AmerGen

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An Exelon Company

10 CFR 50.90

October 20, 2004
5928-04-20065

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

Three Mile Island, Unit 1 (TMI Unit 1)
Facility Operating License No. DPR-50
NRC Docket No. 50-289

Subject: Technical Specification Change Request No. 324 – Reactor Protection System Test Interval Extension

In accordance with 10 CFR 50.4(b)(1), enclosed is Technical Specification Change Request (TSCR) No. 324.

The purpose of this TSCR is to revise the TMI Unit 1 Technical Specification requirements for reactor protection system instrumentation channel functional testing from a monthly interval to a semi-annual interval in accordance with NRC approved Topical Report BAW-10167, "Justification for Increasing the Reactor Trip System On-Line Test Interval," dated May 1986, BAW-10167A, Supplement 1, dated August 1992, and NRC Safety Evaluation Report (SER) for BAW-10167A, Supplement 1, dated December 5, 1988. This TSCR also revises the TMI Unit 1 Technical Specification requirements for functional testing of reactor protection system control rod drive trip breakers, protective channel coincidence logic, and electronic trip relays from a monthly interval to a quarterly interval in accordance with NRC approved Topical Report BAW-10167A, Supplement 3, dated February 1998, and NRC SER for BAW-10167A, Supplement 3, dated January 7, 1998.

The proposed amendment has been reviewed by the TMI Unit 1 Plant Operations Review Committee and approved by the Nuclear Safety Review Board in accordance with the requirements of the AmerGen Quality Assurance Program.

Using the standards in 10 CFR 50.92, AmerGen Energy Company, LLC (AmerGen) has concluded that these proposed changes do not constitute a significant hazards consideration, as described in the enclosed analysis performed in accordance with 10 CFR 50.91(a)(1). Pursuant to 10 CFR 50.91(b)(1), a copy of this Technical Specification Change Request is provided to the designated official of the Commonwealth of Pennsylvania, Bureau of Radiation Protection, as well as the chief executives of the township and county in which the facility is located.

NRC approval of this change is requested by October 20, 2005. Once approved, the amendment shall be implemented within 60 days.

Regulatory commitments as a result of this submittal are identified in Enclosure 3. If any additional information is needed, please contact David J. Distel at (610) 765-5517.

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

Sincerely,

10/20/04
Executed On

B C Williams
Bruce C. Williams
Vice President, TMI Unit 1
AmerGen Energy Company, LLC

Enclosures: (1) TMI Unit 1 Technical Specification Change Request No. 324 Evaluation of Proposed Changes
(2) TMI Unit 1 Technical Specification Change Request No. 324 Markup of Proposed Technical Specification Page Changes
(3) Regulatory Commitments

cc: S. J. Collins, Administrator, USNRC Region I
T. G. Colburn, USNRC Senior Project Manager (Acting), TMI Unit 1
D. M. Kern, USNRC Senior Resident Inspector, TMI Unit 1
D. Allard, Director, Bureau of Radiation Protection – PA Department of Environmental Resources
Chairman, Board of County Commissioners of Dauphin County
Chairman, Board of Supervisors of Londonderry Township
File No. 04071

ENCLOSURE 1

TMI Unit 1 Technical Specification Change Request No. 324

Evaluation of Proposed Changes

1.0 INTRODUCTION

This letter is a request to amend Operating License No. DPR-50.

The proposed changes would revise the Operating License to revise the TMI Unit 1 Technical Specification requirements for: (1) reactor protection system instrumentation channel functional testing from a monthly interval to a semi-annual interval in accordance with NRC approved Topical Report BAW-10167, "Justification for Increasing the Reactor Trip System On-Line Test Interval," dated May 1986, and NRC Safety Evaluation Report (SER) for BAW-10167A, Supplement 1, dated December 5, 1988, and (2) functional testing of reactor protection system control rod drive trip breakers, protective channel coincidence logic, and electronic trip relays from a monthly interval to a quarterly interval in accordance with NRC approved Topical Report BAW-10167A, Supplement 3, dated February 1998, and NRC SER for BAW-10167A, Supplement 3, dated January 7, 1998.

NRC approval of this change is requested by October 20, 2005.

AmerGen Energy Company, LLC (AmerGen) requests that the following changed replacement pages be inserted into the existing Technical Specifications:

Revised Technical Specification Pages: 4-2a, 4-2b, 4-3, 4-4, and 4-7.

