January 29, 1996

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Mr. Donald C. Shelton
Acting Vice President Nuclear - Perry
Centerior Service Company
P. O. Box 97, A200
Perry, OH 44081

SUBJECT: AMENDMENT NO. 79 TO FACILITY OPERATING LICENSE NO. NPF-58 - PERRY NUCLEAR POWER PLANT, UNIT NO. 1 (TAC NO. M88283)

Dear Mr. Shelton:

The Commission has issued the enclosed Amendment No. 79 to Facility Operating License No. NPF-58 for the Perry Nuclear Power Plant, Unit No. 1. This amendment revises the Technical Specifications in response to your application dated November 22, 1993, supplemented May 5 and December 20, 1995.

This amendment revises the Technical Specifications to reflect the replacement of analog temperature instrumentation associated with leak detection with digital equipment.

A copy of the Safety Evaluation is also enclosed. Notice of issuance will be included in the Commission's next biweekly Federal Register notice.

Sincerely,

ORIGINAL SIGNED BY:

Jon B. Hopkins, Sr. Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-440

Enclosures: 1. Amendment No. 79 to License No. NPF-58
2. Safety Evaluation

cc w/encls: See next page

DOCUMENT NAME:G:\PERRY\PER88283.AMD *See Previous Concurrence

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

January 29, 1996

Mr. Donald C. Shelton
Acting Vice President Nuclear - Perry
Centerior Service Company
P. O. Box 97, A200
Perry, OH 44081

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Jon B. Hopkins

Jon B. Hopkins, Sr. Project Manager
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Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-440

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2. Safety Evaluation

cc w/encls: See next page

Mr. Donald C. Shelton
Centerior Service Company

Perry Nuclear Power Plant
Unit Nos. 1 and 2

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Mayor, Village of Perry
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Perry, Ohio 44081

The Honorable Robert V. Orosz
Mayor, Village of North Perry
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Mr. Richard D. Brandt, Plant Manager
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(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 79 are hereby incorporated into this license. The Cleveland Electric Illuminating Company shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

3. This license amendment is effective as of its date of issuance and shall be implemented not later than 120 days following startup from the fifth refueling outage.

FOR THE NUCLEAR REGULATORY COMMISSION

Jon B. Hopkins

Jon B. Hopkins, Senior Project Manager
Project Directorate III-3
Division of Reactor Projects III/IV
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of issuance: January 29, 1996

ATTACHMENT TO LICENSE AMENDMENT NO. 79

FACILITY OPERATING LICENSE NO. NPF-58

DOCKET NO. 50-440

Replace the following pages of the Appendix "A" Technical Specifications including the issued but not yet implemented Improved Technical Specifications (ITS) with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change.

Remove	Insert
1-1	1-1
3/4 3-23	3/4 3-23
3/4 3-24	3/4 3-24
3/4 3-25	3/4 3-25
3/4 3-26	3/4 3-26
ITS 3.3-53	ITS 3.3-53
--	ITS 3.3-53a
ITS 3.3-54	ITS 3.3-54
ITS 3.3-56	ITS 3.3-56
ITS 3.3-57	ITS 3.3-57
ITS 3.3-58	ITS 3.3-58
ITS 3.3-59	ITS 3.3-59
ITS B3.3-170	ITS B3.3-170
ITS B3.3-172	ITS B3.3-172
--	ITS B3.3-172a

1.0 DEFINITIONS

The following terms are defined so that uniform interpretation of these specifications may be achieved. The defined terms appear in capitalized type and shall be applicable throughout these Technical Specifications.

ACTION

1.1 ACTION shall be that part of a Specification which prescribes remedial measures required under designated conditions.

AVERAGE PLANAR EXPOSURE

1.2 The AVERAGE PLANAR EXPOSURE shall be applicable to a specific planar height and is equal to the sum of the exposure of all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

AVERAGE PLANAR LINEAR HEAT GENERATION RATE

1.3 The AVERAGE PLANAR LINEAR HEAT GENERATION RATE (APLHGR) shall be applicable to a specific planar height and is equal to the sum of the LINEAR HEAT GENERATION RATES for all the fuel rods in the specified bundle at the specified height divided by the number of fuel rods in the fuel bundle.

CHANNEL CALIBRATION

1.4 A CHANNEL CALIBRATION shall be the adjustment, as necessary, of the channel output such that it responds with the necessary range and accuracy to known values of the parameter which the channel monitors. The CHANNEL CALIBRATION shall encompass the entire channel including the sensor and alarm and/or trip functions, and shall include the CHANNEL FUNCTIONAL TEST. The CHANNEL CALIBRATION may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is calibrated.

CHANNEL CHECK

1.5 A CHANNEL CHECK shall be the qualitative assessment of channel behavior during operation by observation. This determination shall include, where possible, comparison of the channel indication and/or status with other indications and/or status derived from independent instrument channels measuring the same parameter.

CHANNEL FUNCTIONAL TEST

1.6 A CHANNEL FUNCTIONAL TEST shall be:

a. Analog/digital channels - the injection of a simulated signal into the channel as close to the sensor as practicable to verify OPERABILITY including alarm and/or trip functions and channel failure trips.

b. Bistable channels - the injection of a simulated signal into the sensor to verify OPERABILITY including alarm and/or trip functions.

The CHANNEL FUNCTIONAL TEST may be performed by any series of sequential, overlapping or total channel steps such that the entire channel is tested.

