Attachment 6 ~~contains Proprietary Information. Withhold From Public Disclosure Under 10 CFR 2.390. When separated from Attachment 6, this document is decontrolled.~~

PSEG Nuclear LLC
P.O. Box 236, Hancocks Bridge, NJ 08038-0236

PSEG
Nuclear LLC

MAR 27 2015

10 CFR 50.90

LR-N15-0020
LAR S15-01

U.S. Nuclear Regulatory Commission
ATTN: Document Control Desk
Washington, D.C. 20555-0001

Salem Nuclear Generating Station Units 1 and 2
Renewed Facility Operating License Nos. DPR-70 and 75
NRC Docket Nos. 50-272 and 50-311

Subject: **License Amendment Request to Revise Technical Specification 3/4.3.1, Reactor Trip System Instrumentation**

In accordance with 10 CFR 50.90, PSEG Nuclear LLC (PSEG) hereby requests an amendment to Renewed Facility Operating License Nos. DPR-70 and 75 for Salem Nuclear Generating Station Units 1 and 2. In accordance with 10 CFR 50.91(b)(1), a copy of this request for amendment has been sent to the State of New Jersey.

The proposed changes revise Technical Specification (TS) 3/4.3.1,"Reactor Trip System Instrumentation," Table 3.3-1, Action 2 and establish two new action notes for the power range nuclear instrumentation. These changes support the installation and use of bypass test capability for the power range nuclear instrumentation.

There are no regulatory commitments contained in this letter.

Attachment 1 provides an evaluation supporting the proposed changes.
Attachment 2 contains marked-up TS pages to indicate the proposed changes.
Attachment 3 contains the proposed changes to the TS Bases for information only.
Attachment 4 provides an affidavit for withholding signed by Westinghouse, the owner of the proprietary information provided in Attachment 6.
Attachment 5 provides Westinghouse report WCAP-17947-NP, "Power Range Nuclear Instrumentation System Bypass Test Instrumentation for Salem Units 1 and 2," March 2015 (Non-Proprietary)
Attachment 6 provides Westinghouse report WCAP-17947-P, "Power Range Nuclear Instrumentation System Bypass Test Instrumentation for Salem Units 1 and 2," March 2015 (Proprietary)

Attachment 6 contains proprietary information as defined by 10 CFR 2.390. Westinghouse Electric Company LLC, as the owner of the proprietary information, has executed the Attachment 4 affidavit identifying that the proprietary information has been handled and classified as proprietary, is customarily held in confidence, and has been withheld from public disclosure. Westinghouse requests that the proprietary information is Attachment 6 be withheld from public disclosure, in accordance with the requirements of 10 CFR 2.390(a)(4).

MAR 27 2015

PSEG requests NRC approval of the proposed License Amendment by March 31, 2016, to support implementation of the power range bypass test instrumentation modification for Salem Unit 1 during the spring 2016 refueling outage (1R24) and Salem Unit 2 during the spring 2017 refueling outage (2R22).

These proposed changes have been reviewed by the Plant Operations Review Committee.

If you have any questions or require additional information, please contact Brian Thomas at (856) 339-2022.

I declare under penalty of perjury that the foregoing is true and correct.

Executed on MAR 27 2015
(Date)

Respectfully,

John F. Perry
Site Vice President
Salem Generating Station

Attachments:

1. License Amendment Request to revise Technical Specification 3/4.3.1, Reactor Trip System Instrumentation
2. Technical Specification Proposed Changes (mark-up pages)
3. Technical Specification Bases Proposed Changes (for information only)
4. Westinghouse Application for Withholding and Affidavit
5. Westinghouse WCAP-17947-P-NP (Non-Proprietary)
6. Westinghouse WCAP-17947-P (Proprietary)

cc: Mr. D. Dorman, Administrator, Region I, NRC
Ms. C. Sanders-Parker, Project Manager, NRC
NRC Senior Resident Inspector, Salem
Mr. P. Mulligan, Manager IV, NJBNE
Mr. L. Marabella, Corporate Commitment Tracking Coordinator
Mr. T. Cachaza, Salem Commitment Tracking Coordinator

SALEM GENERATING STATION
RENEWED FACILITY OPERATING LICENSE NOS. DPR-70 AND DPR-75
DOCKET NO. 50-272 AND 50-311

License Amendment Request to Revise Technical Specifications 3/4.3.1, Reactor Trip System Instrumentation

Table of Contents

1.0 DESCRIPTION

The proposed changes to Technical Specification (TS) 3/4.3.1 "Reactor Trip Instrumentation" are needed to support the installation and use of bypass test capability for the power range (PR) nuclear instrumentation. Testing the PR nuclear instrumentation channels in bypass would reduce the likelihood of reactor trips due to human error, channel failure, or spurious transient in a redundant channel; thereby increasing plant availability while still ensuring that the PR nuclear instrumentation channels are capable of performing their intended plant protection function. Westinghouse Electric Company LLC (Westinghouse) Reports WCAP-17947-NP (Attachment 5) and WCAP-17947-P (Attachment 6) provide the Salem plant-specific basis for testing the PR nuclear instrumentation in bypass.

2.0 PROPOSED CHANGE

The proposed changes are described below and indicated on the marked-up TS pages provided in Attachment 2 of this submittal. Proposed changes to the TS Bases are provided in Attachment 3 for information only. Changes to the affected TS Bases pages will be incorporated per TS 6.17 (Unit 1) and TS 6.16 (Unit 2), "Technical Specifications (TS) Bases Control Program."

1. TS 3/4.3.1, Table 3.3-1, Action 2.b is revised to read:

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

2. TS 3/4.3.1, Table 4.3-1

   Add Note 17 reference to Functional Unit 2, CHANNEL CALIBRATION

   Add Note 18 reference to Functional Unit 2, CHANNEL FUNCTIONAL TEST

   Add Note 18 reference to Functional Unit 3, CHANNEL FUNCTIONAL TEST

3. TS 3/4.3.1, Table 4.3-1 (continued) NOTATION

   Add Note 17 – In MODES 1 and 2, the SSPS input relays are excluded from this Surveillance when the installed bypass test capability is used to perform this Surveillance.

   Add Note 18 – The SSPS input relays are excluded from this Surveillance when the installed bypass test capability is used to perform this Surveillance.

3.0 BACKGROUND

Salem UFSAR Section 7.2 states that the Reactor Trip System (RTS) consists of the Nuclear Instrumentation System (NIS), process control system, and the Solid State Protection System (SSPS). The NIS power range channels use a two-out-of-four coincidence logic from redundant channels to initiate protective actions. Within this system, analog channel comparators are typically placed in the tripped condition for channel testing. This situation essentially changes the normal two-out-of-four coincidence trip logic to a one-out-of-three trip logic. In this condition, a human error, channel failure, or spurious transient in a redundant channel could result in a reactor trip. With the implementation of PR nuclear instrumentation channel testing in bypass,

the spurious reactor trip is avoided because the partial trip condition that would have been present is eliminated. The coincidence trip logic becomes two-out-of-three from the channels not in test, thereby maintaining the requirement for two channels to actuate the protective function. A decrease in reactor trips reduces challenges to the RTS, avoids transients associated with reactor trips, and increases plant availability.

4.0 TECHNICAL ANALYSIS

Operating plants have experienced many inadvertent reactor trips and safeguards actuations during the performance of instrumentation surveillances, causing unnecessary transients and challenges to safety systems. In the early 1980s, in response to growing concern regarding the impact of TS surveillance testing and maintenance activities on plant operations, particularly as related to instrumentation systems, the Pressurized Water Reactor Owners Group (PWROG) (formerly the Westinghouse Owners Group) initiated a program to justify extending the RTS and Engineered Safety Features Actuation System (ESFAS) bypass test times, completion times, and surveillance frequencies, to provide additional time to perform surveillance and maintenance activities. WCAP-10271-P-A and Supplements 1 and 2 (References 1-4) justified extending the RTS and ESFAS bypass test times, completion times, and surveillance frequencies. One of the provisions discussed was to allow routine surveillance testing of the RTS and ESFAS channels in a bypassed condition rather than a tripped condition. The NRC Safety Evaluation Reports (SERs) for WCAP-10271-P-A and Supplements 1 and 2 (References 5-7) stated that using temporary jumpers or lifting leads was not an acceptable method of performing a channel bypass during routine surveillance testing.

The proposed change does not modify trip setpoints, surveillance frequencies, or channel responses. Hardware modifications will be made so that testing in bypass can be accomplished without lifting leads or installing temporary jumpers. This meets the conditions specified by the NRC in SERs issued during the review of WCAP-10271-P-A and its supplements. The impact of testing in bypass upon reactor safety was previously evaluated by the NRC during their review of WCAP-10271-P-A, and determined to be acceptable.

With the proposed change, the plant would be able to perform routine PR nuclear instrumentation surveillance testing with a channel in bypass instead of placing the analog channel comparators in a tripped condition. However, the SSPS input relays must be included in the Channel Calibration at least once every 18 months. This requirement is stated in the revised TS Bases.

The existing TS allows an inoperable channel to be placed in bypass for up to 4 hours to allow testing of other channels; but the analog channel comparators are currently placed in the tripped state for channel testing. With the new bypass test capability, a channel may be bypassed for surveillance testing with an inoperable channel in the tripped state (two channels are required for the trip function). The ability to place a channel in trip will still exist with the new hardware installation; therefore, placing an inoperable channel in the tripped condition is not affected.

A bypass panel will be installed in each power range NIS rack to provide a second source of 118 VAC power in place of a bistable function output. When a bistable is bypassed, the bypass test panel provides the 118 VAC signal to the SSPS input relays and then disconnects the bistable output. A make-before-break scheme is used to prevent power interruption. Status of the bypassed condition will be provided both in the control room and locally.

The bypass panel design has considered fault conditions, failure detection, reliability, and equipment qualification (refer to WCAP-17947-P, Sections 3.2, 3.3, 3.5, and 3.8 (Attachment 6)). Hardware changes to facilitate testing in bypass will be implemented per 10 CFR 50.59.

Administrative controls are used to prevent the simultaneous bypassing of more than one redundant protection set at any one time, and to restore the system to normal operation (refer to WCAP-17947-P, Section 3.4 (Attachment 6)). These administrative controls are described below:

1. Each bypass panel is enclosed in a NIS rack. To access the bypass panel, the rack door would have to be opened. These doors are locked with the keys under Operations control.

2. The NIS bypass panels have keylock switches that require a specific key to move the keylock switch to the bypass position. When the keylock switch is moved to the bypass position, a control room annunciator actuates. This alerts the operator to the specific bypass panel that has been placed in bypass. The keys are kept under Operations control and are unit specific, so the Unit 1 key cannot be used on Unit 2 and vice versa. This prevents using two keys to bypass two channels on one unit at the same time.

3. There is local indication (LED) on the bypass panel when an individual channel has been placed in the bypass condition (i.e., bypass toggle switch is placed in the bypass position). The technician is aware of channels in bypass without relying on remote (control room) indication.

4. Surveillance procedures specify that they may be performed on only one channel, and its associated protection set, at a time. Verifications are also conducted per surveillance procedures.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

In accordance with 10 CFR 50.90, PSEG Nuclear LLC (PSEG) hereby requests an amendment to Renewed Facility Operating License Nos. DPR-70 and 75 for Salem Nuclear Generating Station Units 1 and 2.