The marked up pages showing the requested changes are provided in Enclosure 2.

2.0 DESCRIPTION OF PROPOSED AMENDMENT

TMI Unit 1 Technical Specification Table 4.1-1 specifies the instrument surveillance requirements for reactor protection system instrumentation channel functional testing and reactor protection system control rod drive trip breakers, protective channel coincidence logic, and electronic trip relay functional testing.

The proposed change revises the Technical Specification Table 4.1-1 functional testing surveillance interval from monthly to semi-annually for the following reactor protection system instrument channels: Table 4.1-1, Item No. 4 Power Range Channel, Item No. 7 Reactor Coolant Temperature Channel, Item No. 8 High Reactor Coolant Pressure Channel, Item No. 9 Low Reactor Coolant Pressure Channel, Item No. 10 Flux-Reactor Coolant Flow Comparator, Item No. 11 Reactor Coolant Pressure-Temperature Comparator, Item No. 12 Pump Flux Comparator, Item No. 13 High Reactor Building Pressure Channel, Item No. 45 Loss of Feedwater Reactor Trip, and Item No. 46 Turbine Trip/Reactor Trip. The Technical Specification Section 4.1 Bases is revised to reflect the proposed change from monthly testing to semi-annual testing and to specify that one channel is being tested every 46 days on a continual sequential rotation, which is consistent with the calculations of BAW-10167A, Supplement 1 and associated NRC SER that indicate the reactor protection system retains a high level of reliability for this test interval.

The proposed change also revises the Technical Specification Table 4.1-1 functional testing surveillance interval from monthly to quarterly for the following reactor protection system reactor trip devices: Table 4.1-1 Item No. 1 Protection Channel Coincidence Logic, and Item No. 2 Control Rod Drive Trip Breaker and Regulating Rod Power SCRs. The Technical Specification Section 4.1 Bases is revised to reflect the proposed change from monthly testing to quarterly and to specify that one channel is being tested every 23 days on a continual sequential rotation, which is consistent with the calculations of BAW-10167A, Supplement 3, February 1998, and the NRC SER for BAW-10167A, Supplement 3, dated January 7, 1998, that indicate the reactor trip system retains a high level of reliability for this test interval.

In summary, the proposed change revises the reactor protection system and reactor trip system channel functional test interval from monthly to semi-annually and quarterly, respectively, consistent with the NRC SER's for Topical Report BAW-10167A, Supplement 1, August 1992 and BAW-10167A, Supplement 3, February 1998. NRC NUREG-1430, Babcock and Wilcox Plants, Revision 2, Standard Technical Specifications incorporate the proposed reactor protection system surveillance interval extensions as justified by BAW-10167. The proposed change is essentially consistent with the Standard Technical Specifications.

3.0 BACKGROUND

The reactor protection system monitors parameters related to safe operation and trips the reactor to protect the reactor core against fuel rod cladding damage. It also assists in protecting against reactor coolant system damage caused by high system pressure by limiting energy input to the system through reactor trip action. The system, as described in the TMI Unit 1 Updated Final Safety Analysis Report (UFSAR), Section 7.1, consists of four identical protection channels, each terminating in a trip relay within a reactor trip module. In the normal untripped state, each protection channel functions as an AND gate, passing current to the terminating relay and holding it energized as long as all inputs are in the normal energized (untripped) state. Should any one or more inputs become denergized (tripped), the terminating relay in that protective channel deenergizes (trips).

Each protective channel trip relay has four contacts, each controlling a logic relay in one reactor trip module. Therefore, each reactor trip module has four logic relays controlled by the four protection channels. The four logic relays combine to form a 2 out of 4 coincidence network in each reactor trip module. The coincidence logics in all reactor trip modules trip whenever any two of the four protection channels trip. The coincidence logic in each reactor trip module controls one or more breakers or contactors in the control rod drive power system.

The four reactor protection system protective channels are identical in their functions, which combine in the system logic to trip the reactor automatically and protect the reactor core for the following conditions:

1) When the reactor power, as measured by neutron flux, exceeds a fixed maximum limit.
2) When the reactor power, as measured by neutron flux, exceeds the limit set by the reactor coolant flow and power imbalance.

3) When the reactor power exceeds the limit set by the number and combination of reactor coolant pumps in operation.
4) When the reactor outlet temperature exceeds a fixed maximum limit.
5) When a specified reactor pressure-outlet temperature relationship is exceeded.
6) When the reactor pressure falls below a fixed minimum limit or exceeds a fixed maximum limit.
7) When Reactor Building pressure exceeds a fixed maximum limit.