TABLE 4.3.2.1-1

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. PRIMARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low, Level 2	S	Q	$R^{(b)}$	1, 2, 3 and #
b. Drywell Pressure - High ##	S	Q	$R^{(b)}$	1, 2, 3
c. Containment and Drywell Purge Exhaust Plenum Radiation - High	S	Q	R	1, 2, 3 and *
d. Reactor Vessel Water Level - Low, Level 1	S	Q	$R^{(b)}$	1, 2, 3 and #
e. Manual Initiation	NA	R	NA	1, 2, 3 and *
2. MAIN STEAM LINE ISOLATION				
a. Reactor Vessel Water Level - Low, Level 1	S	Q	$R^{(b)}$	1, 2, 3
b. Main Steam Line Radiation - High	S	Q	R	***
c. Main Steam Line Pressure - Low	S	Q	$R^{(b)}$	1
d. Main Steam Line Flow - High	S	Q	$R^{(b)}$	1, 2, 3
e. Condenser Vacuum - Low	S	Q	$R^{(b)}$	1, 2**, 3**
f. Main Steam Line Tunnel Temperature - High				
1. Division 1 and 2	S	SA	R	1, 2, 3
2. Division 3 and 4	S	Q	R	1, 2, 3
g. Main Steam Line Tunnel Δ Temperature - High				
1. Division 1 and 2	S	SA	R	1, 2, 3
2. Division 3 and 4	S	Q	R	1, 2, 3
h. Turbine Building Main Steam Line Temperature - High	S	Q	R	1, 2, 3
i. Manual Initiation	NA	R	NA	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
3. SECONDARY CONTAINMENT ISOLATION				
a. Reactor Vessel Water Level - Low, Level 2	S	Q	R(b)	1, 2, 3 and #
b. Drywell Pressure - High ##	S	Q	R(b)	1, 2, 3
c. Manual Initiation	NA	R	NA	1, 2, 3 and *
4. REACTOR WATER CLEANUP SYSTEM ISOLATION				
a. Δ Flow - High	S	Q	R	1, 2, 3
b. Δ Flow Timer	NA	Q	R	1, 2, 3
c. Equipment Area Temperature - High	S	SA	R	1, 2, 3
d. Equipment Area Ventilation Δ Temperature - High	S	SA	R	1, 2, 3
e. Reactor Vessel Water Level - Low, Level 2	S	Q	R(b)	1, 2, 3
f. Main Steam Line Tunnel Ambient Temperature - High	S	SA	R	1, 2, 3
g. Main Steam Line Tunnel Δ Temperature - High	S	SA	R	1, 2, 3
h. SLCS Initiation	NA	Q(a)	NA	1, 2, 3
i. Manual Initiation	NA	R	NA	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION				
a. RCIC Steam Line Flow - High	S	Q	$R^{(b)}$	1, 2, 3
b. RCIC Steam Supply Pressure - Low	S	Q	$R^{(b)}$	1, 2, 3
c. RCIC Turbine Exhaust Diaphragm Pressure - High	S	Q	$R^{(b)}$	1, 2, 3
d. RCIC Equipment Room Ambient Temperature - High	S	SA	R	1, 2, 3
e. Deleted				
f. Main Steam Line Tunnel Ambient Temperature - High	S	SA	R	1, 2, 3
g. Main Steam Line Tunnel Δ Temperature - High	S	SA	R	1, 2, 3
h. Main Steam Line Tunnel Temperature Timer	NA	SA	R	1, 2, 3
i. RHR Equipment Room Ambient Temperature - High	S	SA	R	1, 2, 3
j. RHR Equipment Room Δ Temperature - High	S	SA	R	1, 2, 3
k. RCIC Steam Line Flow High Timer	NA	Q	R	1, 2, 3
l. Drywell Pressure - High	S	Q	$R^{(b)}$	1, 2, 3
m. Manual Initiation	NA	R	NA	1, 2, 3

TABLE 4.3.2.1-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION SURVEILLANCE REQUIREMENTS

TRIP FUNCTION	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
6. RHR SYSTEM ISOLATION				
a. RHR Equipment Area Ambient Temperature - High	S	SA	R	1, 2, 3
b. RHR Equipment Area Δ Temperature - High	S	SA	R	1, 2, 3
c. RHR/RCIC Steam Line Flow - High	S	Q	R(b)	1, 2, 3
d. Reactor Vessel Water Level - Low, Level 3 ##	S	Q	R(b)	1, 2, 3
e. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	S	Q	R(b)	1, 2, 3
f. Drywell Pressure - High ##	S	Q	R(b)	1, 2, 3
g. Manual Initiation	NA	R	NA	1, 2, 3

* When handling irradiated fuel in the primary containment and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

** When any turbine stop valve is greater than 90% open and/or the key locked bypass switch is in the normal position.

*** OPERATIONAL CONDITION 1 or 2 when the mechanical vacuum pump lines are not isolated.

\# During CORE ALTERATION and operations with a potential for draining the reactor vessel.

(a) Each train or logic channel shall be tested at least every other 92 days.

(b) Calibrate trip unit setpoint at least once per 92 days.

\## These Trip Functions (1b, 3b, 6d, and 6f) utilize instruments which are common to RPS instrumentation.

SURVEILLANCE REQUIREMENTS

------------------------------------NOTES------------------------------------
1. Refer to Table 3.3.6.1-1 to determine which SRs apply for each Function.

2. When a channel is placed in an inoperable status solely for performance of required Surveillances, entry into associated Conditions and Required Actions may be delayed for up to 6 hours, provided the associated Function maintains isolation capability.

SURVEILLANCE		FREQUENCY
SR 3.3.6.1.1	Perform CHANNEL CHECK.	12 hours
SR 3.3.6.1.2	----------------NOTE-------------------- For Function 1.e in Table 3.3.6.1-1, this SR is applicable only to the Division 3 and 4 instruments. --- Perform CHANNEL FUNCTIONAL TEST.	92 days
SR 3.3.6.1.3	Calibrate the trip unit.	92 days
SR 3.3.6.1.4	Perform CHANNEL CALIBRATION.	18 months
SR 3.3.6.1.5	Perform LOGIC SYSTEM FUNCTIONAL TEST.	18 months
SR 3.3.6.1.6	Verify the ISOLATION SYSTEM RESPONSE TIME for the main steam isolation valves is within limits.	18 months on a STAGGERED TEST BASIS

(continued)

SURVEILLANCE		FREQUENCY
SR 3.3.6.1.7	------------------NOTE------------------ For Function 1.e in Table 3.3.6.1-1, this SR is applicable only to the Division 1 and 2 instruments. -- Perform CHANNEL FUNCTIONAL TEST.	184 days

Table 3.3.6.1-1 (page 1 of 6)
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
1. Main Steam Line Isolation					
a. Reactor Vessel Water Level – Low Low Low, Level 1	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 14.3 inches
b. Main Steam Line Pressure – Low	1	2	E	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6	≥ 795.0 psig
c. Main Steam Line Flow – High	1,2,3	2 per MSL	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.6	≤ 191 psid
d. Condenser Vacuum – Low	1,2[(a)], 3[(a)]	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ 7.6 inches Hg vacuum
e. Main Steam Line Pipe Tunnel Temperature – High	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.7	≤ 158.9°F
f. Main Steam Line Turbine Building Temperature—High	1,2,3	2	D	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 138.9°F
g. Manual Initiation	1,2,3	2	G	SR 3.3.6.1.5	NA
2. Primary Containment and Drywell Isolation					
a. Reactor Vessel Water Level – Low Low, Level 2	1,2,3	2[(b)]	H	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ 127.6 inches

(continued)

(a) With any turbine stop valve not closed.