The proposed changes to Technical Specification (TS) 3/4.3.1 "Reactor Trip Instrumentation" are needed to support the installation and use of bypass test capability for the power range (PR) nuclear instrumentation. Testing the PR nuclear instrumentation channels in bypass promotes improved maintenance practices that could potentially result in a reduction in the number of reactor trips due to human error, channel failure, or spurious transient in a redundant channel. Westinghouse Electric Company LLC (Westinghouse) Report WCAP-17947-P, "Power Range Nuclear Instrumentation System Bypass Test Instrumentation for Salem Units 1 and 2," provides the plant specific basis for testing the PR nuclear instrumentation in bypass.

PSEG has evaluated whether or not a significant hazards consideration is involved with the proposed amendment(s) by focusing on the three standards set forth in 10 CFR 50.92, "Issuance of amendment," as discussed below:

1. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The power range (PR) nuclear instrumentation is not an accident initiator or precursor. The PR nuclear instrumentation provides indication and plant protection through a reactor trip. The reactor trip is part of the plant's accident mitigation response. With the existing system, analog channel comparators are placed in the tripped condition for channel testing. This changes the normal two-out-of-four coincidence trip logic to a one-out-of-three trip logic. In this condition, a human error, channel failure, or spurious transient in a redundant channel could result in a reactor trip. Testing the PR nuclear instrumentation channels in bypass eliminates the spurious reactor trip because the trip logic becomes two-out-of-three; thereby retaining the two channels required to actuate the protective function.

The proposed change does not affect how the Reactor Trip System (RTS) functions. The proposed change does not alter or prevent any structures, systems, or components from performing their intended design basis function(s) to mitigate the consequences of an initiating event within the applicable acceptance criteria. Surveillance testing in the bypass condition will not cause any design or analysis acceptance criteria to be exceeded.

PR channel testing in bypass does not affect the source term, containment isolation, or radiological release assumptions used in evaluating the radiological consequences of an accident previously evaluated. The proposed change does not increase the types or amounts of radioactive effluent that may be released offsite, or significantly increase individual or cumulative occupational/public radiation exposures. The change is consistent with safety analysis assumptions and resultant consequences. Implementation of the PR nuclear instrumentation bypass testing capability does not affect the integrity of the fission product barriers utilized for the mitigation of radiological dose consequences as a result of a design basis accident. The plant response as assumed in the safety analyses is unaffected by this change.

Therefore, the proposed changes do not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The manner in which the RTS provides plant protection is not changed. Surveillance testing in bypass does not affect accident initiation sequences or response scenarios as modeled in the safety analyses. The PR nuclear instrumentation will continue to have the same setpoints. No new failure modes are created for any plant equipment. The bypass test instrumentation has been designed and qualified to applicable regulatory and industry standards. Fault conditions, failure detection, reliability, and equipment qualification have been considered. Existing accident scenarios remain unchanged and new or different accident scenarios are not created. The types of accidents defined in

the Updated Final Safety Analysis Report (UFSAR) continue to represent the credible spectrum of events analyzed to determine safe plant operation.

Therefore, the proposed changes do not create the possibility of a new or different kind of accident from any previously evaluated.

3. Do the proposed changes involve a significant reduction in a margin of safety?

Response: No.

Safety analyses are not changed or modified as a result of the proposed Technical Specification (TS) changes to reflect installed PR nuclear instrumentation bypass test capability. The changes do not alter the manner in which the safety limits, limiting safety system setpoints, or limiting conditions for operation are determined. Margins associated with the applicable safety analyses acceptance criteria are unaffected. The current safety analyses remain bounding; their assumptions and conclusions are not affected by performing PR nuclear instrumentation surveillance testing in bypass. The safety systems credited in the safety analyses continue to remain available to perform their required mitigation functions. The impact of testing in bypass upon reactor safety was previously evaluated by the NRC during their review of WCAP-10271-P-A, and determined to be acceptable.

Therefore, the proposed changes do not involve a significant reduction in the margin of safety.

Based upon the above, PSEG Nuclear LLC concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92, and, accordingly, a finding of no significant hazards consideration is justified.

5.2 Applicable Regulatory Requirements and Criteria

Section 4.0 of Attachment 6 provides the regulatory requirements and criteria for bypass test instrumentation, including the General Design Criteria (GDC), Regulatory Guides (RG), and Institute of Electrical and Electronics Engineers (IEEE) Standards. Salem Nuclear Generating Station, Units 1 and 2 were designed in accordance with the Atomic Industrial Forum (AIF) GDC. In addition to the AIF GDC, they were designed to comply with Public Service Electric and Gas Company's understanding of the intent of the Atomic Energy Commission (AEC) proposed GDC published in July 1967. The applicable AEC proposed criteria, as documented in Salem UFSAR Section 3.1, were compared to the 10 CFR 50, Appendix A GDC discussed in Section 4.0 of Attachment 6, to ensure that the appropriate Salem design criteria were met.

In conclusion, based on the considerations discussed above, (1) there is a reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the NRC's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

5.3 Precedents

The NRC has approved similar license amendment changes as indicated below:

1. Braidwood Station, Units 1 and 2, Amendments 169 and 169, March 30, 2012 (TAC Nos. ME5836 and ME5837), Accession No. ML120660494

2. Byron Station, Units 1 and 2, Amendments 176 and 176, March 30, 2012, (TAC Nos. ME5838, and ME5839), Accession No. ML120660494

3. Comanche Peak, Units 1 and 2, Amendments 121 and 121, September 29, 2005 (TAC Nos. MC6482 and MC6483), Accession No. ML052380208

6.0 ENVIRONMENTAL CONSIDERATION

A review has determined that the proposed amendment would change a requirement with respect to installation or use of a facility component located within the restricted area, as defined in 10 CFR 20, or would change an inspection or surveillance requirement. However, the proposed amendment does not involve (i) a significant hazards consideration, (ii) a significant change in the types or significant increase in the amounts of any effluent that may be released offsite, or (iii) a significant increase in individual or cumulative occupational radiation exposure. Accordingly, the proposed amendment meets the eligibility criterion for categorical exclusion set forth in 10 CFR 51.22(c)(9). Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the proposed amendment.

7.0 REFERENCES

1. Westinghouse Electric Company LLC, WCAP-10271-P-A, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System,"

2. Westinghouse Electric Company LLC, WCAP-10271-P-A, Supplement 1, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System Supplement 1,"

3. Westinghouse Electric Company LLC, WCAP-10271-P-A, Supplement 2, "Evaluation of Surveillance Frequencies and Out of Service Times for the Engineered Safety Features Actuation System,"

4. Westinghouse Electric Company LLC, WCAP-10271-P-A, Supplement 2, Revision 1, "Evaluation of Surveillance Frequencies and Out of Service Times for the Engineered Safety Features Actuation System,"

5. Letter from C.O. Thomas (NRC) to J.J. Sheppard (WOG), NRC Safety Evaluation Report for WCAP-10271 and Supplement 1

6. Letter from Charles E. Rossi (NRC) to Roger A. Newton (WOG), NRC Safety Evaluation Report for WCAP-10271 Supplement 2 and Supplement 2, Revision 1

7. Letter from Charles E. Rossi (NRC) to Gerald T. Goering (WOG), NRC Supplemental Safety Evaluation Report for WCAP-10271 Supplement 2, Revision 1

TECHNICAL SPECIFICATION PROPOSED CHANGES

The following Technical Specifications for Renewed Facility Operating License DPR-70 are affected by this change request:

Technical Specification	Page
3/4.3.1, Table 3.3-1	3/4 3-5
3/4.3.1, Table 4.3-1	3/4 3-11
3/4.3.1, Table 4.3-1 (continued)	3/4 3-13

The following Technical Specifications for Renewed Facility Operating License DPR-75 are affected by this change request:

Technical Specification	Page
3/4.3.1, Table 3.3-1	3/4 3-5
3/4.3.1, Table 4.3-1	3/4 3-11
3/4.3.1, Table 4.3-1 (continued)	3/4 3-13

NO CHANGE
INCLUDED FOR REFERENCE TO ACTION 2

TABLE 3.3-1
REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NUMBER OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1,2 and *	12
2. Power Range, Neutron Flux	4	2	3	1,2, and 3*	2
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1,2	2
4. Deleted					
5. Intermediate Range, Neutron Flux	2	1	2	1,2 and *	3
6. Source Range, Neutron Flux					
A. Startup	2	1	2	2## and *	4
B. Shutdown	2	0	1	3,4, and 5	5
7. Overtemperature ΔT	4	2	3	1,2	6
8. Overpower ΔT	4	2	3	1,2	6
9. Pressurizer Pressure-Low	4	2	3	1,2	6
10. Pressurizer Pressure--High	4	2	3	1,2	6

TABLE 3.3-1 (Continued)

TABLE NOTATION

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

High voltage to detector may be de-energized above P-6.

If ACTION Statement 1 is entered as a result of Reactor Trip Breaker (RTB) or Reactor Trip Bypass Breakers (RTBB) maintenance testing results exceeding the following acceptance criteria, NRC reporting shall be made within 30 days in accordance with Specification 6.9.2:

1. A RTB or RTBB trip failure during any surveillance test with less than or equal to 300 grams of weight added to the breaker trip bar.

2. A RTB or RTBB time response failure that results in the overall reactor trip system time response exceeding the Technical Specification limit.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, be in HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.1 provided the other channel is OPERABLE.

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

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

b. The Minimum Channels OPERABLE requirement is met; however, ONE ~~the inoperable~~ channel may be bypassed for up to 4 hours for surveillance testing ~~of other channels~~ per Specification 4.3.1.1.1.

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

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK[15]	CHANNEL CALIBRATION[15]	CHANNEL FUNCTIONAL TEST[15]	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip Switch	N.A.	N.A.	(9)	1, 2, and *
2. Power Range, Neutron Flux		(2), (3) (6) (17)	(18)	1, 2, and 3*
3. Power Range, Neutron Flux, High Positive Rate	N.A.	(6)	(18)	1, 2
4. Deleted				
5. Intermediate Range, Neutron Flux		(6)	S/U[1]	1, 2 and *
6. Source Range, Neutron Flux	(7)	(6)	(16) and S/U[1]	2, 3, 4, 5 and *
7. Overtemperature ΔT				1, 2
8. Overpower ΔT				1, 2
9. Pressurizer Pressure--Low				1, 2
10. Pressurizer Pressure--High				1, 2
11. Pressurizer Water Level--High				1, 2
12. Loss of Flow - Single Loop				1

TABLE 4.3-1 (Continued)

NOTATION

* With the reactor trip system breakers closed and the control rod drive system capable of rod withdrawal.

(1) - If not performed in previous 31 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER.

(3) - Compare incore to excore axial offset above 15% of RATED THERMAL POWER. Recalibrate if absolute difference ≥ 3 percent.

(4) - Manual SSPS functional input check in accordance with the Surveillance Frequency Control Program.

(5) - Each train or logic channel shall be tested in accordance with the Surveillance Frequency Control Program.

(6) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(7) - Below P-6 (Block of Source Range Reactor Trip) setpoint.

(8) - Deleted

(9) - The CHANNEL FUNCTIONAL TEST shall independently verify the OPERABILITY of the Undervoltage and Shunt Trip mechanism for the Manual Reactor Trip Function.

The Test shall also verify OPERABILITY of the Bypass Breaker Trip circuits.

(10) - DELETED

(11) - The CHANNEL FUNCTIONAL TEST shall independently verify the OPERABILITY of the Reactor Trip Breaker Undervoltage and Shunt Trip mechanisms.