In addition to the above protective trips, an anticipatory trip is provided to the reactor protection system to trip the reactor on loss of both main feedwater pumps or a main steam turbine trip.

The reactor protection system is a four channel, redundant system in which the four protection channels are brought together in four identical 2 out of 4 logic networks of the reactor trip modules. A trip in any two of the four protection channels initiates a trip of all four logic networks. Each of the reactor trip modules (2 out of 4 logic networks) controls a control rod drive breaker or contactor. Thus, a trip in any two of the four protection channels initiates a trip of all the breakers and contactors. The power breakers and contactors, however, are arranged in what is effectively a 1 out of 2 x 2 logic. This system combines the advantages of the 2 out of 4 and the 1 out of 2 x 2 systems, while eliminating some of the disadvantages of the 1 out of 2 x 2 alone. The combination results in a system that is considered superior to either of the basic systems alone.

The reactor protection system instrument channels and reactor protection system reactor trip devices are currently tested once per month. This testing includes channel trip as well as coincidence logic functions. The tests are staggered so that one channel is tested each week. These tests trip the associated reactor trip breaker and therefore have the effect of causing a "half-trip" of the control rod drives, which can result in a reactor trip if another reactor protection system channel trips. Some of the spurious trips a reactor experiences can be traced to surveillance testing of the reactor trip system. By increasing the test interval of the reactor trip system instrument strings, the risk of spurious trips can be reduced. As shown in BAW-10167, the test interval extension is not a significant contributor to reactor trip system unavailability or the risk of core damage.

Extension of the reactor protection system instrument channel functional testing from monthly to semi-annually is justified in Topical Report BAW-10167, "Justification for Increasing the Reactor Trip System On-Line Test Interval," dated May 1986, and BAW-10167A, Supplement 1, dated August 1992, as approved by the NRC in Safety Evaluation Report, dated December 5, 1988.

Extension of the reactor protection system control rod drive trip breakers, protective channel coincidence logic, and electronic trip relay functional testing from monthly to quarterly is justified in Topical Report BAW-10167A, Supplement 3 "Justification for Increasing the Reactor Trip System On-Line Test Interval," dated February 1998, as approved by the NRC in Safety Evaluation Report, dated January 7, 1998. TMI Unit 1 was not represented by the B&WOG on this issue as originally described in BAW-10167A, Supplement 3 and the associated NRC safety evaluation report.

However, AmerGen and AREVA have subsequently completed an evaluation (Reference 8) of the TMI Unit 1 system design and operating experience, and confirmed that the approved BAW-10167A, Supplement 3 methodology, modeling, and data are applicable to TMI Unit 1.

4.0 TECHNICAL ANALYSIS

Reactor Protection System Instrumentation Channels

The NRC Safety Evaluation Report, dated December 5, 1988, for Topical Report BAW-10167A, Supplement 1, "Justification for Increasing the Reactor Trip System On-Line Test Interval," requires that each licensee confirm that they have reviewed drift information including as found and as left values for each instrument channel involved and have determined that drift occurring in that channel over the period of the extended surveillance test interval will not cause the setpoint value to exceed the allowable values as calculated for that channel by their setpoint methodology. In addition, each licensee should maintain onsite records showing the actual setpoint calculations and supporting data that are available for planned future NRC staff audits. This data should consist of monthly information taken over an extended period of time (approximately 2-3 years) and the setpoint methodology used to derive the safety margins.

The TMI Unit 1 reactor protection system design and components were represented in the modeling of the reactor trip system for Babcock & Wilcox plants as described in NRC Safety Evaluation Report, dated December 5, 1988, for BAW-10167 and BAW-10167A, Supplement 1, "Justification for Increasing the Reactor Trip System On-Line Test Interval." Drift information including as found and as left values for each reactor protection system instrument channel has been reviewed for TMI Unit 1 (Reference 7). The scope of the drift evaluation includes components from downstream of the reactor protection system test modules to and including the bistables for all the instrument strings of the reactor protection system, consistent with the scope of the test interval relief addressed in BAW-10167, Volume 1. The three-years of data examined justify that drift occurring over the proposed semi-annual surveillance test interval will not cause the setpoint value to exceed the allowable value for that channel. The surveillance data confirm that the reactor protection system instrument strings are capable of operating with a semi-annual surveillance test interval without drifting outside of the limits established for one-month drift. Therefore, the proposed test interval extension can be implemented without requiring revision to the existing setpoint methodology. Records showing the actual setpoint calculations and supporting data for TMI Unit 1 are available onsite.