(b) Required to initiate the associated drywell isolation function.

Table 3.3.6.1-1 (page 3 of 6)
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
2. Primary Containment and Drywell Isolation					
g. Containment and Drywell Purge Exhaust Plenum Radiation – High (continued)	(d)	2	K	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 4.0 mR/hr above background
h. Manual Initiation	1,2,3	2(b)	G	SR 3.3.6.1.5	NA
	(d)	2	K	SR 3.3.6.1.5	NA
3. Reactor Core Isolation Cooling (RCIC) System Isolation					
a. RCIC Steam Line Flow – High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 298.5 inches water
b. RCIC Steam Line Flow Time Delay	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ 3 seconds and ≤ 13 seconds
c. RCIC Steam Supply Line Pressure – Low	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ 55 psig
d. RCIC Turbine Exhaust Diaphragm Pressure – High	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 20 psig
e. RCIC Equipment Area Ambient Temperature – High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.7	≤ 145.9°F
f. Main Steam Line Pipe Tunnel Temperature – High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.7	≤ 158.9°F

(continued)

(b) Required to initiate the drywell isolation function.

(d) During CORE ALTERATIONS, operations with a potential for draining the reactor vessel, and movement of irradiated fuel assemblies in primary containment.

Table 3.3.6.1-1 (page 4 of 6)
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
3. RCIC System Isolation (continued)					
g. Main Steam Line Pipe Tunnel Temperature Timer	1,2,3	1	F	SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.7	≤ 30 minutes
h. RHR Equipment Area Ambient Temperature – High	1,2,3	1 per area	F	SR 3.3.6.1.1 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.7	≤ 159.9°F
i. RCIC/RHR Steam Line Flow – High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 55.6 inches water
j. Drywell Pressure – High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 1.88 psig
k. Manual Initiation	1,2,3	1	G	SR 3.3.6.1.5	NA
4. Reactor Water Cleanup (RWCU) System Isolation					
a. Differential Flow – High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 77.1 gpm
b. Differential Flow – Timer	1,2,3	1	F	SR 3.3.6.1.2 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 10.85 minutes
c. RWCU Heat Exchanger Room Temperature – High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.7	≤ 138.9°F

(continued)

Table 3.3.6.1-1 (page 5 of 6)
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
4. RWCU System Isolation (continued)					
d. RWCU Pump Rooms Temperature – High	1,2,3	1 per room	F	SR 3.3.6.1.1 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.7	≤ 137.9°F
e. RWCU Valve Nest Room Temperature – High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.7	≤ 137.9°F
f. RWCU Demineralizer Valve Room Temperature–High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.7	≤ 143.7°F
g. RWCU Demin Receiving Tank Room Temperature–High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.7	≤ 143.7°F
h. RWCU Demineralizer Room Temperature-High	1,2,3	1 per room	F	SR 3.3.6.1.1 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.7	≤ 143.7°F
i. Main Steam Line Pipe Tunnel Temperature - High	1,2,3	1	F	SR 3.3.6.1.1 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.7	≤ 158.9°F
j. Reactor Vessel Water Level-Low Low, Level 2	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ 127.6 inches
k. Standby Liquid Control System Initiation	1,2	1	I	SR 3.3.6.1.5	NA
l. Manual Initiation	1,2,3	2	G	SR 3.3.6.1.5	NA

(continued)

Table 3.3.6.1-1 (page 6 of 6)
Primary Containment and Drywell Isolation Instrumentation

FUNCTION	APPLICABLE MODES OR OTHER SPECIFIED CONDITIONS	REQUIRED CHANNELS PER TRIP SYSTEM	CONDITIONS REFERENCED FROM REQUIRED ACTION C.1	SURVEILLANCE REQUIREMENTS	ALLOWABLE VALUE
5. RHR System Isolation					
a. RHR Equipment Area Ambient Temperature – High	$2^{(e)}$, $3^{(e)}$	1 per area	F	SR 3.3.6.1.1 SR 3.3.6.1.4 SR 3.3.6.1.5 SR 3.3.6.1.7	≤ 159.9°F
b. Reactor Vessel Water Level-Low, Level 3	1, $2^{(g)}$, $3^{(g)}$	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ 177.1 inches
	$2^{(e)}$, $3^{(e)}$, 4, 5	$2^{(f)}$	J	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≥ 177.1 inches
c. Reactor Vessel Steam Dome Pressure – High	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.6.1.1.3 SR 3.6.1.1.4 SR 3.6.1.1.5	≤ 150 psig
d. Drywell Pressure – High	1,2,3	2	F	SR 3.3.6.1.1 SR 3.3.6.1.2 SR 3.3.6.1.3 SR 3.3.6.1.4 SR 3.3.6.1.5	≤ 1.88 psig
e. Manual Initiation	1,2,3	2	G	SR 3.3.6.1.5	NA

(e) With reactor vessel steam dome pressure less than the RHR cut in permissive pressure.

(f) Only one trip system required in MODES 4 and 5 with RHR Shutdown Cooling System integrity maintained.

(g) With reactor steam dome pressure greater than or equal to the RHR cut in permissive.

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.1.1

Performance of the CHANNEL CHECK once every 12 hours ensures that a gross failure of instrumentation has not occurred. A CHANNEL CHECK is normally a comparison of the parameter indicated on one channel to a similar parameter on other channels. It is based on the assumption that instrument channels monitoring the same parameter should read approximately the same value. Significant deviations between the instrument channels could be an indication of excessive instrument drift in one of the channels or something even more serious. A CHANNEL CHECK will detect gross channel failure; thus, it is key to verifying the instrumentation continues to operate properly between each CHANNEL CALIBRATION.

Agreement criteria are determined by the plant staff, based on a combination of the channel instrument uncertainties, including indication and readability. If a channel is outside the criteria, it may be an indication that the instrument has drifted outside its limit. The Frequency is based on operating experience that demonstrates channel failure is rare.

The CHANNEL CHECK supplements less formal, but more frequent, checks of channels during normal operational use of the displays associated with the channels required by the LCO.

SR 3.3.6.1.2

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The quarterly Frequency is based on reliability analysis described in References 5 and 6.