(12) - DELETED

(13) - Verify operation of Bypass Breakers Shunt Trip function from local pushbutton while breaker is in the test position prior to placing breaker in service.

(14) - Perform a functional test of the Bypass Breakers U.V. Attachment via the SSPS.

(15) - Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

(16) - At the frequency specified in the Surveillance Frequency Control Program.

(17) - INSERT 1

(18) - INSERT 2

INSERT 1

In MODES 1 and 2, the SSPS input relays are excluded from this Surveillance when the installed bypass test capability is used to perform this Surveillance.

INSERT 2

The SSPS input relays are excluded from this Surveillance when the installed bypass test capability is used to perform this Surveillance.

NO CHANGE
INCLUDED FOR REFERENCE TO ACTION 2

TABLE 3.3-1
REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NUMBER OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
1. Manual Reactor Trip	2	1	2	1,2 and *	12
2. Power Range, Neutron Flux	4	2	3	1,2 and 3*	2
3. Power Range, Neutron Flux High Positive Rate	4	2	3	1,2	2
4. Deleted					
5. Intermediate Range, Neutron Flux	2	1	2	1,2 and *	3
6. Source Range, Neutron Flux					
A. Startup	2	1	2	2##, and *	4
B. Shutdown	2	0	1	3,4 and 5	5
7. Overtemperature ΔT	4	2	3	1,2	6
8. Overpower ΔT	4	2	3	1,2	6
9. Pressurizer Pressure-Low	4	2	3	1,2	6
10. Pressurizer Pressure--High	4	2	3	1,2	6

TABLE 3.3-1 (Continued)

TABLE NOTATION

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

High voltage to detector may be de-energized above P-6.

If ACTION Statement 1 is entered as a result of Reactor Trip Breaker (RTB) or Reactor Trip Bypass Breaker (RTBB) maintenance testing results exceeding the following acceptance criteria, NRC reporting shall be made within 30 days in accordance with Specification 6.9.2:

1. A RTB or RTBB trip failure during any surveillance test with less than or equal to 300 grams of weight added to the breaker trip bar.
2. A RTB or RTBB time response failure that results in the overall reactor trip system time response exceeding the Technical Specification limit.

ACTION STATEMENTS

ACTION 1 - With the number of channels OPERABLE one less than required by the Minimum Channels OPERABLE requirement, be in HOT STANDBY within 6 hours; however, one channel may be bypassed for up to 2 hours for surveillance testing per Specification 4.3.1.1.1 provided the other channel is OPERABLE.

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

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

b. The Minimum Channels OPERABLE requirement is met; however, ONE ~~the inoperable~~ channel may be bypassed for up to 4 hours for surveillance testing ~~of other channels~~ per Specification 4.3.1.1.1.

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

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

TABLE 4.3-1

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK[15]	CHANNEL CALIBRATION[15]	CHANNEL FUNCTIONAL TEST[15]	MODES IN WHICH SURVEILLANCE REQUIRED
1. Manual Reactor Trip Switch	N.A.	N.A.	(9)	1, 2, and *
2. Power Range, Neutron Flux		(2), (3) (6) (17)	(18)	1, 2, and 3*
3. Power Range, Neutron Flux, High Positive Rate	N.A.	(6)	(18)	1, 2
4. Deleted				
5. Intermediate Range, Neutron Flux		(6)	S/U[1]	1, 2 and *
6. Source Range, Neutron Flux	(7)	(6)	(16) and S/U[1]	2, 3, 4, 5 and *
7. Overtemperature ΔT				1, 2
8. Overpower ΔT				1, 2
9. Pressurizer Pressure--Low				1, 2
10. Pressurizer Pressure--High				1, 2
11. Pressurizer Water Level--High				1, 2
12. Loss of Flow - Single Loop				1

TABLE 4.3-1 (Continued)

NOTATION

* With the reactor trip system breakers closed and the control rod drive system capable of rod withdrawal.

(1) - If not performed in previous 31 days.

(2) - Heat balance only, above 15% of RATED THERMAL POWER.

(3) - Compare incore to excore axial offset above 15% of RATED THERMAL POWER. Recalibrate if absolute difference ≥ 3 percent.

(4) - Manual SSPS functional input check in accordance with the Surveillance Frequency Control Program.

(5) - Each train or logic channel shall be tested in accordance with the Surveillance Frequency Control Program.

(6) - Neutron detectors may be excluded from CHANNEL CALIBRATION.

(7) - Below P-6 (Block of Source Range Reactor Trip) setpoint.

(8) - Deleted

(9) - The CHANNEL FUNCTIONAL TEST shall independently verify the OPERABILITY of the Undervoltage and Shunt Trip mechanism for the Manual Reactor Trip Function.

The Test shall also verify OPERABILITY of the Bypass Breaker Trip circuits.

(10) - DELETED

(11) - The CHANNEL FUNCTIONAL TEST shall independently verify the OPERABILITY of the Reactor Trip Breaker Undervoltage and Shunt Trip mechanisms.

(12) - DELETED

(13) - Verify operation of Bypass Breakers Shunt Trip function from local pushbutton while breaker is in the test position prior to placing breaker in service.

(14) - Perform a functional test of the Bypass Breakers U.V. Attachment via the SSPS.

(15) Frequencies are specified in the Surveillance Frequency Control Program unless otherwise noted in the table.

(16) At the frequency specified in the Surveillance Frequency Control Program.

(17)- INSERT 1

(18)- INSERT 2

INSERT 1

In MODES 1 and 2, the SSPS input relays are excluded from this Surveillance when the installed bypass test capability is used to perform this Surveillance.

INSERT 2

The SSPS input relays are excluded from this Surveillance when the installed bypass test capability is used to perform this Surveillance.

TECHNICAL SPECIFICATION BASES PROPOSED CHANGES (FOR INFORMATION ONLY)

3/4.3.1 and 3/4.3.2 PROTECTIVE AND ENGINEERED SAFETY FEATURES (ESF) INSTRUMENTATION

The OPERABILITY of the protective and ESF instrumentation systems and interlocks ensure that 1) the associated ESF action and/or reactor trip will be initiated when the parameter monitored by each channel or combination thereof exceeds its setpoint, 2) the specified coincidence logic and sufficient redundancy is maintained to permit a channel to be out of service for testing or maintenance consistent with maintaining an appropriate level of reliability of the Reactor Protection and Engineered Safety Features instrumentation and, 3) sufficient system functional capability is available from diverse parameters.

The OPERABILITY of these systems is required to provide the overall reliability, redundance and diversity assumed available in the facility design for the protection and mitigation of accident and transient conditions. The integrated operation of each of these systems is consistent with the assumptions used in the accident analyses.

The Trip Setpoints are the nominal values at which the bistables are set. Any bistable is considered to be properly adjusted when the "as-left" value is within the band for CHANNEL CALIBRATION accuracy (i.e., ± rack calibration + comparator setting accuracy).

The Trip Setpoints used in the bistables are based on the analytical limits stated in the UFSAR. The selection of these Trip Setpoints is such that adequate protection is provided when all sensor and processing time delays are taken into account. To allow for calibration tolerances, instrumentation uncertainties, instrument drift, and severe environment errors for those Reactor Protection System (RPS) channels that must function in harsh environments as defined by 10 CFR 50.49, the Trip Setpoints and Allowable Values specified in the Technical Specification Limiting Conditions for Operation (LCO's) are conservatively adjusted with respect to the analytical limits. The methodology used to calculate the Trip Setpoints is consistent with Instrument Society of America standard ISA-S67.04-1982, which is endorsed via NRC Regulatory Guide 1.105, Rev. 2. The actual nominal Trip Setpoint entered into the bistable is more conservative than that specified by the Allowable Value to account for changes in random measurement errors detectable by a CHANNEL FUNCTIONAL TEST. One example of such a change in measurement error is drift during the surveillance interval. If the measured setpoint does not exceed the Allowable Value, the bistable is considered OPERABLE.

Setpoints in accordance with the Allowable Value ensure that the safety analyses which demonstrate that safety limits are not violated remain valid (provided the unit is operated within the LCO's at the onset of any design basis event and the equipment functions as designed).

The Trip Setpoints and Allowable Values listed in the LCO's incorporate all of the known uncertainties applicable for each channel. The magnitudes of these uncertainties are factored into the determination of each Trip Setpoint. All field sensors and signal processing equipment for these channels are assumed to operate within the allowances of these uncertainty magnitudes.

The surveillance requirements specified for these systems ensure that the overall system functional capability is maintained comparable to the original design standards. The periodic surveillance tests are sufficient to demonstrate this capability. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. Specified surveillance and maintenance outage times have been determined in accordance with WCAP-10271, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection

Instrumentation System," and Supplements to that report. Out of service times were determined based on maintaining an appropriate level of reliability of the Reactor Protection System and Engineered Safety Features instrumentation.

The verification of response time provides assurance that the reactor trip and the engineered safety features actuation associated with each channel is completed within the time limit assumed in the safety analysis. The Surveillance Frequency is based on operating experience, equipment reliability, and plant risk and is controlled under the Surveillance Frequency Control Program. Response Time acceptance criteria have been relocated to UFSAR Sections 7.2 and 7.3 tables. No credit is taken in the analysis for those channels with response times indicated as not applicable (i.e., N.A.). The FSAR tables 7.3-8 Note 8 response times for feedwater isolation are based on WCAP-16503, "Salem Unit 1 and Unit 2 Containment Response to LOCA and MSLB for Containment Fan Cooler Unit (CFCU) Margin Recovery Project," Revision 3, (LCR S06-10). SGFP trip and FIV closure are credited in the containment analyses for LOCA and MSLB in case an FRV fails open.

Response time may be verified by actual response time tests in any series of sequential, overlapping or total channel measurements, or by the summation of allocated sensor response times with actual response time tests on the remainder of the channel. Allocations for sensor response times may be obtained from: (1) historical records based on acceptable response time tests (hydraulic, noise, or power interrupt tests), (2) inplace, onsite, or offsite (e.g. vendor) test measurements, or (3) utilizing vendor engineering specifications. WCAP-13632-P-A, Revision 2, "Elimination of Pressure Sensor Response Time Testing Requirements" provides the basis and methodology for using allocated sensor response times in the overall verification of the channel response time for specific sensors identified in the WCAP. Response time verification for other sensor types, and other components that do not have plant-specific NRC approval to use alternate means of verification, must be demonstrated by test.

The allocation for sensor response times must be verified prior to placing the component in operational service and re-verified following maintenance that may adversely affect response time. In general, electrical repair work does not impact response time provided the parts used for repair are of the same type and value. One example where response time could be affected is replacing the sensing assembly of a transmitter.

Channel testing in a bypassed condition shall be performed without lifting leads or jumpering bistables.

INSERT 1

3/4.3.3 MONITORING INSTRUMENTATION

3/4.3.3.1 RADIATION MONITORING INSTRUMENTATION

The OPERABILITY of the radiation monitoring channels ensures that 1) the radiation levels are continually measured in the areas served by the individual channels and 2) the alarm or automatic action is initiated when the radiation level trip setpoint is exceeded.

In the postulated Fuel Handling Accident, the revised dose calculations, performed using 10 CFR 50.67 and Regulatory Guide 1.183, Alternative Source Term, do not take credit for automatic containment purge isolation thus allowing for continuous monitoring of containment activity until containment closure is achieved. If required, containment purge isolation can be initiated manually from the control room.