The as found and as left values will continue to be recorded and reviewed for consistency with the assumptions of the surveillance interval extension analysis. The review will verify the test results meet acceptance criteria. Out of tolerance results will be evaluated to determine if they meet the requirements outlined in BAW-10167.

Reactor Protection System Trip Devices

Although TMI Unit 1 was not represented in the initial reliability analyses described in BAW-10167A, Supplement 3, "Justification for Increasing the Reactor Trip System On-Line Test Intervals," AmerGen and AREVA have subsequently performed this analysis based on the TMI Unit 1 system design and installed components (Reference 8). This analysis for TMI Unit 1 utilized the methodology and reliability models described in BAW-10167A, Supplement 3, as approved by the NRC in Safety Evaluation Report, dated January 7, 1998. The TMI Unit 1 reactor protection system reactor trip devices are within the scope of the Oconee reactor trip system design group as described in BAW-10167A, Supplement 3. Therefore, the reliability models were confirmed to be representative of the reactor protection system reactor trip devices as designed and installed at TMI Unit 1. One minor change implemented at TMI Unit 1, involving the addition of indicator lights for the undervoltage and shunt trip devices of the reactor trip breakers to enhance testability, does not affect the reliability models used in BAW-10167.

The proposed change to the surveillance test interval for the reactor trip devices is based on Supplement 3 to Topical Report BAW-10167A, "Justification for Increasing the Reactor Trip System On-Line Test Intervals." The methodology used in Supplement 3 is the same as used for the BWOG submittal on reactor protection system instrument channel test intervals, BAW-10167A, Supplement 1, which the NRC has reviewed and approved in Safety Evaluation Report dated December 5, 1988.

For reactor trip devices, operating experience indicates an improvement in reliability since the Topical Report BAW-10167 was originally submitted. As stated in NRC SER for BAW-10167A, Supplement 3, the reactor trip breaker failure data reflected reliability improvements and reduction in the potential for common mode failures due to the implementation of Generic Letter 83-28, "Required Actions Based On Generic Implications of Salem ATWS Events." NRC sponsored research, reported in NUREG-1366, "Improvements to Technical Specifications Surveillance Requirements," supports the assertion that improved reactor trip device reliability can be attributed to Generic Letter 83-28 improvements, and also found that a primary stress mechanism for the breakers is routine mechanical cycling associated with testing.

Topical Report BAW-10167A, Supplement 3, dated February 1998, "Justification for Increasing the Reactor Trip System On-Line Test Intervals," provides the analysis and technical justification for increasing the reactor trip device test interval. TMI Unit 1 uses General Electric manufactured model AK breakers as analyzed in BAW-10167. The results of the analysis show that the test interval extension is not a significant contributor to system unavailability or core damage risk. For the Oconee design group as described in the topical report, which includes TMI Unit 1 as outlined above, the incremental change in unavailability is shown to be 2.9×10^{-7} failure/demand and the net incremental risk is shown to be 2.6×10^{-8} frequency of core damage/reactor year.

The recent operating experience for TMI Unit 1 has been reviewed to ensure that the reactor trip breaker failure rate assumptions made in BAW-10167A, Supplement 3 are still valid for TMI Unit 1. Relevant plant-specific operating history for reactor trip breaker failure to open (i.e., failure to trip) from reactor protection system monthly functional tests, the Maintenance Rule functional failure database, and other sources were reviewed. These data covered approximately the most recent ten years of operation. In that time period, only one failure to open was identified. This failure occurred in November of 1997 and involved failure to open of an undervoltage (UV) device during monthly reactor trip breaker functional testing. The redundant shunt trip device (temporarily bypassed to perform the UV device test) was operable and capable of opening the reactor trip breaker. No failures to trip were found of the other subcomponents of the reactor trip breaker (i.e., shunt trip device and breaker frame/armature assembly) during the most recent ten-year period. The demand-related failure rate for the UV device that was used in BAW-10167A, Supplement 3 is 1.21×10^{-3}/demand or one failure to trip per 826 demands. Assuming one demand per month per each of six installed breakers yields an expected frequency of one UV failure per 11 years. This is consistent with the observed single occurrence found in the recent TMI Unit 1 operating history. Therefore, the TMI Unit 1 reactor trip breaker failure data are consistent with the assumptions of BAW-10167. The failure experience from TMI Unit 1 since publication of the report does not contradict any of the assumptions or conclusions of BAW-10167A, Supplement 3 or the NRC's associated Safety Evaluation Report.

