located in different rack sets ensuring that they will be independent and physically separated. Since all equipment within any rack is associated with a single Protection Channel Set (PCS), there is no requirement for separation of wiring and components within the rack. This implementation is consistent with the design of the analog system which is being replaced.

C. Equipment Qualification

The Westinghouse Equipment Qualification test program demonstrated the Eagle 21 Process Protection Equipment is capable of performing its designated safety related functions under all specified environmental and seismic conditions. The equipment qualification test plan and results consists of the following documents:

1. WCAP-8587 "Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety Related Electrical Equipment" which is a Westinghouse Class 3 (Non-Proprietary) report and represents the generic program parent document and describes the basis methodology for the Westinghouse equipment qualification program.

2. WCAP-8587, Supplement 1, EQDP-ESE-69 "Equipment Qualification Data Package" is a Westinghouse Class 3 (Non-Proprietary) report which presents a summary of the Eagle-21 test parameters, performance specifications, acceptance criteria, and test results.

3. WCAP-8687, Supplement 2-E69A and Supplement 2-E69B "Equipment Qualification Test Reports," which are Westinghouse Class 2 (Proprietary) reports and present a detailed description of the Eagle-21 test parameters, performance specifications, acceptance criteria, and test results.

D. Noise, Fault, Surge Withstand Capability, and Radio Frequency Interference.

The Noise, Fault, Surge Withstand Capability, and Radio Frequency Interference (RFI) test program demonstrated that the Eagle 21 process protection equipment is capable of performing its designated safety-related functions when subjected to these specified conditions. The test plan and results are documented in WCAP-11733 (Proprietary) and WCAP-11896 (Non-Proprietary).

E. Control and Protection System Interaction

The Eagle-21 Process Protection System functions completely independent from the control systems. Its operation in protecting the plant from unsafe conditions is not affected by any fault or malfunction in the control systems.

The transmission of signals from the Eagle-21 Process Protection System to the control systems is through isolation devices that are classified as part of the protection system. No credible fault at the output of an isolation device can prevent the associated Eagle-21 Protection System channel from meeting the minimum performance requirements specified in the design bases.

The same type of electrical isolation is also used to separate from the Eagle-21 Protection System, those signals (such as Steam Generator Narrow Range Level), which are required and used to control actual plant variables.

For this use, however, consideration must be given to possible protection channel failures that can both prevent a particular trip signal from that channel and cause the control system to drive the plant toward the unsafe condition for which the particular trip signal is needed. In each case where this is possible, either four protection channels have been provided and 2 out of 4 logic is used to ensure the plant remains fully protected even when degraded by a second random failure, or a diverse means for providing a reactor trip is available. The exception to this is the steam generator low-low water level protective function which relies on two out of three trip logic and a control system Median Signal Selector (MSS). The use of a control system MSS prevents any protection system failure from causing a control system reaction resulting in a need for subsequent protective action. The testability and reliability of the MSS are commensurate with that of the process protection system. Reference 5 provides further details on the MSS.

F. Capability for Test and Calibration

The Eagle-21 Process Protection System performs automatic surveillance testing of the digital process protection racks via a portable Man Machine Interface (MMI) test cart. The MMI test cart is connected to the process rack by inserting a connector into the process rack test panel. Using the MMI, the "Surveillance Test" option is then selected. Following instructions entered through the MMI, the rack test processor automatically performs the following operations:

1. Selection of the individual process channel to be tested.

2. Calibration of the test reference signals and verification of the tester time base.

3. Placement of the individual channel trip outputs in either "Channel Trip" or "Bypass" (password protected) mode.

    A. Bypass Mode -- disables the individual channel bistable trip circuitry which forces the associated logic input relays to remain in the non-tripped state until the "bypass" is removed.

    B. Channel Trip Mode -- Interrupts the individual channel bistable outputs to the logic circuitry to de-energize the associated logic input relay(s).