INSERT 1

The CHANNEL CALIBRATION Surveillance for the Power Range Neutron Flux Function instrumentation is modified by Note 17. Note 17 states that in MODES 1 and 2 the SSPS input relays are excluded from this Surveillance when the installed bypass test capability is used to perform this Surveillance. When the installed bypass test capability is used, the channel is tested in a bypassed versus tripped condition. To preclude placing the channel in a tripped condition, the SSPS input relays are excluded from this Surveillance. The exclusion of the SSPS input relays from this test is intended to reduce the potential for an inadvertent reactor trip during Surveillance testing. Therefore, the exclusion of the SSPS input relays from the Surveillance is only applicable in MODES 1 and 2. The SSPS input relays must be included in the CHANNEL CALIBRATION surveillance at least once every 18 months.

The CHANNEL FUNCTIONAL TEST Surveillances for the Power Range Neutron Flux and Power Range Neutron Flux High Positive Rate Function instrumentation are modified by Note 18. Note 18 states, that the SSPS input relays are excluded from this Surveillance when the installed bypass test capability is used to perform this Surveillance. When the installed bypass test capability is used, the channel is tested in a bypassed versus tripped condition. To preclude placing the channel in a tripped condition, the SSPS input relays are excluded from this Surveillance. The exclusion of the SSPS input relays from this test is intended to reduce the potential for an inadvertent reactor trip during Surveillance testing. The SSPS input relays must be included in the CHANNEL CALIBRATION surveillance at least once every 18 months.

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Attachment 4

Westinghouse Application for Withholding and Affidavit

Westinghouse

Westinghouse Electric Company
Engineering, Equipment and Major Projects
1000 Westinghouse Drive, Building 3
Cranberry Township, Pennsylvania 16066
USA

U.S. Nuclear Regulatory Commission
Document Control Desk
11555 Rockville Pike
Rockville, MD 20852

Direct tel: (412) 374-4643
Direct fax: (724) 940-8560
e-mail: greshaja@westinghouse.com
Proj letter: PSE-15-16

CAW-15-4117

March 11, 2015

APPLICATION FOR WITHHOLDING PROPRIETARY
INFORMATION FROM PUBLIC DISCLOSURE

Subject: WCAP-17947-P, Revision 0, "Power Range Nuclear Instrumentation System Bypass Test Instrumentation for Salem Units 1 and 2" (Proprietary)

The proprietary information for which withholding is being requested in the above-referenced report is further identified in Affidavit CAW-15-4117 signed by the owner of the proprietary information, Westinghouse Electric Company LLC. The Affidavit, which accompanies this letter, sets forth the basis on which the information may be withheld from public disclosure by the Commission and addresses with specificity the considerations listed in paragraph (b)(4) of 10 CFR Section 2.390 of the Commission's regulations.

Accordingly, this letter authorizes the utilization of the accompanying Affidavit by Public Service Electric and Gas (PSEG).

Correspondence with respect to the proprietary aspects of the Application for Withholding or the Westinghouse Affidavit should reference CAW-15-4117, and should be addressed to James A. Gresham, Manager, Regulatory Compliance, Westinghouse Electric Company, 1000 Westinghouse Drive, Building 3 Suite 310, Cranberry Township, Pennsylvania 16066.

Very truly yours,

James A. Gresham, Manager
Regulatory Compliance

CAW-15-4117
March 11, 2015

AFFIDAVIT

COMMONWEALTH OF PENNSYLVANIA:

ss

COUNTY OF BUTLER:

I, James A. Gresham, am authorized to execute this Affidavit on behalf of Westinghouse Electric Company LLC (Westinghouse), and that the averments of fact set forth in this Affidavit are true and correct to the best of my knowledge, information, and belief.

James A. Gresham, Manager
Regulatory Compliance

(1) I am Manager, Regulatory Compliance, Westinghouse Electric Company LLC (Westinghouse), and as such, I have been specifically delegated the function of reviewing the proprietary information sought to be withheld from public disclosure in connection with nuclear power plant licensing and rule making proceedings, and am authorized to apply for its withholding on behalf of Westinghouse.

(2) I am making this Affidavit in conformance with the provisions of 10 CFR Section 2.390 of the Commission's regulations and in conjunction with the Westinghouse Application for Withholding Proprietary Information from Public Disclosure accompanying this Affidavit.

(3) I have personal knowledge of the criteria and procedures utilized by Westinghouse in designating information as a trade secret, privileged or as confidential commercial or financial information.

(4) Pursuant to the provisions of paragraph (b)(4) of Section 2.390 of the Commission's regulations, the following is furnished for consideration by the Commission in determining whether the information sought to be withheld from public disclosure should be withheld.

(i) The information sought to be withheld from public disclosure is owned and has been held in confidence by Westinghouse.

(ii) The information is of a type customarily held in confidence by Westinghouse and not customarily disclosed to the public. Westinghouse has a rational basis for determining the types of information customarily held in confidence by it and, in that connection, utilizes a system to determine when and whether to hold certain types of information in confidence. The application of that system and the substance of that system constitute Westinghouse policy and provide the rational basis required.

Under that system, information is held in confidence if it falls in one or more of several types, the release of which might result in the loss of an existing or potential competitive advantage, as follows:

(a) The information reveals the distinguishing aspects of a process (or component, structure, tool, method, etc.) where prevention of its use by any of Westinghouse's competitors without license from Westinghouse constitutes a competitive economic advantage over other companies.

(b) It consists of supporting data, including test data, relative to a process (or component, structure, tool, method, etc.), the application of which data secures a competitive economic advantage, e.g., by optimization or improved marketability.

(c) Its use by a competitor would reduce his expenditure of resources or improve his competitive position in the design, manufacture, shipment, installation, assurance of quality, or licensing a similar product.

(d) It reveals cost or price information, production capacities, budget levels, or commercial strategies of Westinghouse, its customers or suppliers.

(e) It reveals aspects of past, present, or future Westinghouse or customer funded development plans and programs of potential commercial value to Westinghouse.

(f) It contains patentable ideas, for which patent protection may be desirable.

(iii) There are sound policy reasons behind the Westinghouse system which include the following:

(a) The use of such information by Westinghouse gives Westinghouse a competitive advantage over its competitors. It is, therefore, withheld from disclosure to protect the Westinghouse competitive position.

(b) It is information that is marketable in many ways. The extent to which such information is available to competitors diminishes the Westinghouse ability to sell products and services involving the use of the information.

(c) Use by our competitor would put Westinghouse at a competitive disadvantage by reducing his expenditure of resources at our expense.

(d) Each component of proprietary information pertinent to a particular competitive advantage is potentially as valuable as the total competitive advantage. If competitors acquire components of proprietary information, any one component

may be the key to the entire puzzle, thereby depriving Westinghouse of a competitive advantage.

(e) Unrestricted disclosure would jeopardize the position of prominence of Westinghouse in the world market, and thereby give a market advantage to the competition of those countries.

(f) The Westinghouse capacity to invest corporate assets in research and development depends upon the success in obtaining and maintaining a competitive advantage.

(iv) The information is being transmitted to the Commission in confidence and, under the provisions of 10 CFR Section 2.390, it is to be received in confidence by the Commission.

(v) The information sought to be protected is not available in public sources or available information has not been previously employed in the same original manner or method to the best of our knowledge and belief.

(vi) The proprietary information sought to be withheld in this submittal is that which is appropriately marked in WCAP-17947-P, Revision 0, "Power Range Nuclear Instrumentation System Bypass Test Instrumentation for Salem Units 1 and 2" (Proprietary), dated March 2015, for submittal to the Commission, being transmitted by Public Service Electric and Gas (PSEG) letter and Application for Withholding Proprietary Information from Public Disclosure, to the Document Control Desk. The proprietary information as submitted by Westinghouse is that associated with a modification that allows testing of the Reactor Trip System (RTS) channels in a "bypassed" condition, as opposed to the "tripped" condition by installing Bypass Test Instrumentation (BTI), and may be used only for that purpose.

(a) This information is part of that which will enable Westinghouse to:

(i) Assist PSEG with obtaining NRC approval of a License Amendment Request that would allow installation of BTI hardware.

(ii) Provide licensing support for customer submittals.

(b) Further this information has substantial commercial value as follows:

(i) Westinghouse plans to sell the use of similar information to its customers for the purpose of installation of BTI hardware.

(ii) Westinghouse can sell support and defense of industry guidelines and acceptance criteria for plant-specific applications.

(iii) The information requested to be withheld reveals the distinguishing aspects of a methodology which was developed by Westinghouse.

Public disclosure of this proprietary information is likely to cause substantial harm to the competitive position of Westinghouse because it would enhance the ability of competitors to provide similar technical evaluation justifications and licensing defense services for commercial power reactors without commensurate expenses. Also, public disclosure of the information would enable others to use the information to meet NRC requirements for licensing documentation without purchasing the right to use the information.

The development of the technology described in part by the information is the result of applying the results of many years of experience in an intensive Westinghouse effort and the expenditure of a considerable sum of money.

In order for competitors of Westinghouse to duplicate this information, similar technical programs would have to be performed and a significant manpower effort, having the requisite talent and experience, would have to be expended.

Further the deponent sayeth not.

PROPRIETARY INFORMATION NOTICE

Transmitted herewith are proprietary and non-proprietary versions of documents furnished to the NRC in connection with requests for generic and/or plant-specific review and approval.

In order to conform to the requirements of 10 CFR 2.390 of the Commission's regulations concerning the protection of proprietary information so submitted to the NRC, the information which is proprietary in the proprietary versions is contained within brackets, and where the proprietary information has been deleted in the non-proprietary versions, only the brackets remain (the information that was contained within the brackets in the proprietary versions having been deleted). The justification for claiming the information so designated as proprietary is indicated in both versions by means of lower case letters (a) through (f) located as a superscript immediately following the brackets enclosing each item of information being identified as proprietary or in the margin opposite such information. These lower case letters refer to the types of information Westinghouse customarily holds in confidence identified in Sections (4)(ii)(a) through (4)(ii)(f) of the Affidavit accompanying this transmittal pursuant to 10 CFR 2.390(b)(1).

COPYRIGHT NOTICE

Attachment 5

Westinghouse WCAP-17947-NP (Non-Proprietary)

WCAP-17947-NP
Revision 0

March 2015

Power Range Nuclear Instrumentation System Bypass Test Instrumentation for Salem Units 1 and 2

Westinghouse

WCAP-17947-NP
Revision 0

Power Range Nuclear Instrumentation System Bypass Test Instrumentation for Salem Units 1 and 2

Frank P. Ferri*
Plant Licensing

Jorge V. Carvajal*
Global Technology Development

March 2015

Reviewer: James D. Andrachek*
Plant Licensing

Frederick W. Hantz*
Nuclear Instrumentation System

Approved: Dewey C. Olinski*, Manager
Plant Licensing

Robert E. Single*, Manager
Nuclear Instrumentation System

*Electronically approved records are authenticated in the electronic document management system.

Westinghouse Electric Company LLC
1000 Westinghouse Drive
Cranberry Township, PA 16066, USA

ABSTRACT

In order to reduce the potential for spurious reactor trips, which reduces the potential transient associated with a trip, a modification can be implemented that allows testing of the Reactor Trip System (RTS) channels in a “bypassed” condition, as opposed to the “tripped” condition. If a channel is in the tripped condition, and a second comparator trips in a redundant channel, which can be caused by a human error, spurious transient, or channel failure, will result in a reactor trip. With the Bypass Test Instrumentation (BTI), a spurious reactor trip will be avoided, which reduces the potential transient associated with a reactor trip. Routine bypass testing capability is being provided for the Power Range Nuclear Instrumentation System (NIS) reactor trip functions.

