Mr. Leon R. Eliason
Chief Nuclear Officer & President-
Nuclear Business Unit
Public Service Electric & Gas
Company
Post Office Box 236
Hancocks Bridge, NJ 08038

August 7, 1995

SUBJECT: SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2 (TAC NOS. M85782 AND M85783)

Dear Mr. Eliason:

The Commission has issued the enclosed Amendment Nos. 173 and 154 to Facility Operating License Nos. DPR-70 and DPR-75 for the Salem Nuclear Generating Station, Unit Nos. 1 and 2. These amendments consist of changes to the Technical Specifications (TSs) in response to your application dated February 5, 1993, supplemented April 13, June 11 and November 17, 1993.

These amendments eliminate the Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level Reactor Trip due to the installation of the digital feedwater control system incorporating a median signal selector.

A copy of our safety evaluation is also enclosed. Notice of Issuance will be included in the Commission's biweekly Federal Register notice.

Sincerely,

/S/

Leonard N. Olshan, Senior Project Manager
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Docket Nos. 50-272/311

Enclosures: 1. Amendment No. 173 to License No. DPR-70
2. Amendment No. 154 to License No. DPR-75
3. Safety Evaluation

cc w/encls: See next page

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

August 7, 1995

Mr. Leon R. Eliason
Chief Nuclear Officer & President-
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Docket Nos. 50-272/311

Enclosures:
1. Amendment No. 173 to License No. DPR-70
2. Amendment No. 154 to License No. DPR-75
3. Safety Evaluation

cc w/encls: See next page

Mr. Leon R. Eliason
Public Service Electric & Gas
Company

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Units 1 and 2

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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

PUBLIC SERVICE ELECTRIC & GAS COMPANY

PHILADELPHIA ELECTRIC COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-272

SALEM NUCLEAR GENERATING STATION, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 173
License No. DPR-70

1. The Nuclear Regulatory Commission (the Commission or the NRC) has found that:

   A. The application for amendment filed by the Public Service Electric & Gas Company, Philadelphia Electric Company, Delmarva Power and Light Company and Atlantic City Electric Company (the licensees) dated February 5, 1993, supplemented April 13, June 11 and November 17, 1993, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-70 is hereby amended to read as follows:

(2) <u>Technical Specifications and Environmental Protection Plan</u>

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 173 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance, to be implemented by the startup following the twelfth refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: August 7, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 173

FACILITY OPERATING LICENSE NO. DPR-70

DOCKET NO. 50-272

Revise Appendix A as follows:

Remove Pages	Insert Pages
2-6	2-6
B 2-6	B 2-6
B 2-7	B 2-7
3/4 3-3	3/4 3-3
3/4 3-10	3/4 3-10
3/4 3-12	3/4 3-12

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
13. Steam Generator Water Level--Low-Low	≥ 9.0% of narrow range instrument span-each steam generator	≥ 8.0% of narrow range instrument span-each steam generator
14. Deleted		
15. Undervoltage-Reactor Coolant Pumps	≥ 2900 volts-each bus	≥ 2850 volts-each bus
16. Underfrequency-Reactor Coolant Pumps	≥ 56.5 Hz - each bus	≥ 56.4 Hz - each bus
17. Turbine Trip		
A. Low Trip System Pressure	≥ 45 psig	≥ 45 psig
B. Turbine Stop Valve Closure	≤ 15% off full open	≤ 15% off full open
18. Safety Injection Input from SSPS	Not Applicable	Not Applicable
19. Reactor Coolant Pump Breaker Position Trip	Not Applicable	Not Applicable

BASES

through the pressurizer safety valves. No credit was taken for operation of this trip in the accident analyses; however, its functional capability at the specified trip setting is required by this specification to enhance the overall reliability of the Reactor Protection System.

Loss of Flow

The Loss of Flow trips provide core protection to prevent DNB in the event of a loss of one or more reactor coolant pumps.