4. Activation of the test injection signal.

5. Performance of Analog to Digital (A/D) converter test, and engineering unit values conversion test.

6. Performance of bistable setpoint tests.

7. Performance of channel time response test.

8. Completion of test cycle and automatically remove "Channel Trips".

9. Verify calibration of the test injection signals.

10. Display of test results on the MMI screen.

Interruption of the bistable output to the logic circuitry for any reason (test, maintenance purposes, or removed from service) causes that portion of the logic to be actuated and accompanied by a channel trip alarm and channel status light in the control room. Status lights on the process rack test panel indicate when the associated bistables have tripped. Each channel is fully testable via the portable MMI test cart.

The Eagle-21 Process Protection System provides for continuous on-line self-calibration of analog input signals. The Digital Filter Processor (DFP) addresses high and low reference signals via a multiplexer circuit on each analog input channel. The Loop Calculation Processor (LCP) then compares the output of the DFP Analog to Digital (A/D) Converters to stored high and low reference values to determine if any errors have been introduced by analog signal processing and A/D conversion. If necessary, the LCP automatically compensates gain and offset coefficients to eliminate any errors that have been introduced.

G. Channel Bypass

The Eagle 21 Process Protection equipment is designed to permit any one channel to be maintained, and when required, tested during power operation without initiating a protective action at the systems level. During such operation, the process protection system continues to satisfy single failure criterion.

If an Eagle 21 protection channel has been bypassed for any purpose, a signal is provided to allow this condition to be continuously indicated in the control room. In addition, the Eagle 21 design has provided for administrative controls and multiple levels of security for bypassing a protection channel.

H. Access to Setpoint Adjustments

The Eagle 21 design has provided for administrative controls and multiple levels of security for access to setpoint and tuning constant adjustments.

As part of the Eagle 21 installation, the following functional upgrades are being incorporated:

A. Temperature Averaging System (TAS) for RTD Bypass Elimination
B. New Steamline Break Protection
C. Environmental Allowance Modifier
D. Trip Time Delay
E. Low Feedwater Flow Reactor Trip Removal (Via Median Signal Selector)

Conclusion

In Part B of page 1 of this Safety Evaluation, it was answered YES (Question 4.3) that the possibility of an accident which is different than already evaluated in the FSAR may be created and it was also answered YES (Question 4.6) that the possibility of a malfunction of equipment important to safety different than already evaluated in the FSAR may be created. These two questions were answered YES because the FSAR does not evaluate accidents which are the result of the process protection system failing to perform its intended function. Common mode failure of more than one process protection channel could lead to the possibility of an accident being created which is different than any already evaluated in the FSAR.

However, it is critical to note that Question 4.4 was answered NO. That is, the probability of a malfunction of equipment important to safety previously evaluated in the FSAR will NOT be increased. The following preventative measures have been taken to ensure that the integrity of the process protection system will not be subject to common mode failure:

Seismic and Environmental Qualification

Noise, Fault, Surge Withstand Capability, EMI and RFI Qualification Testing

Process Protection System Reliability Study

Verification and Validation (V&V) Program

- o Independent Design Specification Verification
- o Independent Software Verification
- o Independent System Validation Testing

Based on a review of the Eagle 21 system design and qualification test programs, it has been determined that all applicable Instrumentation and Control system regulatory criteria have been satisfied. Thus, it is concluded that installation of the Eagle 21 process protection upgrade is adequate for assuring that the Sequoyah Unit 1 protection system will reliably perform its safety related functions under all postulated conditions. This conclusion is contingent upon successful completion of the Eagle 21 Design, Verification and Validation program for Sequoyah Nuclear Power Plant Unit 1 in accordance with Reference 6.

MAR 21 1990

REFERENCES

1. T. W. T. Burnett, et. al., "LOFTRAN Code Description," WCAP-7907-P-A (Proprietary), WCAP-7907-A (Non-Proprietary), April 1984.

2. J. M. Geets. "MARVEL - A Digital Computer Code For Transient Analysis of a Multiloop PWR System," WCAP-7909, June 1972.