Various aspects of the BTI installation are addressed in this topical report (TR). These aspects include a demonstration of the functionality of the BTI hardware, the BTI design features which comply with the applicable U.S. Nuclear Regulatory Commission (NRC) regulations, regulatory guidance, and industry standards associated with testing in bypass. The administrative controls that will be implemented are also identified.

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ACRONYMS

AOT	Allowed Outage Time
BTI	Bypass Test Instrumentation
ESFAS	Engineered Safety Feature Actuation System
FAT	Factory Acceptance Test
GDC	General Design Criteria
IEEE	Institute of Electrical and Electronics Engineers
LED	Light Emitting Diode
NIS	Nuclear Instrumentation System
NRC	U.S. Nuclear Regulatory Commission
OBE	Operating Basis Earthquake
PR	Power Range
PSEG	Public Service Electric & Gas
PWROG	Pressurized Water Reactor Owners Group
RG	Regulatory Guide
RTS	Reactor Trip System
SER	Safety Evaluation Report
SSE	Safe Shutdown Earthquake
SSPS	Solid State Protection System
TR	Topical Report
TS	Technical Specifications
VAC	Voltage Alternating Current
WOG	Westinghouse Owners Group

1 INTRODUCTION

The Reactor Trip System (RTS) Power Range (PR) Nuclear Instrumentation System (NIS) utilizes a two-out-of-four coincidence logic from redundant channels to initiate protective actions. In the current design, the RTS NIS PR analog channel comparators are placed in the "tripped" condition during channel testing or if a channel is inoperable. In the current design of testing a channel in the tripped condition, a redundant channel cannot be tested without bypassing an inoperable channel that is in the tripped condition. Additionally, routine testing of a channel in the tripped condition, can result in the potential for an unnecessary reactor trip, if a second comparator trips in a redundant channel, which can be caused by a human error, spurious transient, or channel failure.

The benefits associated with the installation of the BTI at Salem Units 1 and 2 are:

- Analog channel online surveillance testing can be performed with the comparator outputs bypassed, rather than tripped, thus reducing the potential for unnecessary reactor trips due to a failure or transient in a redundant channel.

- Surveillance testing can be performed on an operable channel in bypass, if a redundant channel is inoperable and in the tripped condition, reducing the potential for a spurious reactor trip by testing in bypass.

- One channel of the PR reactor trip function can be bypassed for maintenance, as opposed to being in the tripped condition.

- The BTI is integral to the existing racks, thus eliminating the need for portable test equipment, lifting leads, or using jumpers to bypass the channel, which is prohibited to done on a routine basis.

This TR provides the licensing basis for the BTI for Salem Units 1 and 2. It is structured into five sections, as follows:

1. An introduction of the concept of the BTI and its purpose.

2. A brief background of the issue of bypass testing and prior regulatory positions on this subject.

3. A detailed description of the design of the bypass systems with figures to illustrate the operation. ([

   ]^{a,c})

4. A discussion of how the BTI design complies with the applicable General Design Criteria (GDC), Regulatory Guides (RGs) and Institute of Electrical and Electronics Engineers (IEEE) Standards.

5. A conclusion supporting the implementation of the BTI.

2 BACKGROUND

In response to a concern regarding the impact of RTS and Engineered Safety Features Actuation System (ESFAS) Technical Specification (TS) instrumentation surveillance testing and maintenance activities on plant operations, i.e. inadvertent reactor trips and ESF actuations, the Pressurized Water Reactor Owners Group (PWROG) (formerly the Westinghouse Owners Group [WOG]) initiated a program to justify extending the RTS and ESFAS instrumentation bypass test times, Completion Times , Allowed Outage Times (AOTs) and Surveillance Frequencies to provide additional time to perform surveillance and maintenance activities. WCAP-10271-P-A and Supplements 1 and 2, "Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System" (References 1–3) (and WCAP-14333-P-A, "Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times" [Reference 4]) justified extending the RTS and ESFAS instrumentation bypass test times, Completion Times, AOTs and Surveillance Frequencies. One of the provisions for surveillance testing discussed in References 1–3 was to allow routine testing of the RTS and ESFAS instrumentation channels in a bypassed condition instead of a tripped condition.

The NRC Safety Evaluation Reports (SERs) for WCAP-10271-P-A and Supplements 1 and 2 that were issued in February 1985 (RTS instrumentation) and in February 1989 (ESFAS instrumentation) state that the use of temporary jumpers or the lifting of leads is unacceptable for bypassing a channel for routine surveillance testing.

The installation of the BTI at the Salem Units 1 and 2 will allow testing of the PR NIS channels in a bypassed condition using installed instrumentation, and does not utilize temporary jumpers or the lifting of leads as discussed above.

The use of the installed BTI for the PR NIS will result in a reduction in the potential number of inadvertent reactor trips that could potentially occur during testing in a tripped condition. Testing in bypass eliminates the partial trip condition that is associated with testing in the tripped condition for the affected PR RTS functions.

3 DETAILED DESIGN DESCRIPTION

The bypass system allows the channel to be tested without tripping the channel. The bypass system accomplishes this by imposing a signal in parallel, thus maintaining the Solid State Protection System (SSPS) in an untripped condition.

3.1 NIS BYPASS PANEL

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The potential for failure of the NIS bypass panel is low, based on the following characteristics. All components are purely mechanical or electro-mechanical and will perform at least 50,000 operations (based on manufacturers' reports) under normal conditions without failure. Over the 40 year life of the plant, it is expected that these components will be exercised less than 200 times. The keylock switch, toggle switch, and relay were cycled 300 times for testing purposes. This constitutes one cycle per quarter for 60 years with an added 25 percent margin.

3.2 FAULT CONDITIONS

Each NIS bypass panel is separated by a protection set; therefore, a single fault in a bypass panel would not prevent the other three channels from performing the specified safety function. The portions of the BTI panels that are non-Class 1E are isolated from the Class 1E circuits by the K1 relay coil to contact as shown in Figure 5-1. Therefore, there is no possibility that a control system fault could propagate to all of the bypass panels and simultaneously adversely affect all protection sets. Subsection 4.2.3 discusses the isolation and separation of the Class 1E and non-Class 1E equipment in the bypass panels.

The NIS PR bypass panel is protected by a circuit breaker to prevent damage to the panel. The breaker status is monitored by the same LED that indicates that the bypass panel is enabled. This LED will not light if the breaker is tripped. Since this LED is also the indication that the panel is enabled, if this LED is not lit, due to a lack of power to the bypass panel, the bypass panel will not allow any function to go into bypass. This will prevent a channel from being placed into bypass with no bypass signal available.

3.3 FAILURE DETECTION

The different types of potential credible failure modes in the NIS PR bypass panel are as follows:

1. Power unavailable to the bypass panel
2. Breaker in the bypass panel tripped
3. An LED failure
4. A contact failure

With power unavailable to the bypass panel, the panel is unable to put a channel in bypass. This is easily detected by the absence of a lit LED when the keylock switch is turned from "NORMAL" to "BYPASS ENABLE." Additionally, there is no control room annunciation of the attempt to place a channel in bypass.

The circuit breaker status is monitored by the same LED that indicates that the bypass panel is enabled, i.e., that a channel is bypassed. This LED will not light if the breaker is tripped. Since this LED also provides the indication that the panel is enabled, if this LED is not lit, due to a lack of power, the bypass panel will not allow any channel to go into bypass. This will prevent a channel from being placed into bypass with no bypass signal available (see Figure 5-1).

3.4 HUMAN FACTORS/ADMINISTRATIVE CONTROL

Human Factors and Administrative Controls have been designed into the BTI for Salem Units 1 and 2. The design features incorporated that address the Human Factors and Administrative Controls are:

- A keylock switch (NIS PR bypass panel)
- LEDs on the bypass panels
- Control board annunciation of the bypass condition
- Permanently installed bypass test capability

The bypass system is located in the cabinets where the protection channels are located. Therefore, the test technician will be aware of the channel that is in bypass and the channels that are not in bypass, without having to depend on a non-local indication.

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3.5 RELIABILITY

Steps have been taken to ensure the operation of the BTI. The key to ensuring proper BTI operation is associated with the BTI reliability. The BTI is designed with the reliability characteristics necessary to preserve the total integrity of the SSPS. The BTI is designed to reduce the frequency of failures through the utilization of highly reliable components.

IEEE Standard 603-1991 delineates certain functional performance requirements regarding the aspects of system reliability for the protection systems. Because the BTI will be implemented to support the SSPS, it has been evaluated against those criteria that are applicable to its design.

All of the components of the BTI are mechanical or electro-mechanical and will be reliable for at least 50,000 operations under normal environmental conditions.

3.6 INDICATION AND ANNUNCIATION

The BTI is provided with the capability to provide timely and accurate information to the control room operator, as well as the test technician performing the bypass testing. In accordance with IEEE Standard 603-1991 and RG 1.47, control room annunciation must be provided for the status of any RTS NIS PR channel that is put into a bypassed condition. Main control room alarm/status light indicators are provided to ensure that the operator knows which BTI panel has a protection set channel instrumentation loop in the bypass condition.

The BTI has the capability to provide local indication of the status of the channels and the bypass panel. It can be determined from the position of the keylock switch on the NIS bypass panel that the technician has attempted to put the channel in test, and the lighting of the LED on the bypass panel will indicate that power is available to the bypass panel. The LEDs that are associated with the locking toggle switches will identify to the technician that an individual channel has been placed in the bypass condition.

3.7 OPERATOR ACTIONS

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3.8 EQUIPMENT QUALIFICATION

Equipment qualification for the BTI must address several issues. Since the NIS PR bypass panels are installed in the Class 1E instrumentation racks, it must be shown that: (1) the installation of these bypass system in these instrumentation racks will not adversely affect the seismic qualification of the Class 1E racks, and (2) the panels are able to withstand the required seismic levels associated with Salem Units 1 and 2 and still continue to show structural integrity and electrical isolation. All components used in the cards and bypass panels are acceptable for the environment expected in the cabinets. The BTI equipment to be installed in the Class 1E instrumentation cabinets was subjected to multi-axis, multi-frequency inputs in accordance with RG 1.100. The equipment was subjected to Westinghouse generic operating basis earthquake (OBE) and safe shutdown earthquake (SSE) testing. The BTI generic seismic qualification and environmental evaluation bounds the PSEG Nuclear current licensing basis and was performed in accordance with WCAP-8587 (Reference 6).

3.9 ELECTROMAGNETIC COMPATIBILITY

The NIS PR bypass panels and associated wiring are completely contained inside a metal cabinet; therefore, the dominant entry of electromagnetic interference would be expected to be conducted through the field cabling. Additionally, the NIS bypass panels employ high level signals (118 VAC) that are not susceptible to radiated or conducted interference. The ability of the BTI panel to affect other equipment within the same cabinet is minimal due to the panel metal assembly and, more importantly, the very low duty cycle of the bypass relays. In the event that the BTI panel is required to be placed in bypass, the relays are manually actuated only twice. Additionally, the relays have been provided with arc suppression circuits in order to minimize any interference issues.

3.10 DISCUSSION OF DIFFERENCE BETWEEN UNITS

The comparators that PSEG Nuclear selected to be bypassed affect the Power Range NIS Functions as identified in Table 5-1.