Above 11 percent of RATED THERMAL POWER, an automatic reactor trip will occur if the flow in any two loops drop below 90% of nominal full loop flow. Above 36% (P-8) of RATED THERMAL POWER, automatic reactor trip will occur if the flow in any single loop drops below 90% of nominal full loop flow. This latter trip will prevent the minimum value of the DNBR from going below the design DNBR value during normal operational transients and anticipated transients when 3 loops are in operation and the Overtemperature ΔT trip set point is adjusted to the value specified for all loops in operation. With the Overtemperature ΔT trip set point adjusted to the value specified for 3 loop operation, the P-8 trip at 76% RATED THERMAL POWER will prevent the minimum value of the DNBR from going below the design DNBR value during normal operational transients and anticipated transients with 3 loops in operation.

Steam Generator Water Level

The Steam Generator Water Level Low-Low trip provides core protection by preventing operation with the steam generator water level below the minimum volume required for adequate heat removal capacity. The specified setpoint provides allowance that there will be sufficient water inventory in the steam generators at the time of trip to allow for starting delays of the auxiliary feedwater system.

Undervoltage and Underfrequency - Reactor Coolant Pump Busses

The Undervoltage and Underfrequency Reactor Coolant Pump bus trips provide reactor core protection against DNB as a result of loss of voltage or underfrequency to more than one reactor coolant pump. The specified set points assure a reactor trip signal is generated before the low flow trip set point is reached. Time delays are incorporated in the underfrequency and undervoltage trips to prevent spurious reactor trips from momentary electrical power transients. For undervoltage, the delay is set so that the time required for a signal to reach the reactor trip breakers following the simultaneous trip of two or more reactor coolant pump bus circuit breakers shall not exceed 0.9 seconds. For underfrequency, the delay is set so that the time required for a signal to reach the reactor trip breakers after the underfrequency trip setpoint is reached shall not exceed 0.3 seconds.

Turbine Trip

A Turbine Trip causes a direct reactor trip when operating above P-9. Each of the turbine trips provide turbine protection and reduce the severity of the ensuing transient. No credit was taken in the accident analyses for operation of these trips. Their functional capability at the specified trip settings is required to enhance the overall reliability of the Reactor Protection System.

TABLE 3.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	TOTAL NUMBER OF CHANNELS	CHANNELS TO TRIP	MINIMUM CHANNELS OPERABLE	APPLICABLE MODES	ACTION
11. Pressurizer Water Level--High	3	2	2	1, 2	6#
12. Loss of Flow - Single Loop (Above P-8)	3/loop	2/loop in any operating loop	2/loop in each operating loop	1	6#
13. Loss of Flow - Two Loops (Above P-7 and below P-8)	3/loop	2/loop in two operating loops	2/loop in each operating loop	1	6#
14. Steam Generator Water Level--Low-Low	3/loop	2/loop in any operating loops	2/loop in each operating loop	1, 2	6#
15. Deleted					
16. Undervoltage-Reactor Coolant Pumps	4-1/bus	1/2 twice	3	1	6#
17. Underfrequency-Reactor Coolant Pumps	4-1/bus	1/2 twice	3	1	6#

TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME
12. Loss of Flow - Single Loop (Above P-8)	≤ 1.0 seconds
13. Loss of Flow - Two Loops (Above P-7 and below P-8)	≤ 1.0 seconds
14. Steam Generator Water Level-- Low-Low	≤ 2.0 seconds
15. Deleted	
16. Undervoltage-Reactor Coolant Pumps	≤ 1.2 seconds
17. Underfrequency-Reactor Coolant Pumps	≤ 0.6 seconds
18. Turbine Trip	
A. Low Fluid Oil Pressure	NOT APPLICABLE
B. Turbine Stop Valve	NOT APPLICABLE
19. Safety Injection Input from ESF	NOT APPLICABLE
20. Reactor Coolant Pump Breaker Position Trip	NOT APPLICABLE
21. Reactor Trip Breakers	NOT APPLICABLE
22. Automatic Trip Logic	NOT APPLICABLE