3. J. M. Geets, R. Salvatori, "Long Term Transient Analysis Program for PWRs (BLKOUT Code)," WCAP-7898, June 1972.

4. WCAP-7817, Supplement 3 Seismic Testing of Electrical and Control Equipment (W Solid State Protection System) (Low Seismic Plants)

5. WCAP-12374 "Topical Report - Eagle 21 Microprocessor Based Process Protection System"

6. Design Specification No. 408A47, Rev. 3 dated May 12, 1989 "Eagle 21 Replacement Hardware Design, Verification and Validation Plan"

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DCN M01443A
Page 101

Westinghouse Electric Corporation

Water Reactor Divisions

Nuclear Services Integration Division

Box 2728
Pittsburgh Pennsylvania 15230-2728

TVA-84-101

June 5, 1984

Mr. John A. Raulston
Chief Nuclear Engineer
Tennessee Valley Authority
400 West Summit Hill Drive
W10C126
Knoxville, TN 37902

REF: W Letter
TVA-84-076

Tennessee Valley Authority
Sequoyah Units 1 and 2
<u>SSPS Nuclear Safety Checklist</u>

Dear Mr. Raulston:

Attached for TVA's information and use is the Westinghouse Nuclear Safety Checklist covering the wiring change to relay test circuits for detecting switch failures. This checklist applies to Field Change Notices TVAO-40523 and TENO-40522 submitted on Westinghouse letter TVA-84-076.

Very truly yours,

WESTINGHOUSE ELECTRIC CORPORATION

MCHattrick, for
R. S. Howard, Manager
Customer Programs
Mid-South Area

MEH/jan

Attachment

cc: J. A. Raulston
C. C. Mason
R. U. Mathieson
S. A. Moser

0535e:12

DCN No. F028780 Page 62 of ___	DCN M01443A Page 102

Customer Reference No(s).
GENERIC

MAR 21 1990

Westinghouse Reference No(s).
(Change Control or RFQ as Applicabl
6156 , 10659

WESTINGHOUSE
NUCLEAR SAFETY EVALUATION CHECK LIST

(1) NUCLEAR PLANT(S) ALL W SSPS PLANTS – FOREIGN & DOMESTIC

(2) CHECK LIST APPLICABLE TO: WIRING CHANGE TO RELAY TEST CIRCUIT FOR DETECTING SWITCH FAILURES
(Subject of Change)

(3) The written safety evaluation of the revised procedure, design change or modificat required by 10CFR50.59 has been prepared to the extent required and is attached. If a safety evaluation is not required or is incomplete for any reason, explain on Page 3.

Parts A and B of this Safety Evaluation Check List are to be completed only on the basis of the safety evaluation performed.

CHECK LIST - PART A

(3.1) Yes ✓ No ___ A change to the plant as described in the FSAR?

(3.2) Yes ___ No ✓ A change to procedures as described in the FSAR?

(3.3) Yes ___ No ✓ A test or experiment not described in the FSAR?

(3.4) Yes ___ No ✓ A change to the plant technical specifications (Appendix A to the Operating License)?

(4) CHECK LIST - PART B (Justification for Part B answers must be included on Page 3.)

(4.1) Yes ___ No ✓ Will the probability of an accident previously evaluated in the FSAR be increased?

(4.2) Yes ___ No ✓ Will the consequences of an accident previously evaluated in the FSAR be increased?

(4.3) Yes ___ No ✓ May the possibility of an accident which is different than any already evaluated in the FSAR be created?

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DCN M01443A
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MAR 21 1990

Customer Reference No(s).

Westinghouse Reference No(s).
(Change Control or RFQ as Applicabl
6156, 10659

WESTINGHOUSE

NUCLEAR SAFETY EVALUATION CHECK LIST

PAGE 2 OF 3

(4.4) Yes ___ No ✓ Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?

(4.5) Yes ___ No ✓ Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?

(4.6) Yes ___ No ✓ May the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR be created?

(4.7) Yes ___ No ✓ Will the margin of safety as defined in the bases to any technical specification be reduced?

If the answers to any of the above questions are unknown, indicate under (5) REMARKS and explain on Page 3.

If the answer to any of the above questions in (4) cannot be answered in the negative, based on written safety evaluation, the change cannot be approved without an applicatior for license amendment submitted to NRC pursuant to 10CFR50.90.