For Function 1.e, "Main Steam Line Pipe Tunnel Temperature - High", this SR is applicable only to the Division 3 and 4 ambient temperature channels. Divisions 1 and 2 are monitored by digital instrument channels, which are functionally tested on a semiannual basis by SR 3.3.6.1.7.

(continued)

BASES

SURVEILLANCE REQUIREMENTS (continued)

SR 3.3.6.1.6

This SR ensures that the individual channel response times are less than or equal to the maximum values assumed in the accident analysis. Testing is performed only on channels where the assumed response time does not correspond to the diesel generator (DG) start time. For channels assumed to respond within the DG start time, sufficient margin exists in the 10 second start time when compared to the typical channel response time (milliseconds) so as to assure adequate response without a specific measurement test. The instrument response times must be added to the PCIV closure times to obtain the ISOLATION SYSTEM RESPONSE TIME. ISOLATION SYSTEM RESPONSE TIME acceptance criteria are included in References 7 and 8. ISOLATION SYSTEM RESPONSE TIME tests are conducted on an 18 month STAGGERED TEST BASIS. This test Frequency is consistent with the typical industry refueling cycle and is based upon plant operating experience that shows that random failures of instrumentation components causing serious response time degradation, but not channel failure, are infrequent.

SR 3.3.6.1.7

A CHANNEL FUNCTIONAL TEST is performed on each required channel to ensure that the entire channel will perform the intended function.

Any setpoint adjustment shall be consistent with the assumptions of the current plant specific setpoint methodology.

The semiannual Frequency is based on the reduced drift and the design features inherent in digital systems (Ref. 9).

For Function 1.e, "Main Steam Line Pipe Tunnel Temperature - High", this SR is applicable only to the Division 1 and 2 ambient temperature channels. Divisions 3 and 4 are monitored by analog instrument channels, which are functionally tested on a quarterly basis by SR 3.3.6.1.2.

(continued)

BASES

REFERENCES

1. USAR, Section 6.3.
2. USAR, Chapter 15.
3. NEDO-31466, "Technical Specification Screening Criteria Application and Risk Assessment," November 1987.
4. USAR, Section 9.3.5.
5. NEDC-31677-P-A, "Technical Specification Improvement Analysis for BWR Isolation Actuation Instrumentation," June 1989.
6. NEDC-30851-P-A, Supplement 2, "Technical Specifications Improvement Analysis for BWR Isolation Instrumentation Common to RPS and ECCS Instrumentation," March 1989.
7. USAR, Section 15.1.3.
8. USAR, Section 15.6.
9. Letter PY-CEI/NRR-1654L, "License Amendment Request: Replacement of Selected Analog Leak Detection System Instruments with GE NUMAC Leak Detection Monitors," November 22, 1993.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 79 TO FACILITY OPERATING LICENSE NO. NPF-58

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY, ET AL.

PERRY NUCLEAR POWER PLANT, UNIT NO. 1

DOCKET NO. 50-440

1.0 INTRODUCTION

By letter dated November 22, 1993, Cleveland Electric Illuminating (licensee) requested an amendment to License No. NPF-58 to change the technical specifications for the Perry Nuclear Power Plant Unit 1 (plant). By letter dated May 5, 1995, the licensee provided information on the electromagnetic and radio frequency interference mapping done at the plant and a summary of the plant-specific seismic analysis for the General Electric (GE) Nuclear Measurement Analysis and Control (NUMAC) steam leak detection monitor equipment. This proposed amendment reflects the replacement of existing analog Riley leak detection equipment with the digital GE NUMAC leak detection equipment. The proposed amendment also revises technical specification surveillance requirements by reducing the channel functional test surveillance frequency for several area temperature and differential temperature trip functions from monthly to semiannually in Technical Specification Table 4.3.2.1-1, "Isolation Actuation Instrumentation Surveillance Requirements."

By letter December 20, 1995, the licensee submitted Technical Specification pages that took into account previously issued license amendments. The supplemental letters of May 5 and December 20, 1995, did not change the licensee's request or affect the staff's notice of no significant hazards consideration.

The staff evaluated replacement of the existing analog system with digital GE NUMAC equipment, and increasing the CHANNEL FUNCTIONAL TEST surveillance intervals for the associated trip functions of the Reactor Water Cleanup (RWCU), Reactor Core Isolation Cooling (RCIC) (including the Main Steam Line Tunnel Temperature Timer), and Residual Heat Removal (RHR) systems. The staff also evaluated replacement of the Division 1 and 2 logic for main steam line isolation (for the temperature instrumentation located in the steam tunnel area). The proposed change does not include the main steam line temperature isolation logic for the temperature instrumentation located in the turbine building.

2.0 SYSTEM DESCRIPTION

The NUMAC Leak Detection Monitor (LDM) performs safety-related and nonsafety related functions using Class 1E and non-Class 1E components. The safety-related functions of the LDM System:

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- process temperature signals from the ambient and differential temperature sensors of the RWCU system, high pressure coolant injection (HPCI) system, and RCIC system, and
- provide isolation signals by comparing the input signals with preselected setpoints.

The LDM system also:

- measures input current from the thermocouple input unit and performs the specified temperature calculations;
- provides high voltage DC power to the detector;
- provides -15 VDC power to associated electronics;
- provides output trip signals to external equipment;
- performs automatic calibrations;
- performs automatic self-tests and alarms;
- displays self-test status on demand; and
- provides security by keylock and password against unauthorized changes to setpoints.

2.1 Equipment Description

The LDM is a GE NUMAC Class-1E system, with architecture consisting of a family of firmware-based 80C86 (16 bit) and 80386 (32 bit) controllers with application-specific analog and digital modules connected via a NUMAC bus. An independent display controller connects to the Class-1E processor via a serial link and provides the man-machine interface without affecting the Class-1E processes. The NUMAC architecture also includes both hardware and software watchdog timers and an integral self-test system. The LDM chassis is comprised of the following modules:

- Thermocouple Input Unit (TCIU)
- High Speed Parallel Data Bus
- Essential Microcomputer
- Relay Output and Analog Output Modules
- Redundant Instrument Power Supplies
- Serial Data Link

- Display Microcomputer

- Front Panel Interface and Display

The TCIU connects the ambient and differential thermocouples (TCs) to the LDM instrument chassis. The unit contains an isothermal terminal board interface and transmits the temperature measurements to the LDM chassis. Six solid state temperature devices in the TCIU are used for determining the cold junction temperature. The LDM cold calibration may be performed with the TCs in place. Up to six TC input modules may be used to accommodate a maximum of 36 TC inputs. Maintenance is accomplished by module replacement. Each input is assigned to a channel within each input module. The modules are isolated from one another such that each module will not be affected by a failure in a different module.