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CALIBRATION	CHANNEL FUNCTIONAL TEST	MODES IN WHICH SURVEILLANCE REQUIRED
13. Loss of Flow Two Loops	S	R	N.A.	1
14. Steam Generator Water Level--Low-Low	S	R	Q	1, 2
15. Deleted				
16. Undervoltage - Reactor Coolant Pumps	N.A.	R	Q	1
17. Underfrequency - Reactor Coolant Pumps	N.A.	R	Q	1
18. Turbine Trip				
a. Low Autostop Oil Pressure	N.A.	N.A.	S/U(1)	1, 2
b. Turbine Stop Valve Closure	N.A.	N.A.	S/U(1)	1, 2
19. Safety Injection Input from ESF	N.A.	N.A.	M(4)(5)	1, 2
20. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	R	N.A.
21. Reactor Trip Breaker	N.A.	N.A.	S/U(10), M(11,13), SA(12,13) and R(14)	1, 2 and *
22. Automatic Trip Logic	N.A.	N.A.	M(5)	1, 2 and *

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

PUBLIC SERVICE ELECTRIC & GAS COMPANY

PHILADELPHIA ELECTRIC COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

DOCKET NO. 50-311

SALEM NUCLEAR GENERATING STATION, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 154
License No. DPR-75

1. The Nuclear Regulatory Commission (the Commission or the NRC) has found that:

   A. The application for amendment filed by the Public Service Electric & Gas Company, Philadelphia Electric Company, Delmarva Power and Light Company and Atlantic City Electric Company (the licensees) dated February 5, 1993, supplemented April 13, June 11 and November 17, 1993, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance: (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations set forth in 10 CFR Chapter I;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-75 is hereby amended to read as follows:

(2) Technical Specifications and Environmental Protection Plan

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 154 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of its date of issuance, to be implemented by the startup following the current outage.

FOR THE NUCLEAR REGULATORY COMMISSION

John F. Stolz

John F. Stolz, Director
Project Directorate I-2
Division of Reactor Projects - I/II
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of Issuance: August 7, 1995

ATTACHMENT TO LICENSE AMENDMENT NO. 154

FACILITY OPERATING LICENSE NO. DPR-75

DOCKET NO. 50-311

Revise Appendix A as follows:

Remove Pages	Insert Pages
2-6	2-6
B 2-6	B 2-6
3/4 3-3	3/4 3-3
3/4 3-10	3/4 3-10
3/4 3-12	3/4 3-12

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A. Low Trip System Pressure	≥ 45 psig	≥ 45 psig
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18. Safety Injection Input from SSPS	Not Applicable	Not Applicable
19. Reactor Coolant Pump Breaker Position Trip	Not Applicable	Not Applicable

LIMITING SAFETY SYSTEM SETTINGS

BASES

Loss of Flow

The Loss of Flow trips provide core protection to prevent DNB in the event of a loss of one or more reactor coolant pumps.

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TABLE 3.3-1 (Continued)

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14. Steam Generator Water Level--Low-Low	3/loop	2/loop in any operating loops	2/loop in each operating loop	1, 2	6#
15. Deleted					
16. Undervoltage-Reactor Coolant Pumps	4-1/bus	1/2 twice	3	1	6
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TABLE 3.3-2 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION RESPONSE TIMES

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A. Low Fluid Oil Pressure	NOT APPLICABLE
B. Turbine Stop Valve	NOT APPLICABLE
19. Safety Injection Input from ESF	NOT APPLICABLE
20. Reactor Coolant Pump Breaker Position Trip	NOT APPLICABLE
21. Reactor Trip Breakers	NOT APPLICABLE
22. Automatic Trip Logic	NOT APPLICABLE