(5) REMARKS:

(6) APPROVAL LADDER (Signatures):

(6.1) Prepared by (Nuclear Safety): F. W. Maiasco Date: 1-12-

(6.2) Coordinated with (Engineer(s): [signature] Date: 1/12/

(6.3) Coordinating Group Manager(s): [signature] Date: 1/16/

(6.4) Nuclear Safety Group Manager: [signature] Date: 1/17/9

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MAR 21 1990

Customer Reference No(s).

Westinghouse Reference No(s).
(Change Control or RFQ as Applicabl
6156, 10659

WESTINGHOUSE

NUCLEAR SAFETY EVALUATION CHECK LIST

PAGE 3 OF 3

The following summarizes the justification, based upon the written safety evaluation[1] for answers given in Part B of the Safety Evaluation Check List:

The wiring change described in the subject change controls assures that plant operators / technicians will be aware of SSPS relay test switch failure which could affect system operation.

I have evaluated the changes from a functional viewpoint and believe that the changes provide the desired result.

I have also concluded that the changes have no impact on the test circuits which confirm system integrity and operability.

Attachments 1 thru 3 show, briefly, the technical changes & circuit operation

[1] Reference to document(s) containing written safety evaluation: This document

PREPARED BY: F. W. Marasco DATE: 1-12-89

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Customer Reference No(s).

___

Westinghouse Reference No(s).
(Change Control or RFQ as Applicab
6156, 10659

WESTINGHOUSE
CHECK LIST
FOR FSAR UPDATE

(1) Unit: Generic

(2) FSAR Section: ___

(3) FSAR Page(s): ___

FSAR Table(s): ___

FSAR Figure(s): ___

(4) Reason for/Description of Change:

(5) Prepare Nuclear Safety Evaluation Check List and Attach.

See page 3

PREPARED BY: F. W. Mereno

DATE: 1-12-84

DCN No. F02878
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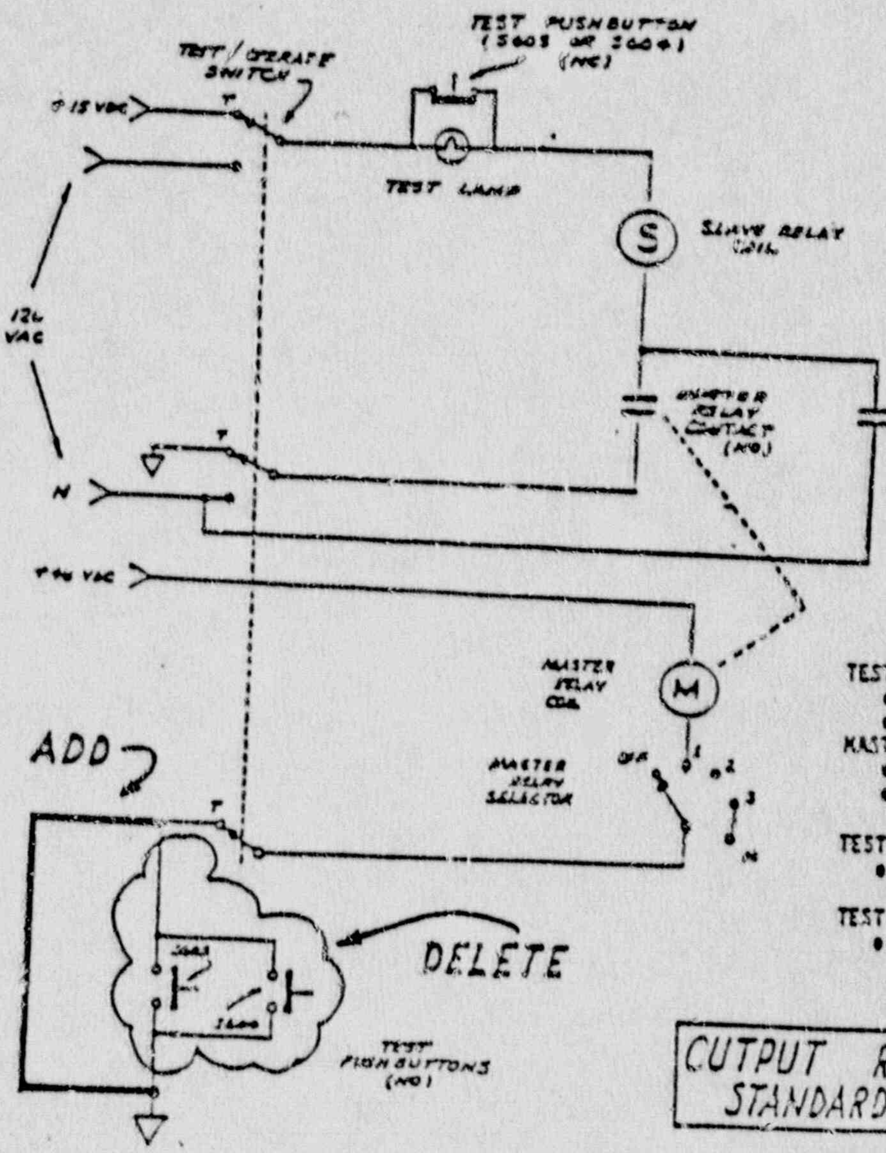
Page 106