Each NIS BTI panel is operated by unit-specific keylock switches. The Unit 1 bypass panel assemblies are part numbers 10105D37G01 thru G04, and will use the 3A98714G02 keylock switch. The Unit 2 bypass panel assemblies are part numbers 10105D37G05 thru G08, and will use the 3A98714G05 keylock switch. The keylock switch operates the same at both units; however, the use of unit specific keylock switches provides an administrative control to prevent two channels from being placed in bypass at one time. This change in keylock switch consequently changes the base panel part numbers so that the Unit 1 bypass assemblies use sub-panel 4D04921G02 and the Unit 2 bypass assemblies use sub-panel 4D04921G04.

Finally, the NIS PR panels are individually numbered per channel per unit, as shown in Table 5-2.

4 COMPLIANCE WITH THE APPLICABLE REGULATIONS, REGULATORY GUIDES, AND INDUSTRY STANDARDS

As with any modifications to the RTS, compliance with the applicable regulations, Regulatory Guidance and Industry Standards must be addressed. This section addresses the design of the BTI to the current applicable:

- GDCs
- RGs
- IEEE standards

4.1 GDCs

The following GDCs are applicable to the BTI and are discussed in subsections 4.1.1 through 4.1.7:

- GDC 2 – Design Bases for Protection Against Natural Phenomena
- GDC 19 – Control Room
- GDC 20 – Protection System Functions
- GDC 21 – Protection System Reliability and Testability
- GDC 22 – Protection System Independence
- GDC 23 – Protection System Failure Modes
- GDC 24 – Separation of Protection and Control Systems

4.1.1 GDC 2 – Design Bases for Protection from Natural Phenomena

GDC 2 states that "Systems and components important to safety shall be designed to withstand the effects of natural phenomena such as earthquakes, tornadoes, hurricanes, floods, tsunami, and seiches without loss of capability to perform their safety functions." This criterion is applicable to the installation of the BTI at Salem Units 1 and 2 because the BTI is being added to the Class 1E NIS cabinets. The BTI cannot adversely affect the existing seismic qualification of the cabinets, nor can the BTI become a missile in a seismic event and thus adversely affect any safety related equipment.

The BTI must also be shown to maintain its required functionality during and after a seismic event. Equipment qualification reports have been prepared to address all seismic qualification concerns. Section 3.8 discusses the equipment qualification and seismic concerns related to the BTI at Salem Units 1 and 2.

4.1.2 GDC 19 – Control Room

GDC 19 states that "A control room shall be provided from which actions can be taken to operate the nuclear power plant safely under normal conditions and to maintain it in a safe condition under accident conditions." This criterion is applicable to the installation of the BTI at Salem Units 1 and 2 because adequate indication and annunciation of the status of the protection system channels (i.e., normal, bypasses, or tripped) must be available to the operators. The BTI has been designed to meet this criterion by providing the operator as well as the test technician with accurate information concerning the status of

the channels being tested. Section 3.6 describes the indication and annunciation design features of the BTI at Salem Units 1 and 2 and its conformance to this criterion.

4.1.3 GDC 20 – Protection System Functions

GDC 20 states "The protection system shall be designed to initiate automatically the operation of appropriate systems including the reactivity control systems, to assure that specified acceptable fuel design limits are not exceeded..." This criterion is applicable to the installation of the BTI at Salem Units 1 and 2 because the protection system must still be able to perform its function after the installation of the BTI. When the NIS BTI is not powered, it is not within the protection system circuitry (i.e., no protection system signals pass through the BTI). Isolation devices are being used as isolators between Class 1E and non-Class 1E circuits. A complete discussion of the administrative control and operator actions to ensure conformance to this criterion are discussed in Sections 3.4 and 3.7, respectively.

4.1.4 GDC 21 – Protection System Reliability and Testability

GDC 21 states "The protection system shall be designed for high functional reliability and in service testability commensurate with the safety function to be performed. Redundancy and independence designed into the protection system shall be sufficient to assure that no single failure results in loss of the protection function..." This Criterion is applicable to the installation of the BTI at Salem Units 1 and 2 because the BTI design must show sufficient reliability to ensure that a single failure will not cause the protection system to be unable to perform its function. A complete discussion of the conformance of the installation of the BTI to the single failure criterion is contained in subsection 4.2.2.

4.1.5 GDC 22 – Protection System Independence

GDC 22 states "The protection system shall be designed to assure that the effects of natural phenomena and of normal operating, maintenance, testing, and postulated accident conditions on redundant channels do not result in the loss of the protection function, or shall be demonstrated to be acceptable on some other defined basis." This criterion is applicable to the installation of the BTI because the ability exists, without the proper administrative controls, for the simultaneous bypassing of more than one protection set at a time. Section 3.4 discusses the administrative controls that prevent the bypassing of more than one protection set at a time and thus conformance to this criterion.

4.1.6 GDC 23 – Protection System Failure Modes

GDC states "The protection system shall be designed to fail into a safe state... if conditions such as disconnection of the system, loss of energy (e.g., electric power, instrument air) or postulated adverse environments are experienced." This criterion is applicable to the installation of the BTI at Salem Units 1 and 2 because a failure mode of the BTI is the loss of power to the bypass system. Loss of power, due to either a circuit breaker opening or loss of power to the cabinet, will cause the bypass system to terminate any bypassing that was being performed. The bypass systems will return to their normal operating mode. These results demonstrate conformance to this criterion.

4.1.7 GDC 24 – Separation of Protection and Control Systems

GDC 24 states that "The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection system leaves intact a system satisfying all the reliability, redundancy, and independence requirements of the protection system." This criterion is applicable to the installation of the BTI at Salem Units 1 and 2 because the indication and annunciation of the status of the channels in bypass are part of the control system. Subsection 4.2.3 and Section 5.6 (within Section 4.3.1) discuss the BTI conformance to RG 1.75 and IEEE Standard 603-1991, respectively, as pertinent to separation and isolation requirements.

4.2 RGs

The following RGs are applicable to the installation of the BTI:

- RG 1.47, Rev. 1 – "Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems"
- RG 1.53, Rev. 2 – "Application of Single Failure Criterion to Nuclear Power Plant Protection Systems"
- RG 1.75, Rev. 3 – "Physical Independence of Electric Systems"
- RG 1.89, Rev. 1 – "Qualification of Class 1E Equipment for Nuclear Power Plants"
- RG 1.100, Rev. 3 – "Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants"
- RG 1.118, Rev. 3 – "Periodic Testing of Electric Power and Protection Systems"
- RG 1.22, Rev. 0 – "Periodic Testing Of Protection System Actuation Functions"
- RG 1.30, Rev. 0 – "Quality Assurance Requirements for the Installation Inspection, and Testing Instrumentation and Electric Equipment"

4.2.1 RG 1.47, Rev. 1 – Bypassed and Inoperable Status Indication for Nuclear Power Plant Safety Systems

RG 1.47 describes an acceptable method of complying with the requirements of IEEE Standard 279-1971and IEEE Standard 603-1991, and states that automatic indication should be provided in the control room for each bypass or deliberately induced inoperable status that meets all of the following conditions:

a. Renders inoperable any redundant portion of the protection system, systems actuated or controlled by the protection system, and auxiliary or supporting systems that must be operable for the protection system and the system it actuates to perform their safety related functions.

b. Expected to occur more frequently than once per year.

c. Expected to occur when the affected system is normally required to be operable.

The BTI meets all of these conditions. By placing a protection system channel in the bypass mode, that channel of the protection system is rendered inoperable. For any channel that is placed in the bypass mode, an automatic annunciation is initiated in the main control room. Section 3.6 discussed how the BTI conforms to this RG along with the detailed responses to each of the regulatory positions listed below.

Regulatory positions:

1. *Administrative procedures should be supplemented by an indication system that automatically indicates, for each affected safety system or subsystem, the bypass or deliberately induced inoperability of a safety function and the systems actuated or controlled by the safety function. Provisions should also be made to allow the operations staff to confirm that a bypassed safety function has been properly returned to service.*

   The BTI hardware automatically indicates through the annunciator signals that the channel has been placed in bypass. In addition, PSEG Nuclear will have administrative procedures in place to alert the operator when a channel is in bypass.

2. *The indicating system of Position 1 above should also be activated automatically by the bypassing or the deliberately induced inoperability of any auxiliary or supporting system that effectively bypasses or renders inoperable a safety function and the systems actuated or controlled by the safety function.*

   Annunciator signals are provided by the BTI panel. As part of the installation, the site will connect the annunciator signal wires to the main control board (MCB) annunciator panel. These annunciator signals are provided automatically once the BTI panel is switched to bypass mode.

3. *Annunciating functions for system failure and automatic actions based on the self-test or self-diagnostic capabilities of digital computer-based I&C safety systems should be consistent with Positions 1 and 2 above.*

   The BTI panel does not contain any self-test or self-diagnostic capabilities. Furthermore, the BTI panel does not contain any digital computer-based components.

4. *The bypass and inoperable status indication system should include a capability for ensuring its operable status during normal plant operation to the extent that the indicating and annunciating functions can be verified.*

   Status indication is provided locally on the front panel by LEDs and through the MCB annunciator panel.

5. *Bypass and inoperable status indicators should be arranged such that the operator can determine whether continued reactor operation is permissible. The control room of all affected units should receive an indication of the bypass of shared system safety functions.*

   Bypass status visible through the MCB annunciator panel provides the operator clear and unambiguous indication of the status of particular channel. PSEG Nuclear will position the panel such that it is visible and unobstructed to the operator. Also see the discussions to regulatory positions 2 and 4 above.

6. *Bypass and inoperable status indicators should be designed and installed in a manner that precludes the possibility of adverse effects on plant safety systems. The indication system should not be used to perform functions that are essential to safety, unless it is designed in conformance with criteria established for safety systems.*

   The BTI front panel indication and MCB annunciator panel are only designed and used for indication purposes. They do not perform any other function.

4.2.2 RG 1.53, Rev. 2 – Application of Single Failure Criterion to Nuclear Power Plant Protection Systems

RG 1.53 endorses IEEE Standard 379-2000 with some clarification. IEEE Standard 379-2000 addresses the single failure criterion in nuclear power plant protection systems. A discussion of the BTI adherence to IEEE Standard 379-2000, this RG and the single failure criterion in general is contained in Section 4.3.

A single hardware failure of the BTI hardware cannot cause a channel to inadvertently go into bypass. The BTI design only includes analog and passive components. The BTI design does not include any microprocessors or digital hardware.

Channel bypass through the BTI panel is accomplished by following these four steps:

1. Main breaker is switched on.
2. Appropriate key is provided to the technician.
3. Key is inserted and turned on the BTI front panel.
4. BTI front panel switch is enabled.

For a channel to inadvertently stay in the bypass mode, the main breaker, the key switch and the toggle switch would all have to fail at the same time. In addition the MCB annunciator panel and the local BTI panel, a LED must also fail for the operators not to identify that a failure has occurred.

4.2.3 RG 1.75, Rev. 3 – Physical Independence of Electric Systems

RG 1.75 endorses and delineates acceptable methods for complying with the requirements of IEEE Standard 279-1971 with respect to physical independence of electric systems.

RG 1.75 discusses requirements for physical separation between Class 1E and non-Class 1E circuits, electrical isolation between Class 1E and non-Class 1E circuits, and requirements for associated circuits.

The BTI is a safety-related assembly. Furthermore, the cabinet that contains the BTI panel and the NIS drawers is considered a safety related cabinet with only safety related power applied to the panels and drawers. It should be noted that there are non-safety related signals (MCB annunciator) exiting the BTI panel, which are isolated from the safety related signals and hardware by safety related hardware such as the K1 relays. Section 5.6 (within subsection 4.3.1) discusses the BTI conformance to IEEE Standard 603-1991, as pertinent to separation and isolation requirements.