TABLE 4.3-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

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14. Steam Generator Water Level--Low-Low	S	R	Q	1, 2
15. Deleted				
16. Undervoltage - Reactor Coolant Pumps	N.A.	R	Q	1
17. Underfrequency - Reactor Coolant Pumps	N.A.	R	Q	1
18. Turbine Trip				
a. Low Autostop Oil Pressure	N.A.	N.A.	S/U(1)	N.A.
b. Turbine Stop Valve Closure	N.A.	N.A.	S/U(1)	N.A.
19. Safety Injection Input from ESF	N.A.	N.A.	M(4)(5)	1, 2
20. Reactor Coolant Pump Breaker Position Trip	N.A.	N.A.	R	N.A.
21. Reactor Trip Breaker	N.A.	N.A.	S/U(10), M(11,13), SA(12,13) and R(14)	1, 2 and *
22. Automatic Trip Logic	N.A.	N.A.	M(5)	1, 2 and *

UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NOS. 173 AND 154 TO FACILITY OPERATING

PUBLIC SERVICE ELECTRIC & GAS COMPANY

PHILADELPHIA ELECTRIC COMPANY

DELMARVA POWER AND LIGHT COMPANY

ATLANTIC CITY ELECTRIC COMPANY

SALEM NUCLEAR GENERATING STATION, UNIT NOS. 1 AND 2

DOCKET NOS. 50-272 AND 50-311

1.0 INTRODUCTION

By letter dated February 5, 1993, as supplemented April 13, June 11 and November 17, 1993, the Public Service Electric & Gas Company (the licensee) submitted a request for changes to the Salem Nuclear Generating Station, Unit Nos. 1 and 2 (SGS), Technical Specifications (TS). The requested changes would eliminate the Steam/Feedwater Mismatch and Low Steam Generator Water Level Reactor Trip due to the installation of a new advanced digital feedwater control system (ADFWCS) incorporating the median signal selector (MSS). The April 13, June 11 and November 17, 1993, letters provided clarifying information that did not change the initial proposed no significant hazards consideration determination.

The TS changes include deletion of the following:

(1) TS Table 2.2-1, Item 14, regarding the reactor trip system instrumentation trip setpoint for the Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level.

(2) The associated TS bases section for the Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level.

(3) TS Table 3.3-1, Item 15, regarding the reactor trip system instrumentation for the Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level.

(4) TS Table 3.3-2, Item 15, regarding the reactor trip system instrumentation response time for the Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level.

(5) TS Table 4.3-1, Item 15, regarding the reactor trip system instrumentation surveillance requirements for the Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level.

2.0 EVALUATION

Each of the four steam generators (SG) at Salem Units 1 and 2, has three independent narrow-range water level detection instrument channels that provide input to the reactor trip system (RTS) for a reactor trip on two out of three low-low water level signals. This two/three coincident logic also provides a starting signal for the auxiliary feedwater pumps. The low-low steam generator water level reactor trip function is designed to preserve the steam generator as a heat sink for removal of residual heat in the event of a loss of normal feedwater.

In an event of loss of feedwater, the feedwater level in the steam generator falls below the low-low level trip setpoint in the reactor trip circuitry, which in turn trips the reactor. The Salem Units 1 and 2 Updated Final Safety Analysis Report (UFSAR) accident analysis takes credit only for the low-low steam generator water level reactor trip to ensure safe shutdown of the reactor; no credit is taken for the steam/feedwater flow mismatch coincident with low steam generator water level reactor trip function.

In the existing design one of the steam generator water level narrow-range instrument channels also supplies an input to the Feedwater Control System (FWCS). (The FWCS is also referred to as the steam generator water level control system in the SGS UFSAR.) As a result, common instrument channels are used for both RTS and FWCS separated electrically by qualified isolation devices. The steam/feedwater flow mismatch coincident with low steam generator water level reactor trip was installed to satisfy the requirements of the Institute of Electric and Electronics Engineers Standard 279, 1971 (IEEE Standard 279), "Criterion For Protection Systems For Nuclear Power Generating Stations," which is endorsed by the Code of Federal Regulations Title 10 Part 50.55a. IEEE Standard 279, Section 4.7.3, Single Random Failure, states in part, "Where a single random failure can cause a control system action that results in generating a station condition requiring protective action and also prevent proper action of a protective system channel designed to protect against the condition, the remaining redundant protection channels shall be capable of providing the protective action even when degraded by a second random failure." The intent of the steam/feedwater flow mismatch coincident with low steam generator water level reactor trip is to satisfy this criterion.