NOTE: MODE SELECTOR SWITCH SHOWN IN TEST POSITION

OPERATION

TEST/OPERATE SWITCH TO TEST
- Slave 120 VAC reduced to 15 VDC
- 48 VDC master relay ground returned

MASTER RELAY SELECTOR SWITCH TO RELAY BEING TES
- Master relay energizes
- Master relay contacts close 15 VDC to slav (Slave not energized on 15 VDC)

TEST PUSHBUTTON DEPRESSED
- Test lamp ON confirms circuit and slave re coil continuity.

TEST PUSHBUTTON RELEASED
- Test lamp OFF confirms test lamp shunted

OUTPUT RELAY TEST CIRCUIT
STANDARD 2-TRAIN SSPS

ATTACHMENT 1

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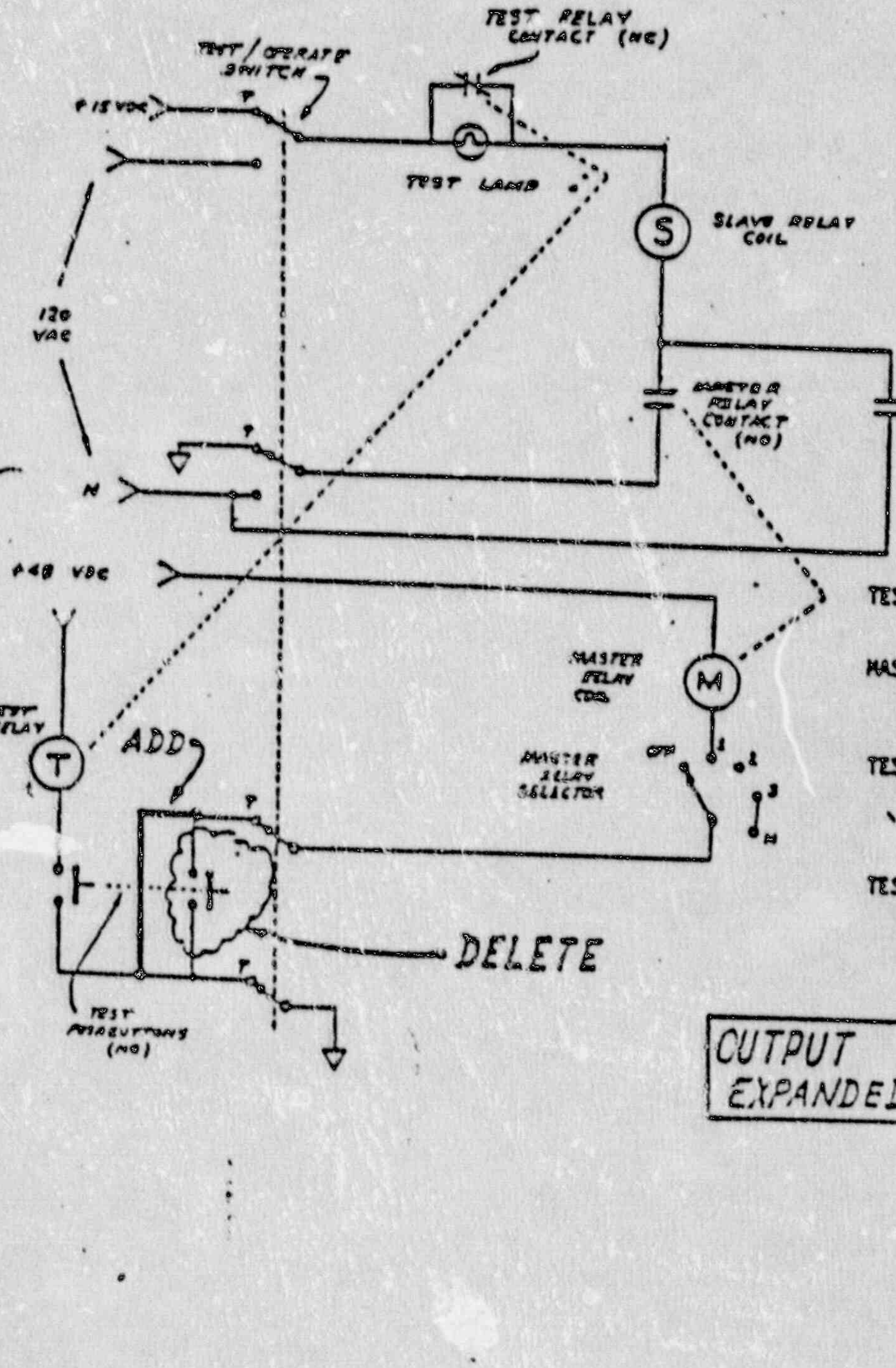
MAR 21 1990