Therefore, it is concluded that BAW-10167A, Supplement 3 is applicable to TMI Unit 1. Hence it is technically justifiable to relax the functional test interval, from one month to quarterly, for the reactor protection system trip devices.

Conclusion

Based on the good performance of these components, the results of BAW-10167, the low potential for significant increase in failure rates of the components under a longer test interval, and the introduction of no new failure modes, it is concluded that there is no adverse effect on nuclear safety by increasing the proposed surveillance test intervals. As found and as left TMI Unit 1 surveillance test data will continue to be monitored to ensure the requirements outlined in BAW-10167, May 1986 are maintained.

Consequently, the proposed Technical Specification changes will not adversely affect nuclear safety or safe plant operations.

5.0 REGULATORY ANALYSIS

5.1 No Significant Hazards Consideration

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

1. **Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?**

   Response: No.

The reactor protection system monitors parameters related to safe operation and trips the reactor to protect the reactor core against fuel rod cladding damage. It also assists in protecting against reactor coolant system damage caused by high system pressure by limiting energy input to the system through reactor trip action. Therefore, this change has no impact on the probability of an accident previously evaluated. The results of the reliability analyses conducted in accordance with NRC approved methodology and criteria show that the test interval extension of the reactor protection system instrument channels and reactor trip devices is not a significant contributor to trip system unavailability or the risk of core damage. The reactor protection system instrument channel and reactor trip device functional test surveillance program will continue to ensure that the reactor protection system is capable of performing its intended safety function in the event of a design basis accident. Therefore, this change has no affect on the consequences of an accident previously evaluated.

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

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

Response: No.

The proposed change involves the reactor protection system instrument channel and reactor trip device surveillance test interval, which is not, in and of itself, considered to be an accident initiator. Postulated failure of the reactor protection system instrument channel or reactor trip device to function is an analyzed condition and does not constitute a new or different kind of accident. The proposed change does not create any new failure modes not bounded by previously analyzed accidents.

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

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The results of the reliability analyses conducted in accordance with NRC approved methodology and criteria show that the test interval extension of the reactor protection system instrument channels and reactor trip devices is not a significant contributor to trip system unavailability or the risk of core damage. The Technical Specifications will continue to require the reactor protection system trip setpoints to remain within the assumptions of the accident analysis and that adequate reliability of the reactor protection system trip devices is maintained, thus preserving existing margins of safety.

Therefore, the proposed change does not involve a significant reduction in a margin of safety.

Based on the above, AmerGen Energy Company, LLC (AmerGen) concludes that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92(c), and, accordingly, a finding of "no significant hazards consideration" is justified.

5.2 Applicable Regulatory Requirements/Criteria

The TMI Unit 1 Reactor Protection System is designed to meet the requirements of IEEE-279, "Proposed Criteria for Nuclear Power Plants Protection Systems," dated August 1968, which specifies single-failure, redundancy, independence, and physical separation design requirements for Class 1E protection systems, in conjunction with Atomic Energy Commission (AEC) General Design Criteria, dated July 1967, Criterion 19 – Protection Systems Reliability, Criterion 20 – Protection System Redundancy and Independence, Criterion 21 – Single Failure Definition, Criterion 22 – Separation of Protection and Control Instrumentation Systems, Criterion 23 – Protection Against Multiple Disability for Protection Systems, and Criterion 26 – Protection Systems Fail-Safe Design. The proposed change does not affect the existing design provisions for single failure, redundancy, independence, or physical separation of the Reactor Protection System design.

AEC General Design Criterion 14 requires that core protection systems and associated equipment be designed to act automatically to prevent or to suppress conditions that could result in exceeding acceptable fuel damage limits. The proposed change has no impact on continued compliance with Criterion 14.

In conclusion, based on the considerations discussed above, (1) there is 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 Commission'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.

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. BAW-10167, "Justification for Increasing the Reactor Trip System On-Line Test Interval," dated May 1986.

2. BAW-10167A, Supplement 1, "Justification for Increasing the Reactor Trip System On-Line Test Interval," dated August 1992.