The staff reviewed WCAP 13502, "Advanced Digital Feedwater Control System for Public Service Electric & Gas Company, Salem Units 1 and 2." This WCAP provides a detailed description of the enhanced feedwater control system to be installed at SGS. The new ADFWCS includes an MSS that is designed to select the median of the three SG narrow-range water level instrument input signals, and reject any signal that is faulty. By selecting the median signal, a single random failure will not cause a control system action that results in a condition requiring protective action. The concern of the adverse interaction between the FWCS and the RTS is eliminated and therefore the requirements of IEEE Standard 279 are satisfied without credit being taken for the steam/feedwater flow mismatch coincident with low steam generator water level reactor trip.

The ADFWCS and MSS verification and validation (V&V) process has been performed by the staff in conjunction with the modification of the FWCS at Prairie Island Nuclear Generating Plant. At that time, the staff audited the software design and its V&V process for the ADFWCS and MSS at the vendor site and concluded that both the MSS and ADFWS are acceptable. This acceptance is based on guidelines provided in ANSI/IEEE-ANS-7.4.3.2 and Regulatory Guide 1.152, "Criteria for Programmable Digital Computer System Software in Safety-Related Systems of Nuclear Power Plants." The acceptance of the MSS and ADFWS is documented in Amendment Nos. 85 and 92, dated March 13, 1990, to Northern States Power, licensee of Prairie Island.

Since the ADFWCS and MSS that will be installed at Salem is identical to that at Prairie Island, a vendor audit was omitted. However, the staff reviewed the software process with emphasis on the configuration management portion of the licensee's software design process. The licensee stated during a conference call that reconfiguration of the MSS is not required at this time. In addition, the licensee stated, that based on the operating history of the MSS at other nuclear power plants future reconfiguration of the algorithm is not likely. However, any modifications to the ADFWCS and MSS will be issued via a Design Control Process in conformance with Procedure NC.NA-AP.ZZ-0064(Q), Revision 0, "Software Quality Assurance," and Procedure ND.DE-AP.AA-0054(Q), Revision 1, "Process Computer Maintenance and Modification Control." The licensee will consider the MSS as critical software, therefore, functional changes made to it will be required to be tested in a manner similar to that used to verify the original design. Furthermore, the licensee stated that it will consider all changes to the MSS as functional changes, and it will involve Westinghouse in any changes to the MSS. However, the staff requests that any configuration change or modification to the MSS be submitted for staff review and approval prior to implementation if it is not consistent with the original software design process.

Prior to the installation of the ADFWCS, the licensee will be performing dynamic testing of the system's interaction with the plant simulator. Additionally, during the installation of the ADFWCS, dynamic testing will be performed to verify adequate interaction with the installed plant equipment.

The licensee has committed to conduct normal operational tests of the MSS once per refueling cycle. This would be identical to the testing frequency for other control instrumentation. In addition, the licensee has stated that for the first year of operation, the MSS will be tested quarterly along with the functional testing of the steam generator narrow-range level channels in order to provide greater assurance of proper system operation. Satisfactory test results are based on observing that an intentionally failed channel is not selected by the MSS for a control function. The MSS function is checked for both a high and low failure of the input signal. The staff agrees to these additional voluntary testing actions associated with the MSS.

In addition, the licensee indicated that a log of the troubles encountered during the first operating cycle testing period will be maintained. This log will also be used to document the changes made to the MSS during the initial cycles of operation. This more formal means of documentation and tracking will provide an aid for evaluating and maintaining the reliability of the MSS. This log can be incorporated with the tracking mechanisms associated with the licensee's Design Control Process that are currently in use.