NOTE: MODE SELECTOR SWITCH SHOWN IN TEST POSITION

OPERATION

TEST/OPERATE SWITCH TO TEST
- Slave 120 VAC reduced to 15 VDC
- 48 VDC master & slave relays ground return

MASTER RELAY SELECTOR SWITCH TO RELAY BEING TES
- Master relay energizes
- Master relay contacts close 15 VDC to slav
  (Slave not energized on 15 VDC)

TEST PUSHBUTTON DEPRESSED
- Test relay energizes
- Test relay contacts open
- Test lamp ON confirms circuit and slave re coil continuity

TEST PUSHBUTTON RELEASED
- Test relay de-energized
- Test relay contacts close
- Test lamp OFF confirms test lamp shunted

CUTPUT RELAY TEST CIRCUIT
EXPANDED 2-TRAIN SSPS

ATTACHMENT 2

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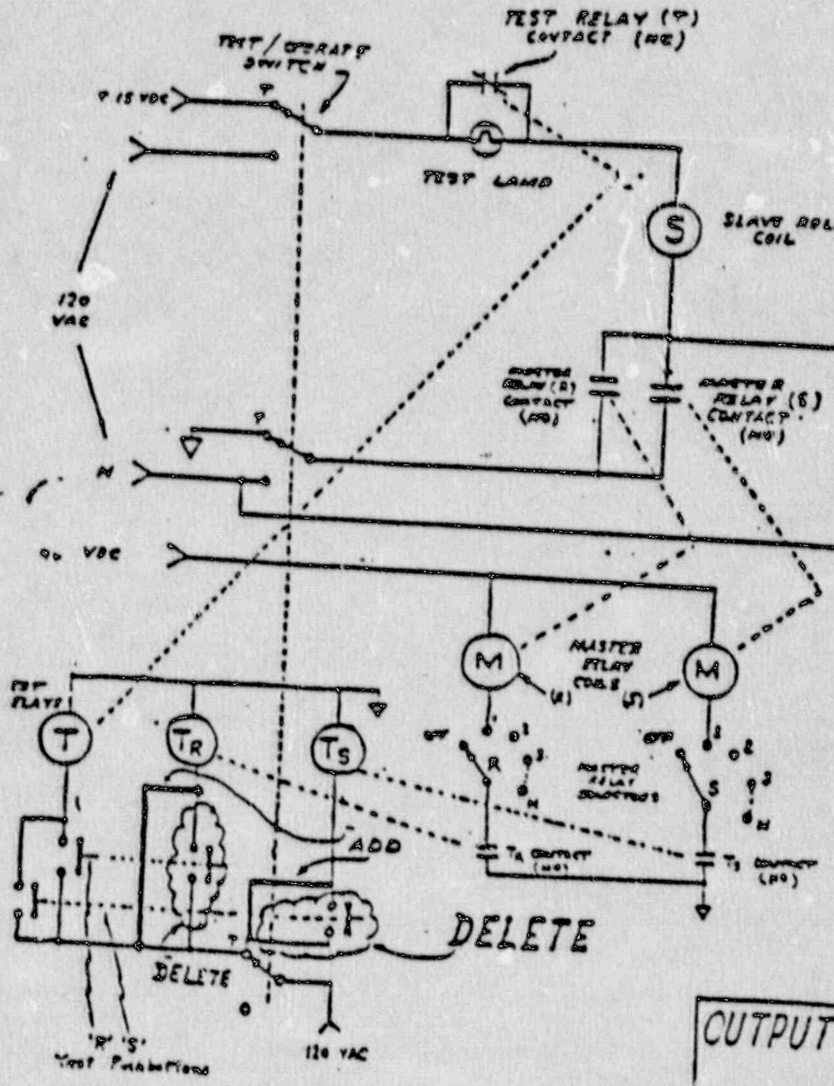
MAR 21 1990