3. BAW-10167A, Supplement 3, "Justification for Increasing the Reactor Trip System On-Line Test Interval," dated February 1998.

4. NRC Safety Evaluation Report for BAW-10167A, Supplement 1, dated December 5, 1988.

5. NRC Safety Evaluation Report for BAW-10167A, Supplement 3, dated January 7, 1998.

6. NUREG-1430, Babcock & Wilcox Plants, Revision 2, Standard Technical Specifications.

7. AREVA Document No. 51-5047232-00, "TMI-1 RPS Test Interval Extension Evaluation of Instrument Channel Drift," dated August 10, 2004.

8. AREVA Document No. 32-1164101-11, "Basis For RTS Test Interval Extension," dated June 30, 2004.

ENCLOSURE 2

TMI Unit 1 Technical Specification Change Request No. 324

Markup of Proposed Technical Specification Page Changes

Revised TS Pages

4-2a
4-2b
4-3
4-4
4-7

Bases (Cont'd)

The 600 ppmb limit in Item 4, Table 4.1-3 is used to meet the requirements of Section 5.4. Under other circumstances the minimum acceptable boron concentration would have been zero ppmb.

Calibration

Calibration shall be performed to assure the presentation and acquisition of accurate information. The nuclear flux (power range) channels amplifiers shall be checked and calibrated if necessary, every shift against a heat balance standard. The frequency of heat balance checks will assure that the difference between the out-of-core instrumentation and the heat balance remains less than 4%.

Channels subject only to "drift" errors induced within the instrumentation itself can tolerate longer intervals between calibrations. Process system instrumentation errors induced by drift can be expected to remain within acceptance tolerances if recalibration is performed at the intervals of each refueling period.

Substantial calibration shifts within a channel (essentially a channel failure) will be revealed during routine checking and testing procedures.

Thus, minimum calibration frequencies set forth are considered acceptable.

Testing

Semi-annually

On-line testing of reactor protection channels is required ~~monthly~~ on a rotational basis. The rotation scheme is designed to reduce the probability of an undetected failure existing within the system and to minimize the likelihood of the same systematic test errors being introduced into each redundant channel (Reference 1).

The rotation schedule for the reactor protection channels is as follows:

Semi-annually

every 46 days

a) Deleted
b) ~~Monthly~~ with one channel being tested ~~per week~~ on a continuous sequential rotation.

every 46 days.

INSERT "A"

The reactor protection system instrumentation test cycle is continued with one channel's instrumentation tested ~~each week~~. Upon detection of a failure that prevents trip action in a channel, the instrumentation associated with the protection parameter failure will be tested in the remaining channels. If actuation of a safety channel occurs, assurance will be required that actuation was within the limiting safety system setting.

The protection channels coincidence logic, the control rod drive trip breakers and the regulating control rod power SCRs electronic trips, are trip tested ~~monthly~~. The trip test checks all logic combinations and is to be performed on a rotational basis.

quarterly with one channel being tested every 23 days on a continuous sequential rotation. Calculations have shown that the frequency of every 23 days maintains a high level of reliability of the Reactor Trip System in Reference 4.

Discovery of a failure that prevents trip action requires the testing of the instrumentation associated with the protection parameter failure in the remaining channels.

For purposes of surveillance, reactor trip on loss of feedwater and reactor trip on turbine trip are considered reactor protection system channels.

Bases (Contd.)

The equipment testing and system sampling frequencies specified in Tables 4.1-2, 4.1-3, **and 4.1-5** are considered adequate to maintain the equipment and systems in a safe operational status.

REFERENCE

(1) UFSAR, Section 7.1.2.3(d) - "Periodic Testing and Reliability"

(2) NRC SER for BAW-10167A, Supplement 1, December 5, 1988.

(3) BAW-10167, May 1986.

(4) BAW-10167A, Supplement 3, February 1998.

INSERT "A" TO PAGE 4-2a:

¶ The frequency of every 46 days on a continuous sequential rotation is consistent with the calculations of Reference 2 that indicate the RPS retains a high level of reliability for this interval.