The isolation devices separating the low-low steam generator water level reactor protection channel and the MSS of the ADFWCS are the Westinghouse Model 131-110 isolation amplifiers. This model isolation amplifier is a qualified isolation device. In addition, the licensee stated that the isolation device will in no way degrade the MSS such that damage would occur to RTS circuit due to a fault in FWCS and/or to the FWCS due to a fault in the RTS circuit.

The ADFWCS has been successfully tested with respect to electromagnetic interference (EMI) using a radiated energy of field strength 3 Volts/meter (V/m), with a frequency band of 20 Mhz to 1 GHz as documented in WCAP 11313, "Results of Electromagnetic Interference Tests Applied to Westinghouse Distributed Processing Family." This test indicated that there was no interference effects on the continuous operation of the ADFWCS.

No limiting conditions of operation are required if the computer performing the median signal selector algorithm should become inoperable, because failure of the computer would not preclude protective action on low-low steam generator water level. The failure would be announced to the operator and feedwater control would automatically be switched to the backup computer before the system would have to be transferred to manual. In case both primary and backup computers fail, the FW valve-control-demand signal will be reduced to zero. Upon loss of the control signal, the valve will fail in the as-is position. The licensee informed the staff that the FW valve position indication in the control room will not be affected by this scenerio, and the operator would be able to open/close the valve manually. However, the position indication of the bypass valve would go to zero. The licensee further added that the bypass valve is used during start-up, and operators have been trained for such an occurrence; to take required action(s) such that transients are not generated and safety systems are not challanged. Also, in the event that two or more channels of steam generator narrow range level have failed, the steam generator water level control of the associated steam generator would automatically switch to manual control. This switch to manual control would also be annunciated to the operator. This is similar to the present, less redundant system where one failure could require control to be transferred to manual.

Based on the staff's review, the ADFWCS which utilized the MSS is found to be acceptable. This acceptance means that the reactor trip initiated by the steam/feedwater flow mismatch and low steam generator water level is not required and can be removed.

No credit is taken for the steam/feedwater flow mismatch and low steam generator water level reactor trip in mitigating the consequences of any of the design bases accidents or transients. The original purpose of installing this trip was to satisfy the single random failure requirement specified in IEEE Standard 279 (1971), Section 4.7.3 for control and protective interaction requirements. The MSS provides an acceptable method of resolving the interaction between feedwater control and the low-low steam generator water level protective function which satisfies the interaction requirement of IEEE Standard 279. On this basis, the staff finds the proposed changes involving the elimination of the steam/feedwater flow mismatch and low steam generator water level reactor trip to be acceptable. The staff concludes that the MSS meets all of the applicable guidelines and regulations, and that its utilization as discussed in this safety evaluation is acceptable.

3.0 STATE CONSULTATION

In accordance with the Commission's regulations, the New Jersey State official was notified of the proposed issuance of the amendments. By letter dated June 29, 1993, the State expressed concern that the advanced digital feedwater control system may be susceptible to electronic failures similar to those that had been experienced in the rod control system. In a meeting summary dated July 28, 1993, of a meeting held on July 19, 1993, the following response was made:

> "The cause of the rod control system failures was a back electromagnetic force (EMF) being generated by step counters. These are inductive loads in the system. There were also human errors involved in the troubleshooting efforts. During the maintenance that was performed on the rod control system, there were no precautions taken to prevent electrostatic discharge (ESD) from degrading sensitive components. The digital feedwater control system will have ESD controls in place throughout the installation and during any maintenance activities. The digital feedwater control system does not have any inductive loads associated with it, and the output contacts all have optical isolators. In addition, all maintenance performed on the system will be controlled."

4.0 ENVIRONMENTAL CONSIDERATION

The amendments change a requirement with respect to installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The NRC staff has determined that the amendments involve no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendments involve no significant hazards consideration, and there has been no public comment on such finding (58 FR 25864). Accordingly, the amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of the amendments.

5.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that: (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 amendments will not be inimical to the common defense and security or to the health and safety of the public.

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