The high speed parallel data bus provides the communication link between the TCIU, the essential microcomputer, and the relay output and analog output modules.

The essential microcomputer controls instrument measurements, performs trip and I/O functions, communicates with the display microcomputer, and performs the tests of the Self-Test System (STS) when not processing instrument data. The essential microcomputer consists of an 80C86 microprocessor, Random Access Memory (RAM), Electrically Alterable Read Only Memory (EAROM), Read Only Memory (ROM), a priority interrupt controller, independent timers, and the STS circuitry. The microcomputer has sufficient computing power to perform digital trips, digital temperature compensation, automatic ranging, automatic calibration, and digital filtering. The microcomputer automatically calibrates the TC inputs to a known internal reference, thereby compensating time dependent drift characteristics to improve accuracy and resolution. Output trips are set digitally and thus do not drift.

The relay output module interfaces with the NUMAC LDM chassis and provides Isolation, Alarm, INOP, and spare outputs using relay output contacts. The assignment of the output contacts to specific functions is programmable. Channel isolation and alarm functions are automatically bypassed (the state of the assigned output relays does not change) when any of the following activities/conditions occur:

- calibration, calibration check, or trip check activities;

- open conditions in the channel TC; or

- critical self-test faults in the channel.

The bypass is automatically removed when the above conditions no longer exist.

A single INOP/Trouble output relay is provided. The INOP/Trouble relay is tripped in the INOP mode and may be programmed to trip whenever there is a self-test failure condition while in the OPERATE mode. Each relay contact may also be manually tested when the channel is in the INOP mode.

The analog module interfaces analog signals from within the LDM chassis to the functional controller.

The NUMAC LDM has instrument power supplies and detector high voltage power supplies. The instrument power supply provides power to the LDM chassis. Each LDM chassis has two redundant diode-auctioneered low voltage power supplies for uninterruptable power in the event of a power supply failure.

The serial data link provides a one-way, isolated, high-speed output communications link between the safety-related essential microcomputer and the nonsafety related display microcomputer and front panel display. The serial data link is used to minimize the possibility of injecting faults into the essential microcomputer safety-related circuits.

The display microcomputer, which is based on the National Semiconductor NSC-800 microprocessor, processes the data from the essential microcomputer for display on the front panel interface and display module. The front panel interface and display module contains all of the circuitry necessary to interface with the display microcomputer, the front panel keyboard, and electro-luminescent display.

2.2 Improvements

By replacing the existing LDM with the NUMAC LDM, the licensee expects to improve the reliability and accuracy of the leak detection function. The current analog LDM uses temperature switch modules that have experienced a high drift rate, have been prone to spurious alarms and trips, and have been difficult to maintain. NRC Information Notice 86-69, "Spurious System Isolations Caused By the Panalarm Model 86 Thermocouple Monitor," and General Electric Service Information Letter (SIL) Number 416 describe problems with the Riley temperature modules that have resulted in spurious system isolations.

The ambient and differential temperature monitoring functions are performed continuously by the Riley temperature modules and are independent of the operation of any controls. When the temperature (or differential temperature) being monitored is in the alarm condition, relay contacts are closed, which automatically initiates further actions such as annunciations or system isolations. Indication to the operator depends on the operation of a "READ/SET" switch, which causes the output of each thermocouple "point" module to be indicated on a separate meter module. The electrical transient caused by the operation of this "READ/SET" switch has been the predominant source of the spurious isolations.

Calibration of the Riley temperature modules is time consuming and has a high potential for creating spurious ESF actuations. Over thirty channel functional surveillance tests of the Riley temperature modules are performed monthly. Testing the Riley ambient and differential temperature modules requires lifting the thermocouple leads and adding jumpers around the isolation relays. Installation of jumpers results in unnecessary isolations and increases the risk of technician errors. Lifting leads is labor intensive, and also could cause a spurious actuation if the leads are incorrectly installed after performing the tests. The instrumentation and controls (I&C) technician can inadvertently ground/short a jumper, a thermocouple wire, or a component while working within the cabinet, leading to an isolation. Additionally, this lead lifting procedure can cause broken thermocouple lead wires.

The Riley leak detection temperature modules have created problems for the operators and I&C technicians in terms of isolations and half isolation signals. A half isolation in the two-division portion of the leak detection system (RCIC, RHR and RWCU Systems) causes an isolation of the flow path to/from the containment, since one of the containment isolation valves in the penetration flow path closes. This reduces the reliability and availability of these systems.

The digital NUMAC LDM contains design features that address the Riley analog temperature module monitoring system problems described above. Compared to the existing analog Riley temperature modules, the NUMAC LDM will reduce the number of temperature switch module instrument drift problems because use of digital equipment with setpoints established in software avoids drift concerns. Since the NUMAC LDM is designed to facilitate channel surveillances without lifting leads or installing jumpers, spurious actuations caused by these actions will be eliminated. In addition, the resultant maintenance activities and potential problems associated with performing channel surveillances will also be reduced by the built in self-test features.

3.0 REVIEW CRITERIA

The NUMAC LDM is a safety-related monitoring instrumentation system. Therefore, General Design Criteria (GDC) 2, 4, 13 and 19, IEEE Standard 279, "Criteria for Protection Systems for Nuclear Power Generation Stations" (10 CFR 50.55 a(h)), and the applicable acceptance criteria in Section 7.5, "Information Systems Important to Safety" of the Standard Review Plan (NUREG-0800) were used as review guidance. Additionally, ANSI/IEEE Standard 7-4.3.2-1982, "Application Criteria for Programmable Digital Computer Systems in Safety Systems of Nuclear Power Generating Stations," and corresponding Regulatory Guidance (RG) 1.152, "Criteria for Programmable Digital Computer System Software in Safety Related Systems of Nuclear Power Plants," were used to evaluate the NUMAC LDM system software design verification and validation processes.

4.0 SYSTEM EVALUATION

The staff reviewed the qualification of the NUMAC LDM hardware and software components, defense against common mode failure, and the training of the operations and maintenance personnel. The results of these reviews are discussed in the following sections.