4.2.4 RG 1.89, Rev. 1 – Qualification of Class 1E Equipment for Nuclear Power Plants

RG 1.89 endorses IEEE Standard 323-1974. A discussion of the BTI conformance to the requirements of IEEE Standard 323-1974 and this RG is contained in Section 4.3.

Regulatory positions:

1. *Section 50.49, "Environmental Qualification of Electric Equipment Important to Safety for Nuclear Power Plants," of 10 CFR Part 50 requires that safety related electric equipment (Class lE) as defined in paragraph 50.49(b)(1) be qualified to perform its intended safety functions.*

   The BTI generic seismic qualification and environmental evaluation was performed in accordance with WCAP-8587 (Reference 6).

2. *Paragraph 50.49(d) and Section 6.2 of IEEE Std 323-1974 require equipment specifications to include performance and environmental conditions. For the requirements called for in item (7) of Section 6.2 of IEEE 323-1974 and paragraph 50.49(d)(3), the following should be included:*

   a. Temperature and Pressure Conditions Inside Containment for LOCA and Main Steam Line Break (MSLB).

   b. Effects of Sprays and Chemicals.

   c. Radiation Conditions Inside and Outside Containment.

   d. Environmental Conditions for Equipment Outside Containment.

   Items 2a, 2b and 2c are not applicable to PSEG Nuclear BTI equipment due the location of the hardware. Item 2d is addressed because the BTI panels are located in the control room environment, which is considered to be a mild environment as described in Reference 5.

3. *Section 6.3, "Type Test Procedures," of IEEE Std. 323-1974 should be supplemented with the following:*

   a. *Electric equipment that could be submerged should be identified and qualified by testing in a submerged condition to demonstrate operability for the duration required.*

   b. *Electric equipment located in an area where rapid pressure changes are postulated simultaneously with the most adverse relative humidity should be qualified to demonstrate that the equipment seals and vapor barriers will prevent moisture from penetrating into the equipment to the degree necessary to maintain equipment functionability.*

   c. *The parameters to which electric equipment is being qualified (e.g., temperature, pressure, radiation) by exposure to a simulated environment in a test chamber should be measured sufficiently close to the equipment to ensure that actual test conditions accurately represent the environment characterized by the test.*

   d. *Performance characteristics that demonstrate the operability of equipment should be verified before, after, and periodically during testing throughout its range of required operability.*

   e. *Chemical spray or dermineralized water spray that is representative of service conditions should be incorporated during simulated event testing at pressure and temperature conditions that would occur when the spray systems actuate.*

   f. *Cobalt-60 or cesium-137 would be acceptable gamma radiation sources for environmental qualification.*

   These regulatory positions are not applicable because the BTI panel is located in the control room environment, which is considered to be a mild environment as discussed in Reference 5. The BTI panel does not have any requirements related to a harsh environment as listed in position 3.

4. *The suggested values in Section 6.3.1.5, "Margin," of IEEE Std. 323-1974, except time margins, are acceptable for meeting the requirements of paragraph 50.49(e)(8). Alternatively, quantified margins should be applied to the environmental parameters discussed in Regulatory Position C.2 to ensure that the postulated accident conditions have been enveloped during testing.*

   The BTI generic seismic qualification and environmental evaluation bounds the PSEG Nuclear current licensing basis, and was performed in accordance with WCAP-8587 (Reference 6).

5. *Section 6.3.3, "Aging," of IEEE Std 323-1974 and paragraph 50.49(e)(5) should be supplemented with the following:*

   a. *If synergistic effects have been identified prior to the initiation of qualification, they should be accounted for in the qualification program. Synergistic effects known at this time are dose rate effects and effects resulting from the different sequence of applying radiation and (elevated) temperature.*

   b. *The expected operating temperature of the equipment under service conditions should be accounted for in thermal aging. The Arrhenius methodology is considered an acceptable method of addressing accelerated thermal aging within the limitation of state-of-the-art technology. Other aging methods will be evaluated on a case-by-case basis.*

   c. *The aging acceleration rate and activation energies used during qualification testing and the basis upon which the rate and activation energy were established should be defined, justified, and documented.*

   d. *Periodic surveillance and testing programs are acceptable to account for uncertainties regarding age related degradation that could affect the functional capability of equipment. Results of such programs will be acceptable as ongoing qualification to modify designated life (or qualified life) of equipment and should be incorporated into the maintenance and refurbishment/ replacement schedules.*

   Aging testing was not performed for the BTI panel because the BTI hardware is not part of the reactor trip system; the BTI is only used for the testing of a channel in bypass. During normal system operation, the BTI hardware is de-energized. Additionally, a failure of the BTI panel will not prevent a reactor trip from occurring.

6. *Replacement electric equipment installed subsequent to February 22, 1983, must be qualified in accordance with the provisions of § 50.49 unless there are sound reasons to the contrary.*

   Future replacement components for the BTI panel will be commercially dedicated in accordance with the original qualification requirements. The commercial dedications will address the critical characteristics required to maintain the qualification of the equipment.

7. *In addition to the requirements of paragraph 50.49(j) of 10 CFR Part 50 and Section 8, "Documentation," of IEEE Std. 323-1974, documentation should address the information identified in Appendix E to this guide. A record of the qualification should be maintained in an auditable file to permit verification that each item of electric equipment is qualified to perform its safety function under its postulated environmental conditions throughout its installed life.*

   This equipment has been qualified in accordance with WCAP-8587 (Reference 6), and the test results are documented in WCAP-8687 (Reference 5) which have been approved by the NRC (Reference 6).

4.2.5 RG 1.100, Rev. 3 – Seismic Qualification of Electric and Mechanical Equipment for Nuclear Power Plants

RG 1.100 endorses IEEE Standard 344-2004 and previous revisions of the Standard. A discussion of the BTI conformance to the IEEE Standard 344-1975 and this RG are discussed in Section 4.3. The BTI panel was seismically qualified per IEEE 344-1975; therefore the differences with the current IEEE 344-2004 were evaluated for compliance. The two differences between IEEE 344-1975 and IEEE 344-2004 listed below do not invalidate the previous seismic qualification.

1. Use of experience data – This is not applicable because experience data was not used since the BTI panel was seismically tested.

2. High frequency ground motion for hard-rock-based plants – The Salem Units 1 and 2 are not considered a hard-rock-based plant.

4.2.6 RG 1.118, Rev. 3 – Periodic Testing of Electric Power and Protection Systems

RG 1.118 endorses IEEE Standard 338-1987 for periodic testing of protection systems subject to providing a method of preventing the expansion of any bypass condition to redundant channels. This is accomplished by administrative control of access to the bypass capability. The latest version of the RG 1.118 was compared to the previous version and the following changes were evaluated.

Position on 2.3 and 3

2.3 Test procedures or administrative controls shall provide for verifying the open circuit or verifying that temporary connections are restored after testing.

PSEG Nuclear will have administrative controls in place to perform periodic surveillance testing when using the BTI Panel.

3. The description for a logic system functional test, as noted in Section 6.3.5 of IEEE Std. 338-1987, implies that the sensor is included. A logic system functional test is to be a test of all logic components (i.e., all relays and contacts, trip units, solid state logic elements, etc.) of a logic circuit, from as close to the sensor as practicable up to but not including the actuated device, to verify operability.

The in-containment sensors (PR ex-core detectors) are not directly connected to the BTI panel. Within a protection channel, the BTI panel is located between the NIS drawers and SSPS; therefore, sensor testing is not applicable to the BTI panel. The BTI panel does not interfere with the required testing of any instrument channel components. The BTI panel only provides a method of testing the instrument channel components in bypass mode.

4.2.7 RG 1.22, Rev. 0 – Periodic Testing Of Protection System Actuation Functions

The following section discusses in detail how the BTI panel complies with RG 1.22.

1. *The protection system should be designed to permit periodic testing to extend to and include the actuation devices and actuated equipment.*

    a. *The periodic tests should duplicate, as closely as practicable, the performance that is required of the actuation devices in the event of an accident.*

    The BTI panel provides a preferred means for periodic testing because it allows testing a channel in bypass versus trip, and reduces the potential for a spurious reactor trip during testing. When the entire protection channel is being tested, the trip signal simply passes through the BTI panel.

    b. *The protection system and the systems whose operation it initiates should be designed to permit testing of the actuation devices during reactor operation.*

    The BTI panel provides a means of verifying that the 120 V signal on the terminal blocks, which are located in the same cabinet. If the 120V signal is not present in the terminal blocks, the SSPS would alarm depending on the BTI state. Within a protection channel, the BTI panel is located between the NIS drawers and SSPS; therefore, the BTI panels do not directly interface with the actuation device. The BTI panel does not interfere with the testing of the actuation devices, and the required testing of the actuation devices will not be affected by the BTI Panel.

2. *Acceptable methods of including the actuation devices in the periodic tests of the protection system.*

    This is not applicable to the BTI panel since the actuation device (such as the reactor trip breaker) is not connected to the BTI panel directly. The BTI panel is connected directly to the SSPS. The BTI panel does not interfere with the testing of the actuation devices, and the required testing of the actuation devices will not be affected by the BTI Panel

3. *Where the ability of a system to respond to a bona fide accident signal is intentionally bypassed for the purpose of performing a test during reactor operation:*

    a. *Positive means should be provided to prevent expansion of the bypass condition to redundant or diverse systems, and*

    b. *Each bypass condition should be individually and automatically indicated to the reactor operator in the main control room.*

    Each channel is isolated from each other; therefore placing one channel in bypass cannot result in placing any other channel in bypass. The BTI panels have individual and automatic annunciation. Refer to Section 3.4 for details.

4. Where actuated equipment is not tested during reactor operation, it should be shown that:

a. There is no practicable system design that would permit operation of the actuated equipment without adversely affecting the safety or operability of the plant;

b. The probability that the protection system will fail to initiate the operation of the actuated equipment is, and can be maintained, acceptably low without testing the actuated equipment during reactor operation, and

c. The actuated equipment can be routinely tested when the reactor is shut down.

This is not applicable to the BTI panel because the actuated equipment is not directly connected to it. The BTI panel does not interfere with the testing of the actuation devices and the required testing of the actuation devices will not be affected by the BTI Panel.

4.2.8 RG 1.30, Rev. 0 – Quality Assurance Requirements for the Installation, Inspection, and Testing of Instrumentation and Electric Equipment

PSEG Nuclear will comply with RG 1.30 for the installation, inspection or maintenance testing. The factory acceptance test (FAT) will be performed at a 10CFR Part 21, Appendix B facility before shipping the hardware to the site.

4.3 IEEE STANDARDS

The following IEEE Standards are applicable to the BTI panel at Salem Units 1 and 2 and are discussed in subsections 4.3.1 through 4.3.6:

- IEEE 603-1991 – IEEE Standard: Criteria for Protection Systems for Nuclear Power Generating Stations
- IEEE 379-2000 – Trial Use Guide for the Application of the Single Failure Criteria to Nuclear Power Generating Station Protection Systems
- IEEE 384-1974 – Trial Use Standard for Separation of Class 1E Equipment and Circuits
- IEEE 344-2004 – IEEE Recommended Practices for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations
- IEEE 338-1987 – IEEE Standard Criteria for the Periodic Testing of Nuclear Power Generating Station Class 1E Power and Protection Systems
- IEEE 323-1974 – IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations

4.3.1 IEEE Standard 603-1991 has several sections which are applicable to the BTI installation at Salem Units 1 and 2. The sections that are applicable are:

Section 5.1 – Single Failure Criterion

This section requires that any single failure in the protection system shall not prevent proper protective action at the system level when required. A discussion of the potential fault conditions and failure detection of the BTI are discussed in Sections 3.2 and 3.3, respectively.