NOTE: MODE SELECTOR SWITCH IS SHOWN IN TEST POSITION



OPERATION

TEST / OPERATE SWITCH TO TEST
- Slave 120 VAC reduced to 15 VDC
- Slave relays ground returned
- Test relays T_R & T_S energized

MASTER RELAY SELECTOR SWITCH TO RELAY BEING TESTED
- Master relay energizes
- Master relay contacts close 15 VDC to slave (Slave not energized on 15 VDC)

TEST PUSHBUTTON DEPRESSED
- Test relay energizes
- Test relay contacts open
- Test lamp ON confirms circuit and slave relay coil continuity

TEST PUSHBUTTON RELEASED
- Test relay de-energized
- Test relay contacts close
- Test lamp OFF confirms test lamp shunted

OUTPUT RELAY TEST CIRCUIT
3-TRAIN SSPS

ATTACHMENT 3

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MAR 21 1990

Westinghouse Electric Corporation

Water Reactor Divisions

QA Record

Energy Systems Service Divisions

Box 2728
Pittsburgh Pennsylvania 15230-2728

TVA-86-525

February 21, 1986

Reference:
CO-41053

FCNs:
TVAO-40556
TENO-40556

Mr. John A. Raulston
Chief Nuclear Engineer
Tennessee Valley Authority
400 West Summit Hill Drive
W10C126
Knoxville, TN 37902

Tennessee Valley Authority
Sequoyah Units 1 and 2
Nuclear Safety Checklist
SSPS LOSS OF AC DETECTION

B45 '86 0227 601

Dear Mr. Raulston:

Attached for TVA's information and use is Westinghouse Nuclear Safety Checklist SECL-85-405 applicable to the Sequoyah general warning alarm reactor trip for the loss of AC power to the SSPS output relays.

If there are any questions, please contact me.

Very truly yours,

WESTINGHOUSE ELECTRIC CORPORATION

M C Hartnett, for

L. L. Williams, Manager
ESSD Projects
Mid South Area

Reviewed by ___ NAR

cc: H. L. Abercrombie, 1L 1A
R. U. Mathieson, 1L 1A
I. R. Williamson, 1L

FEB 28 1986

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NEB-NAL				

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☐ FULL REVIEW
☐ REVIEW OF SPECIFIC ASPECTS
☐ NO REVIEW
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SECL 85-405

Customer Reference No(s).
CO-41053/84P65-835191

MAR 21 1990

Westinghouse Reference No(s).
(Change Control or RFQ as Applicable
OPAD - TVA - 85 - 002

SECL #405

WESTINGHOUSE
NUCLEAR SAFETY EVALUATION CHECK LIST

(1) NUCLEAR PLANT(S) Sequoyah 1 and 2

(2) CHECK LIST APPLICABLE TO: (Subject of Change) GENERAL WARNING ALARM REACTOR TRI

(3) The written safety evaluation of the revised procedure, design change or modificati required by 10CFR50.59 has been prepared to the extent required and is attached. If a safety evaluation is not required or is incomplete for any reason, explain on Page 3.