4.1 Hardware

Part 50, Appendix A, GDC 2 and 4 require that the safety system be designed to withstand the effects of natural phenomena and be qualified to operate in its environment under normal and postulated accident conditions. To ensure that the NUMAC LDM will perform its intended function(s) under design basis conditions, the staff reviewed the environmental qualification of the NUMAC equipment for (1) temperature and humidity, (2) seismic, (3) radiation, and (4) electromagnetic and radio frequency interference.

4.1.1 Temperature and Humidity

GE performed temperature and humidity tests on the NUMAC instrument chassis and associated modules. The test procedures were the same as documented in Appendix C of NEDC-31974P, "Qualification Report for Nuclear Measurement and Control for Reactor Building Vents Radiation Monitor System for TVA Browns Ferry Nuclear Plant, Units 1, 2 and 3" which was previously reviewed and approved by the staff. The staff used IEEE Std 323-1974, "IEEE Standard for Qualifying Class 1E Equipment for Nuclear Power Generating Stations," as review guidance for temperature and humidity qualification.

GE qualified the NUMAC LDM equipment by testing equipment unique to the LDM and analyzing equipment similar to the NUMAC product line equipment. The NUMAC LDM environmental tests consisted of component aging, printed circuit board (module) qualification, instrument qualification, and instrument heat rise. The NUMAC LDM instrument is qualified for continuous operation between 40°F and 122°F and between 10% and 90% noncondensing humidity. The NUMAC LDM modules also demonstrated satisfactory performance up to 158°F. Maximum LDM instrument internal heat rise was measured at 18°F in the vicinity of the power supplies.

The licensee stated that the design ranges for the plant control room temperature and humidity where the NUMAC LDM equipment will be located are:

Temperature:	Normal 64 - 77°F	(75°F Avg.)
	Design Basis Accident	(87°F Avg.)
Humidity:	Normal 20 - 90% RH	(50% Avg.)
	Design Basis Accident	(87% Avg.)

The control room average air temperature is maintained at approximately 75°F, and is required to be observed and recorded daily as directed in plant procedure OI-03.4 "Control Operator Daily Check Sheet." If the temperature exceeds 77°F, the surveillance frequency is increased to once per shift. As part of the licensee's response to the station blackout rule, 10 CFR 50.63, the licensee committed to implement around-the-clock HVAC trouble shooting activities when temperatures exceed 84°F. The basis for the 84°F setpoint was a calculation that demonstrated the control room temperature would not exceed 120°F within one hour after loss of HVAC power starting at an initial ambient temperature of 85°F. Thus, the NUMAC LDM qualification temperature bounds the normal and accident environments. In addition, the margin between the normal control room general area ambient temperature and the NUMAC qualification temperature is adequate to accommodate potential local heating effects inside the panel containing the LDM modules.

Humidity is not a directly controlled parameter at the plant; however, the design humidity range stated above is bounded by the 10% to 90% humidity range to which the NUMAC LDM is qualified.

Based on the foregoing review, the staff finds that the GE temperature and humidity qualification of the GE NUMAC product envelops the licensee's plant specific temperature and humidity conditions, and meets the guidance of IEEE Std 323-1974. Therefore, the staff finds the temperature and humidity qualification acceptable.

4.1.2 Seismic Qualification

The LDM equipment and panels that replace existing equipment and panels are safety-related seismic Category 1 components. GE performed a similarity analysis of the NUMAC LDM chassis and interface panels against the existing equipment for the licensee. GE performed the similarity analysis to show that the plant specific NUMAC LDM devices are mechanically the same or equivalent to devices GE tested. The results of the GE study are summarized in Reference 2. The GE study demonstrated that the NUMAC LDM seismic qualification envelopes the required response spectra at the plant site. The seismic qualification study was performed in accordance with IEEE Std 344-1987, "IEEE Recommended Practice for Seismic Qualification of Class 1E Equipment for Nuclear Power Generating Stations." Therefore, the staff finds the seismic qualification acceptable.

4.1.3 Radiation

The LDM components located in the control room were qualified to a maximum total integrated dose (TID) of 1E+4 rad. This is within the plant normal and accident radiation doses for the associated areas. The test procedures and the test results are documented in NEDC-31974P, Appendix C, which was previously approved by the staff. The staff finds the radiation qualification to be acceptable.

4.1.4 EMI & RFI

Electromagnetic interference and radio frequency interference (EMI/RFI) are random noise produced by systems within the operating environment of the plant. This random noise can affect the safety of the plant since it can potentially lead to common cause failure of redundant safety-related equipment that is vulnerable to the noise.

RFI. Specific guidance and information for the review of EMI/RFI is contained in the following standards and documents:

1. MIL-STD-461(A,B,C), "Electro-magnetic Emission and Susceptibility Requirements for the Control of Electro-magnetic Interference."

2. MIL-STD-462, "Electro-magnetic Interference Characteristics Measurement."

3. MIL-STD-1399, "Interface Standard for Shipboard Systems, DC Magnetic Field Environment."

4. SAMA PMC 33.1-1978, "Electro-magnetic Susceptibility of Process Control Instrumentation."

5. IN 83-83, "Use of Portable Radio Transmitters Inside Nuclear Power Plants."

6. IEC 801-2 "Electromagnetic Compatibility for Industrial-Process Measurement and Control Equipment Part 2: Electrostatic Discharge Requirements."

7. NUREG CR-3270, "Investigation of Electro-magnetic Interference (EMI) Levels in Commercial Nuclear Power Plants."

Using the above as guidance, the staff reviewed the licensee's EMI/RFI environmental qualification of the NUMAC LDM. This review included the EMI/RFI qualification methodology and range of frequencies tested, and the EMI/RFI qualification process for the installed equipment. The qualification of the installed equipment consisted of a site survey to verify that the installed equipment qualification envelopes its EMI/RFI frequency/amplitude environment.

GE tested the LDM for susceptibility to radiated electric fields, radiated magnetic fields, conductive noise, and static discharges using test methodologies from the various standards identified previously. The EMI/RFI test results for the NUMAC LDM are documented in NEDC-31974P, Appendix B.

The radiated electric field susceptibility test was conduced in accordance with SAMA Standard PMC 33.1, with a field strength of 65 V/m over a frequency range of 20 MHz to 990 MHz. In addition, a keying test was performed to simulate the keying of a walkie talkie. The test results demonstrated proper radiated electric field protection.