Any postulated failure in the bypass systems that would inadvertently cause the channel in bypass to trip are failures in a safe direction, i.e., tripped conditions and will not be discussed. Failures in the bypass systems that need to be addressed are those that could potentially:

1. Cause a channel to go into the bypass condition inadvertently.
2. Cause a channel to fail to exit the bypass condition when indications show otherwise.

All of these types of failures could cause the same result. That is, the possibility could exist for more than one redundant protection set to be in bypass at the same time such that a reactor trip may not be generated. It would require several contacts to spuriously close on the NIS bypass system to cause an inadvertent bypass. For a channel to fail to come out of bypass while indicating that it has returned to normal, one contact would have to stick closed in the associated relay. These failures would all be detected by observation of the local bypass status lights. Thus, there is no credible single failure of the BTI that could result in the protection system being degraded to the point of being unable to perform its intended safety function.

Section 5.3 – Quality of Components

This section requires that components and modules be of a high quality. The components used in the BTI are of a quality consistent with minimum maintenance requirements and low failure rates. The quality of the components that are used in the BTI is consistent with components used in the protection system. All of the components are mechanical or electro-mechanical and are reliable through at least 50,000 operations under normal environmental conditions.

Section 5.4 – Equipment Qualification

This section requires that type test data, or reasonable engineering extrapolation based on test data, be available to verify that protection system equipment meet the performance requirements. Generic tests were conducted to verify that the NIS bypass panels that are located in Class 1E instrument cabinets will not go into one of the failure modes identified during a seismic event. The tests were run to show structural integrity and electrical isolation where applicable. A complete discussion of the equipment qualification of the BTI is contained in Section 3.8.

Section 5.6 – Control and Protection System Interaction

Each BTI panel is located within its own protection set; therefore, a single fault would not cause a problem in redundant channels. The components of the NIS BTI panels that are non-Class 1E are isolated from the Class 1E circuits by qualified isolators. Therefore, there is no possibility that a control system fault could propagate to all the bypass panels and simultaneously adversely affect all protection sets. Separation requirements are maintained in the NIS bypass panels through physical separation on the bottom lid of the bypass panel with 6 inches between the safety related and non-safety related 118 VAC. The circuit board maintains this required separation by placing a ground layer between the safety related and non-safety related 118 VAC circuits.

Section 5.7 – Capability for Test and Calibration

The BTI panel complies with this section by providing the capability to functionally test its operation, and it does not require any calibration.

Section 5.8 – Information Displays

5.8.3.1: The annunciator is not part of the safety system.

5.8.3.2: The annunciator signal is provided automatically once the BTI keylock is engaged.

5.8.3.3: The keylock switch is located in the BTI front panel, which is located in the control room.

Section 5.8.3 – Indication of Bypasses

This section requires that for a protective function that has been deliberately bypassed, indication/annunciation of the bypass must be continuously displayed in the control room. The design of the BTI at Salem Units 1 and 2 provides local alarm/status light and annunciators in the control room when a channel is bypassed.

Section 5.9 – Control of Access

This section requires that the BTI design permit administrative control of the means for bypassing channels or protective functions. The design of the BTI for Salem Units 1 and 2 requires the use of the NIS keylock switches for placing a channel in bypass. Administrative control can be implemented by proper control of the distribution of the keys for the NIS panels.

Section 6.6 – Channel Bypass or Removal from Operation

The BTI panel for Salem Units 1 and 2 will not affect the compliance of the protection system to this section. When one channel is bypassed for testing, there will still be sufficient channels available to trip the reactor. The protection system will continue to conform to this section.

Section 6.3 – Interaction Between the Sense and Command Features and Other Systems

Based on Figure 5 (Interpretation of 6.3.1 of IEEE Std 603-1991) from Section 6.3.1 of IEEE 603-1991, the response for the BTI panel with respect to "Does event by itself result in condition requiring safety function" and "Does event cause action by a non-safety system," is No, therefore the criteria of 6.3.1 (1) and 6.3.2 (2) are not applicable. Furthermore, the BTI panel is not directly connected to a field sensor. None of the actions listed in Section 6.3.1 can impact the BTI hardware. The BTI panel does not have the capability to receive an input from a field sensor or a process variable nor does it have the capability to generate a command signal.

Section 6.6 – Operating Bypasses

If one channel is in bypass the protection system can still perform its function, since 2-out-of-3 coincidence logic is still in place.

Section 6.7 – Maintenance Bypass

As described in IEEE 603 Section 6.3 above, the BTI panel does not interface with field sensors or process variables and it does not generate a command signal.

Section 4.20 – Information Read-out

This section requires that the protection system be designed to provide the operator with information pertaining to its own status and the status of the plant. Section 3.6 discusses the annunciation features of the BTI and conformance to this section.

4.3.2 IEEE Standard 379-2000

IEEE Standard 379-2000 describes the application of the single failure criterion to the protection system. The most limiting single failure would be one that would cause a channel to remain in bypass while indicating to the technician and the control room operator that the channel has been removed from bypass. Another redundant channel could then be placed in bypass and there would be two redundant channels in bypass simultaneously. A failure of any component in the bypass system that inadvertently causes a channel to trip is a failure in the conservative direction, and would not prevent that channel from performing its function. There is no credible single failure that could inadvertently place a channel of the protection system into the bypass condition. Power is provided to the NIS bypass panel only when the circuit breaker is closed and the keylock switch is turned from "NORMAL" to "BYPASS ENABLE", and the individual bypass toggle switch is placed in bypass. No single failure could inadvertently provide power to the bypass panel. Furthermore, the BTI panels among the four channels do not share components or functions.

4.3.3 IEEE Standard 384-1974

IEEE Standard 384-1974 discusses the separation requirements for Class 1E circuits and equipment. These separation requirements are required when Class 1E and non-Class 1E equipment is located within close proximity to one another. The information provided in this standard and in Regulatory Guide 1.75 are similar and also support the separation requirements contained in IEEE Standard 603-1991, Section 5.6.

4.3.4 IEEE Standard 344-2004

IEEE Standard 344-2004 discusses the recommended practices for performing seismic qualification of Class 1E equipment. The BTI, since it is being installed in Class 1E instrument racks, must be shown to be seismically qualified. Section 3.8 discusses the generic seismic qualification of the BTI for Salem Units 1 and 2.

The BTI panel was seismically qualified per IEEE 344-1975, therefore the differences with the current IEEE 344-2004 were evaluated for compliance. The two differences between IEEE 344-1975 and IEEE 344-2004 listed below do not invalidate the previous seismic qualification.

1. Use of experience data – This is not applicable because experience data was not used since the BTI panel was seismically tested.

2. High frequency ground motion for hard-rock-based plants – Salem Units 1 and 2 are not considered a hard-rock-based plant.

4.3.5 IEEE Standard 338-1987

IEEE Standard 338-1987 discusses the criteria for performing periodic testing of safety systems. Installation of the BTI does not impact the capability for performing periodic tests that was originally designed into the equipment. The BTI panel provides an alternative means of testing in bypass rather than in a tripped condition.

This IEEE Standard applies to the Nuclear Instrumentation detectors and drawers. This standard provides guidance for the periodic testing of safety related systems. The BTI panel does not perform a safety function when installed in the NIS cabinet. The BTI panel does not have any impact on the capability to perform the required testing as discussed in the IEEE Standard. The BTI panel only allows the drawers to be tested in bypass. It does not change or influence the method of surveillance testing of the drawers. Thus, the installation of the BTI panels has no effect on the NIS system compliance with this IEEE standard.

4.3.6 IEEE Standard 323-1974

IEEE Standard 323-1974 discusses the requirements for qualifying Class 1E equipment for nuclear power plants. Section 3.8 discusses the equipment qualification and conformance of the BTI.

The BTI generic seismic qualification and environmental evaluation bounds the PSEG Nuclear current licensing basis, and was performed in accordance with WCAP-8587 (Reference 6).

5 CONCLUSION

Various aspects of the NIS PR BTI panel are addressed in this TR. These aspects include a demonstration of the functionality of the BTI hardware, and the design features of the BTI panel to demonstrate compliance with the applicable regulations, RGs and IEEE Standards associated with testing in bypass.

The NIS PR BTI panel will reduce the potential for spurious reactor trips, which reduces the potential transients associated with them, and ensures that the NIS PR RTS functions remain capable of performing their specified safety function.

The following tables and figures provide additional details of the NIS PR BTI System. Table 5.1 lists the comparators that can be bypassed and Table 5-2 provides the BTI panel part numbers. Figure 5-1 provides a basic functional diagram to illustrate the operation of a BTI panel.

Table 5-1. NIS Comparators to by Bypassed

Function	Protection Set			
	I	II	III	IV
Power Range – High Flux Reactor Trip (Low Setpoint)	1	1	1	1
Power Range – High Flux Reactor Trip (High Setpoint)	1	1	1	1
Power Range – Overpower Rod Stop C-2	1	1	1	1
Power Range – P-10 Permissive	1	1	1	1
Power Range – P-8 Permissive	1	1	1	1
Power Range – P-9 Permissive	1	1	1	1
Power Range – High Flux Positive Rate Reactor Trip	1	1	1	1
SPARE	1	1	1	1

Table 5-2. NIS BTI Panel Part Numbers

Panel Part Number	Description
10105D37G01	NIS Bypass Panel Assembly (Channel 1, Salem Unit 1)
10105D37G02	NIS Bypass Panel Assembly (Channel 2, Salem Unit 1)
10105D37G03	NIS Bypass Panel Assembly (Channel 3, Salem Unit 1)
10105D37G04	NIS Bypass Panel Assembly (Channel 4, Salem Unit 1)
10105D37G05	NIS Bypass Panel Assembly (Channel 1, Salem Unit 2)
10105D37G06	NIS Bypass Panel Assembly (Channel 2, Salem Unit 2)
10105D37G07	NIS Bypass Panel Assembly (Channel 3, Salem Unit 2)
10105D37G08	NIS Bypass Panel Assembly (Channel 4, Salem Unit 2)

a,c

Figure 5-1. NIS Bypass Panel Diagram

6 REFERENCES

1. Westinghouse Document, WCAP-10271-P-A, Rev. 0, “Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System,” May 1986.

2. Westinghouse Document, WCAP-10271, Supplement 1-P-A, Rev. 0, “Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System Supplement 1,” May 1986.

3. Westinghouse Document, WCAP-10271-P-A, Supplement 2, Rev. 1, “Evaluation of Surveillance Frequencies and Out of Service Times for the Reactor Protection Instrumentation System,” June 1990.

4. Westinghouse Document, WCAP-14333-P-A, Rev. 1, “Probabilistic Risk Analysis of the RPS and ESFAS Test Times and Completion Times,” October 1998.

5. Westinghouse Document, WCAP-8687, Supplement 2, EQTR-E47G, Rev. 1, “Equipment Qualification Test Report Nuclear Instrumentation System (NIS) Bypass Test Instrumentation Panel (Seismic Testing),” March 1996.

6. Westinghouse Document, WCAP-8587, Rev. 6-A, “Methodology for Qualifying Westinghouse WRD Supplied NSSS Safety Related Electrical Equipment,” March 1983.