Amendment No. ~~181~~ ,225

TABLE 4.1-1

INSTRUMENT SURVEILLANCE REQUIREMENTS

	CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
1.	Protection Channel Coincidence Logic	NA	~~M~~ Q	NA	
2.	Control Rod Drive Trip Breaker and Regulating Rod Power SCRs	NA	~~M~~ Q	NA	(1) Includes independent testing of shunt trip and undervoltage trip features.
3.	Power Range Amplifier	D(1)	NA	(2)	(1) When reactor power is greater than 15%. (2) When above 15% reactor power run a heat balance check once per shift. Heat balance calibration shall be performed whenever heat balance exceeds indicated neutron power by more than two percent.
4.	Power Range Channel	S	~~M~~ S/A	M(1)(2)	(1) When reactor power is greater than 60% verify imbalance using incore instrumentation. (2) When above 15% reactor power calculate axial offset upper and lower chambers after each startup if not done within the previous seven days.
5.	Intermediate Range Channel	S(1)	P S/U	NA	(1) When in service.
6.	Source Range Channel	S(1)	P S/A	NA	(1) When in service.
7.	Reactor Coolant Temperature Channel	S	~~M~~ S/A	F	

TABLE 4.1-1 (Continued)

CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
8. High Reactor Coolant Pressure Channel	S	~~M~~ S/A	R	
9. Low Reactor Coolant Pressure Channel	S	~~M~~ S/A	R	
10. Flux-Reactor Coolant Flow Comparator	S	~~M~~ S/A	F	
11. Reactor Coolant Pressure-Temperature Comparator	S	~~M~~ S/A	R	
12. Pump Flux Comparator	S	~~M~~ S/A	R	
13. High Reactor Building Pressure Channel	S	~~M~~ S/A	F	
14. High Pressure Injection Logic Channels	NA	Q	NA	
15. High Pressure Injection Analog Channels				
a. Reactor Coolant Pressure Channel	S(1)	M	R	(1) When reactor coolant system is pressurized above 300 psig or T_{ave} is greater than 200°F
16. Low Pressure Injection Logic Channel	NA	Q	NA	
17. Low Pressure Injection Analog Channels			0	
a. Reactor Coolant Pressure Channel	S(1)	M	R	(1) When reactor coolant system is pressurized above 300 psig or T_{ave} is greater than 200°F
18. Reactor Building Emergency Cooling and Isolation System Logic Channel	NA	Q	NA	

TABLE 4.1-1 (Continued)

	CHANNEL DESCRIPTION	CHECK	TEST	CALIBRATE	REMARKS
38.	OTSG Full Range Level	W	NA	R	
39.	Turbine Overspeed Trip	NA	R	NA	
40.	BWST/NaOH Differential Pressure Indicator	NA	NA	F	
41.	Sodium Hydroxide Tank Level Indicator	NA	NA	F	
42.	Diesel Generator Protective Relaying	NA	NA	R	
43.	4 KV ES Bus Undervoltage Relays (Diesel Start)				
	a. Degraded Grid	NA	M(1)	A	(1) Relay operation will be checked by local test pushbuttons.
	b. Loss of Voltage	NA	M(1)	R	(1) Relay operation will be checked by local test pushbuttons.
44.	Reactor Coolant Pressure DH Valve Interlock Bistable	S(1)	M	R	(1) When reactor coolant system is pressurized above 300 psig or T_{ave} is greater than 200°F.
45.	Loss of Feedwater Reactor Trip	S(1)	S/A ~~M~~(1)	R	(1) When reactor power exceeds 7% power.
46.	Turbine Trip/Reactor Trip	S(1)	S/A ~~M~~(1)	F	(1) When reactor power exceeds 45% power.
47.	a. Pressurizer Code Safety Valve and PORV Tailpipe Flow Monitors	S(1)	NA	F	(1) When T_{ave} is greater than 525°F.
	b. PORV - Acoustic/Flow	NA	M(1)	R	(1) When T_{ave} is greater than 525°F.
48.	PORV Setpoints	NA	M(1)	R	(1) Per Specification 3.1.12 excluding valve operation.

ENCLOSURE 3

Regulatory Commitments

List of Regulatory Commitments

The following table identifies commitments made in this document by AmerGen. Any other actions discussed in the submittal representing intended or planned actions by AmerGen are described to the NRC for the NRC's information and are not regulatory commitments.

COMMITMENTS	COMMITTED DATE
The as-found and as-left values will continue to be recorded and reviewed for consistency with the assumptions of the surveillance interval extension analysis. The review will verify the test results meet acceptance criteria. Out-of-tolerance results will be evaluated to determine if they meet the requirements outlined in BAW-10167.	Within 60 days after issuance of the amendment approving this TSCR.