The radiated magnetic field susceptibility test was conducted in accordance with GE test requirements. The test required fifty-foot wires to be attached to the inputs/outputs of the equipment being tested, and signals from a generator injected into the test wires to simulate the noise induced on the power leads. The results demonstrated qualification against radiated magnetic fields.

The conductive noise test was done in accordance with Swedish Standard Svensk Standard SS 436 15 03 and GE's conductive noise test requirements. Appropriate conductive noise protection was demonstrated.

The NUMAC LDM was qualified by GE against electrostatic discharge effects per IEC Standard 801-2. Protection against electrostatic discharge was effectively demonstrated.

In addition, the staff requested that the licensee confirm by site survey or analysis that the electromagnetic environment at the installation site is within the tested envelope. The licensee performed additional EMI/RFI testing for the frequency ranges that have not been covered by the earlier tests documented in NDC-31974P. The licensee conducted on-site EMI/RFI mapping of the NUMAC LDM equipment locations and surrounding areas. The tests confirmed that the EMI/RFI qualification envelop bounds the plant specific noise environments for the NUMAC LDM equipment.

The staff finds that the above testing methodologies and results demonstrated proper qualification of the NUMAC LDM against EMI/RFI effects, and is, therefore, acceptable.

4.2 Software

The LDM application software consists of the functional software for the microcomputer, including the self-test system, and the front panel keyboard and display software for the display computer. The LDM digital equipment software is written in high level languages to the maximum extent possible to simplify software maintenance over the lifetime of the equipment. The total lines of the code required to perform the LDM functions is under 20,000 lines. The functions performed by the software include (1) sampling and filtering sensor data, (2) comparing data to operator defined trip setpoints, (3) updating operator displays, (4) generating analog and trip output signals, and (5) performing self-tests.

The staff's software review included an audit of the products resulting from the software lifecycle plans for the NUMAC LDM, and examination of the NUMAC generic software development process with particular attention to the software management plan (SMP), the software configuration management plan (SCMP), and the software verification and validation plan (SVVP). The staff examined these plans and their implementation with regard to the guidance provided in RG 1.152 and ANSI/IEEE-7-4.3.2-1982.

GE used safety-related quality programs with supplemental verification and validation (V&V) procedures based on RG 1.152 to develop both Class 1E and non-Class 1E NUMAC LDM software. The GE NUMAC line of instruments is highly modularized and uses NUMAC product code where appropriate. The lines of code are stored in three sets of firmware.

The GE V&V method is based on logical steps with baseline reviews performed at the completion of each phase of the development process. A list of open items is documented and maintained for each review. The V&V reviewers are independent from the software designers and communicated their review in written reports. The validation process includes a matrix relating each validation test to a functional requirement.

The GE software development method is consistently followed and provides an internally reviewed paper trail throughout the software development process. Testing is done using emulators, and every change requires testing. An organizationally independent configuration control engineer is required to sign-off on all baseline reviews (verification steps) and controls the NUMAC library of documents and firmware. The NUMAC review team has nine members and must approve all changes for resolutions of open items.

Strict configuration control standards are in place and all updates to the NUMAC instruments are performed at GE. Each version of the firmware includes all software modules. Each version is controlled with a separate revision and part number. The User's Manual contains an extensive description of the NUMAC LDM system as well as instructions for its use.

The staff finds that GE has a formally established software design, code and test review process with associated formal documentation. The staff also confirmed that (1) the GE formal configuration management plan is being consistently applied, (2) GE maintains a library wherein each software revision is a complete entity which addresses problems associated with controlling different versions of the code for each customer, and (3) an independent software V&V process was implemented in accordance with RG 1.152. The staff, therefore, finds the NUMAC LDM software is acceptable.

4.3 Defense Against the Common Mode Failure

GDC 13 and 19 require that instrumentation be provided for monitoring necessary variables under normal and accident conditions with indication provided in the control room. The staff reviewed the NUMAC LDM to confirm that this capability continued to be provided.

The single failure criterion requires that any single failure within a safety-related system not prevent proper function at the system level. The NUMAC LDM consists of redundant channels in order to meet this requirement. However, common mode and common cause failures can prevent performance of intended functions even in redundant channels. A microprocessor-based digital system, such as the NUMAC LDM, which shares data bases and process equipment, has a potential for common mode or common cause failures in the software, hardware, and software/hardware interaction. Defense against common mode and common cause failures in digital systems is provided by quality and diversity in the system design.

Quality in the NUMAC LDM design is demonstrated as discussed previously in sections 4.1 and 4.2 of this Safety Evaluation Report. Diversity is provided to the ambient and differential temperature monitoring trip functions of the NUMAC LDM temperature-based isolation functions for the various systems by alternative leak detection methods as described in Sections 7.3.1.1.2, 7.6.1.3and 7.3.1 of the Perry Updated Safety Analysis Report (USAR). The alternative leak detection methods are:

System	Alternative Leak Detection Method
RCIC	RCIC steam line flow RCIC steam line pressure
RHR	RHR/RCIC steam line flow
RWCU	RWCU differential flow Reactor water level
MSL	MSL high flow MSL low pressure

These functions are physically separate from those of the NUMAC LDM and constitute diverse, redundant, safety-related backup means, with indication in the control room, that are capable of responding to a design basis line break for the various systems. Nonsafety sump level alarms are also available to the operator. A common mode failure of both divisions of the NUMAC LDM instrumentation would, therefore, not prevent the necessary detection of system line breaks.

In the event of a failure of the NUMAC LDM self-diagnostic capability, the NUMAC architecture includes an INOP relay output contact that causes a System Test/Trouble control room overhead annunciator to alarm under the following circumstances:

- loss of external or internal power to the NUMAC,
- placing the keylock switch out of OPERATE,
- failure of a hardware module during self-test diagnostics (performed approximately each 1-2 minutes in the LDM),
- detection of an open thermocouple or flow transmitter signal circuit,
- failure of the class-1E processor to update the hardware watchdog on regular intervals,
- any software task which is not running at its expected intervals.

In the event of a NUMAC LDM common-mode failure, the operator can perform the LDM actuation functions manually in the control room using the information available from the radiation indicators and alarms in the control room. This capability is diverse from the LDM.

Based on the above review, the staff finds that the NUMAC LDM defenses against common mode and common cause failures are adequate to ensure LDM functions from the control room, and provide acceptable redundancy in the event of loss of both divisions of the NUMAC LDM temperature-based isolation functions. Therefore, the NUMAC LDM meets the requirements of GC 13 and 19, and is acceptable.