Parts A and B of this Safety Evaluation Check List are to be completed only on the basis of the safety evaluation performed.

CHECK LIST - PART A

(3.1) Yes ____ No ✓ A change to the plant as described in the FSAR?

(3.2) Yes ____ No ✓ A change to procedures as described in the FSAR?

(3.3) Yes ____ No ✓ A test or experiment not described in the FSAR?

(3.4) Yes ____ No ✓ A change to the plant technical specifications (Appendix A to the Operating License)?

(4) CHECK LIST - PART B (Justification for Part B answers must be included on Page 3.)

(4.1) Yes ____ No ✓ Will the probability of an accident previously evaluated in the FSAR be increased?

(4.2) Yes ____ No ✓ Will the consequences of an accident previously evaluated in the FSAR be increased?

(4.3) Yes ____ No ✓ May the possibility of an accident which is different than any already evaluated in the FSAR be created?

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MAR 21 1990

SECL 405

Customer Reference No(s).

Westinghouse Reference No(s).
(Change Control or RFQ as Applicable)
OPAD - TVA - 85 - 002

WESTINGHOUSE

NUCLEAR SAFETY EVALUATION CHECK LIST

(4.4) Yes ____ No ✓ Will the probability of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?

(4.5) Yes ____ No ✓ Will the consequences of a malfunction of equipment important to safety previously evaluated in the FSAR be increased?

(4.6) Yes ____ No ✓ May the possibility of a malfunction of equipment important to safety different than any already evaluated in the FSAR be created?

(4.7) Yes ____ No ✓ Will the margin of safety as defined in the bases to any technical specification be reduced?

If the answers to any of the above questions are unknown, indicate under (5) REMARKS and explain on Page 3.

If the answer to any of the above questions in (4) cannot be answered in the negative, based on written safety evaluation, the change cannot be approved without an application for license amendment submitted to NRC pursuant to 10CFR50.90.

(5) REMARKS:

(6) APPROVAL LADDER (Signatures):

(6.1) Prepared by (Nuclear Safety): F.W. Marasco [signature] Date: 9-13-85

(6.2) Coordinated with (Engineer(s): J.M. McNamara [signature] Date: 9/16/85

(6.3) Coordinating Group Manager(s): D.N. Katz [signature] Date: 9/16/85

(6.4) Nuclear Safety Group Manager: C.G. Draughon [signature] Date: 9/16/85

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Page 112

MAR 21 1990

SECL 405

Customer Reference No(s).

Westinghouse Reference No(s).
(Change Control or RFQ as Applicable)
OPAD - TVA - 85 - 002

WESTINGHOUSE
NUCLEAR SAFETY EVALUATION CHECK LIST
PAGE 3 OF 3

The following summarizes the justification, based upon the written safety evaluation(1), for answers given in Part B of the Safety Evaluation Check List:

Loss of AC power to the SSPS output relays is added as a trouble condition to the General Warning Alarm Reactor Trip System. The change is not related to circuitry relied on to automatically initiate reactor trip or engineered safeguards and cannot prevent or degrade their functional performance.

(1) Reference to document(s) containing written safety evaluation: SEE ABOVE EVALUATION STATEMENT

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Customer Reference No(s).

Westinghouse Reference No(s).
(Change Control or RFQ as Applicable
OPAD - TVA - 85 - 002

SECL 405

WESTINGHOUSE
CHECK LIST
FOR FSAR UPDATE

(1) Unit: Sequoyah 1 and 2 FSAR Table(s): ____

(2) FSAR Section: CH. 7 FSAR Figure(s): ____

(3) FSAR Page(s): ____

(4) Reason for/Description of Change:

No Impact

(5) Prepare Nuclear Safety Evaluation Check List and Attach.

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