4.4 Training

The licensee will provide specific training for operations and maintenance personnel on the NUMAC LDM. The maintenance personnel will receive either Level 1 (1-2 hour) or Level 2 (½ to 1 day) courses on changes to the leak detection system, including changes to the physical configuration, technical specifications, manuals and drawings, and surveillance procedures. Level 2 training will include in-depth training on hardware, software, testing, and diagnostics. Both levels will include hands-on experience in the NUMAC training unit.

For the operations staff, all supervisors and on-shift control room operators will receive an overview of the changes to the leak detection system, including changes to the physical configuration, technical specifications, manuals, drawings, and surveillance procedures. They will also receive training on the changes to technical specification criteria and operational procedures. This training will also include hands-on experience in the NUMAC training unit. The staff finds the licensee's training commitments to be acceptable.

5.0 TECHNICAL SPECIFICATION CHANGES

The licensee requested an amendment to revise the technical specifications to reflect the replacement of existing leak detection equipment with GE NUMAC leak detection equipment, and to revise certain surveillance requirements for leak detection instrumentation. The upgrade from analog to digital devices improves several system parameters such as channel accuracy, drift, and loop calibration; however, the means of testing a digital channel is no different from the means of testing an analog channel, except in the data transmission from the sensor. Adding "digital" to the DEFINITION of the CHANNEL FUNCTIONAL TEST along with the currently referenced "analog" channels clarifies that the same test applies to digital channels. The staff concurs with this change.

The requested technical specification change also increases the CHANNEL FUNCTIONAL TEST surveillance interval from M (Monthly) to SA (Semiannually) for the ambient and differential temperature trip functions for the RWCU (including the Main Steam Line Tunnel Temperature Timer), RHR systems, and the Division 1 and 2 logic for main steam line isolation (for the temperature instrumentation located in the Steam Tunnel area). The change does not include the main steam line temperature isolation logic for the temperature instrumentation located in the Turbine Building.

The licensee requested the following specific technical specification changes:

a. Specification 1.6: CHANNEL FUNCTIONAL TEST (page 1-1)

Revise part "a" of the CHANNEL FUNCTIONAL TEST definition to recognize that this same definition applies to testing performed on digital channels. Add "/Digital" after the word "Analog" in Definition 1.6.a.

b. Specification 3.3.2: ISOLATION ACTUATION INSTRUMENTATION: MAIN STEAM LINE ISOLATION, TRIP FUNCTIONS 2.f and 2.g (page 3/4 3-23)

Increase the CHANNEL FUNCTIONAL TEST surveillance interval from M (Monthly) to SA (Semiannually) for the temperature and differential temperature Trip Functions listed below.

Current Item No.	New Item No.	TRIP FUNCTION (SYSTEM) ISOLATION (PAGE) Trip Function Name	Channel Functional Test Req'mt
		MAIN STEAM LINE ISOLATION (Page 3/4 3-23)	
2.f	2.f.1	Main Steam Line Tunnel Temperature - High (Division 1 and 2)	SA
2.g	2.g.1	Main Steam Line Tunnel Δ Temperature - High (Division 1 and 2)	SA

c. Specification 3.3.2: ISOLATION ACTUATION INSTRUMENTATION (pages 3/4 3-24 through 3/4 3-26)

Increase the CHANNEL FUNCTIONAL TEST surveillance interval from M (Monthly) to SA (Semiannually) for the temperature and differential temperature Trip Functions listed below.

TRIP FUNCTION (SYSTEM) ISOLATION (PAGE) Item No.	Trip Function Name	Channel Functional Test Req'mt
	REACTOR WATER CLEANUP SYSTEM ISOLATION (Page 3/4 3-24)	
4.c	Equipment Area Temperature - High	SA
4.d	Equipment Area Ventilation Δ Temperature - High	SA
4.f	Main Steam Line Tunnel Ambient Temperature - High	SA
4.g	Main Steam Line Tunnel Δ Temperature - High	SA

TRIP FUNCTION (SYSTEM) ISOLATION (PAGE) Item No.	Trip Function Name	Channel Functional Test Req'mt
	REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION (Page 3/4 3-25)	
5.d	RCIC Equipment Room Ambient Temperature - High	SA
5.f	Main Steam Line Tunnel Ambient Temperature - High	SA
5.g	Main Steam Line Tunnel Δ Temperature - High	SA
5.h	Main Steam Line Tunnel Temperature Timer	SA
5.i	RHR Equipment Room Ambient Temperature - High	SA
5.j	RHR Equipment Room Δ Temperature - High	SA
	RHR SYSTEM ISOLATION (Page 3/4 3-26)	
6.a	RHR Equipment Area Ambient Temperature - High	SA
6.b	RHR Equipment Area Δ Temperature - High	SA

The basis for the increase in the channel functional test surveillance interval from M (Monthly) to SA (Semi-Annual) is that the NUMAC LDM self-test feature can detect potential failures that the periodic channel functional tests are intended to identify, and that the continuous self-test alerts the operator via annunciation when a problem is detected. The following NUMAC LDM diagnostic and self-test features are provided:

- Continuous monitoring of each flow and each density compensation input signal for out-of-bounds values.
- Continuous monitoring of the two internal power supplies (NUMAC remains functional with only one internal power supply).
- Continuous monitoring of the external power input.
- A self-check of each channel to confirm functionality at least once per 30 minutes.
- Continuous monitoring to assure that the system is not left in an inoperable condition (e.g., card out-of-file, status switch left in the INOP mode).

Based on the staff review of the proposed TS changes, the staff finds that the NUMAC LDM self diagnostic features provide appropriate justification for relaxation in channel functional test frequency, and therefore, the TS changes are acceptable.

Based on the above, the staff concludes that the design changes related to the replacement of the existing LDM systems with the GE NUMAC LDM, and the associated technical specification changes, meet the requirements of GDC 2, 4, 13 and 19, IEEE 279 for safety-related instrumentation and control systems, and the guidelines of RG 1.152 for computer-based digital systems, and are, therefore, acceptable.

6.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Ohio State official was notified of the proposed issuance of the amendment. The State official had no comments.

7.0 ENVIRONMENTAL CONSIDERATION

This amendment involves a change to a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or a change to a surveillance requirement. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluent 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding (59 FR 24752). Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

8.0 CONCLUSION

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

Principal Contributor: M. Waterman

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