April 22, 1994

Docket No. 50-440

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Mr. Robert A. Stratman
Vice President Nuclear - Perry
Centerior Service Company
P. O. Box 97, S270
Perry, Ohio 44081

Dear Mr. Stratman:

SUBJECT: AMENDMENT NO. 58 TO FACILITY OPERATING LICENSE NO. NPF-58 (TAC NO. M84804)

The Commission has issued the enclosed Amendment No. 58 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 September 28, 1992.

This amendment removes the functions associated with the main steam line radiation monitors from Technical Specification (TS) 2.2, Limiting Safety System Settings, TS 3.3.1, Reactor Protection System Instrumentation, and TS 3.3.2, Isolation Actuation Instrumentation. This change is related to Licensing Topical Report NEDO-31400, "Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Line Isolation Valve Closure Function and Scram Function of the Main Steam Line Radiation Monitor (May 1987)." and the associated NRC Saftey Evaluation Report of May 15, 1991.

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

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

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

cc w/enclosures:
See next page

JBH 4-20-94

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DOCUMENT NAME: G:\PERRY\PER84804.AMD

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

April 22, 1994

Docket No. 50-440

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Vice President Nuclear - Perry
Centerior Service Company
P. O. Box 97, S270
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Office of Nuclear Reactor Regulation

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2. Safety Evaluation

cc w/enclosures:
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

THE CLEVELAND ELECTRIC ILLUMINATING COMPANY, ET AL.

DOCKET NO. 50-440

PERRY NUCLEAR POWER PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No.58
License No. NPF-58

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

   A. The application for amendment by The Cleveland Electric Illuminating Company, Centerior Service Company, Duquesne Light Company, Ohio Edison Company, Pennsylvania Power Company, and Toledo Edison Company (the licensees) dated September 28, 1992, 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;

   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. NPF-58 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A and the Environmental Protection Plan contained in Appendix B, as revised through Amendment No. 58 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 90 days after issuance.

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: April 22, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 58

FACILITY OPERATING LICENSE NO. NPF-58

DOCKET NO. 50-440

Replace the following pages of the Appendix "A" Technical Specifications with the attached pages. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. Overleaf pages are provided to maintain document completeness.

Remove	Insert
2-4	2-4
B 2-8	B 2-8
3/4 3-2	3/4 3-2
3/4 3-4	3/4 3-4
3/4 3-6	3/4 3-6
3/4 3-7	3/4 3-7
3/4 3-11	3/4 3-11
3/4 3-15	3/4 3-15
3/4 3-16	3/4 3-16
3/4 3-21	3/4 3-21
3/4 3-23	3/4 3-23
3/4 3-26	3/4 3-26

TABLE 2.2.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
1. Intermediate Range Monitor		
a. Neutron Flux-High	≤ 120/125 divisions of full scale	≤ 122/125 divisions of full scale
b. Inoperative	NA	NA
2. Average Power Range Monitor:		
a. Neutron Flux-High Setdown	≤ 15% of RATED THERMAL POWER	≤ 20% of RATED THERMAL POWER
b. Flow Biased Simulated Thermal Power-High		
1) Flow Biased	≤ 0.66 W+64%, with a maximum of	≤ 0.66 W+67%, with a maximum of
2) High Flow Clamped	≤ 111.0% of RATED THERMAL POWER	≤ 113.0% of RATED THERMAL POWER
c. Neutron Flux-High	≤ 118.0% of RATED THERMAL POWER	≤ 120.0% of RATED THERMAL POWER
d. Inoperative	NA	NA
3. Reactor Vessel Steam Dome Pressure - High	≤ 1064.7 psig	≤ 1079.7 psig
4. Reactor Vessel Water Level - Low, Level 3	≥ 177.7 inches above top of active fuel*	≥ 177.1 inches above top of active fuel*
5. Reactor Vessel Water Level-High, Level 8	≤ 219.5 inches above top of active fuel*	≤ 220.1 inches above top of active fuel*
6. Main Steam Line Isolation Valve - Closure	≤ 8% closed	≤ 12% closed
7. Deleted		
8. Drywell Pressure - High	≤ 1.68 psig	≤ 1.88 psig

*See Bases Figure B 3/4 3-1.

2.2 LIMITING SAFETY SYSTEM SETTINGS

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS

2.2.1 The reactor protection system instrumentation setpoints shall be set consistent with the Trip Setpoint values shown in Table 2.2.1-1.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

With a reactor protection system instrumentation setpoint less conservative than the value shown in the Allowable Values column of Table 2.2.1-1, declare the channel inoperable and apply the applicable ACTION statement requirement of Specification 3.3.1 until the channel is restored to OPERABLE status with its setpoint adjusted consistent with the Trip Setpoint value.

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

Average Power Range Monitor (Continued)

5% of RATED THERMAL POWER per minute and the APRM system would be more than adequate to assure shutdown before the power could exceed the Safety Limit. The 15% neutron flux trip remains active until the mode switch is placed in the Run position.

The APRM trip system is calibrated using heat balance data taken during steady state conditions. Fission chambers provide the basic input to the system and therefore the monitors respond directly and quickly to changes due to transient operation for the case of the Neutron Flux-High setpoint; i.e, for a power increase, the THERMAL POWER of the fuel will be less than that indicated by the neutron flux due to the time constants of the heat transfer associated with the fuel. For the Flow Biased Simulated Thermal Power-High setpoint, a time constant specified in the COLR is introduced into the flow biased APRM in order to simulate the fuel thermal transient characteristics. A more conservative maximum value is used for the flow biased setpoint as shown in Table 2.2.1-1.

The APRM setpoints were selected to provide adequate margin for the Safety Limits and yet allow operating margin that reduces the possibility of unnecessary shutdown.

3. Reactor Vessel Steam Dome Pressure-High

High pressure in the nuclear system could cause a rupture to the nuclear system process barrier resulting in the release of fission products. A pressure increase while operating will also tend to increase the power of the reactor by compressing voids thus adding reactivity. The trip will quickly reduce the neutron flux, counteracting the pressure increase. The trip setting is slightly higher than the operating pressure to permit normal operation without spurious trips. The setting provides for a wide margin to the maximum allowable design pressure and takes into account the location of the pressure measurement compared to the highest pressure that occurs in the system during a transient. This trip setpoint is effective at low power/flow conditions when the turbine control valve fast closure and turbine stop valve closure trips are bypassed. For a load rejection or turbine trip under these conditions, the transient analysis indicated an adequate margin to the thermal hydraulic limit.

REACTOR PROTECTION SYSTEM INSTRUMENTATION SETPOINTS (Continued)

4. Reactor Vessel Water Level-Low

The reactor vessel water level trip setpoint has been used in transient analyses dealing with coolant inventory decrease. The scram setting was chosen far enough below the normal operating level to avoid spurious trips but high enough above the fuel to assure that there is adequate protection for the fuel and pressure limits.

5. Reactor Vessel Water Level-High

A reactor scram from high reactor water level, approximately two feet above normal operating level is intended to offset the addition of reactivity effect associated with the introduction of a significant amount of relatively cold feedwater. An excess of feedwater entering the vessel would be detected by the level increase in a timely manner. This scram feature is only effective when the reactor mode switch is in the Run positionbecause at THERMAL POWER levels below 10% to 15% of RATED THERMAL POWER, the approximate range of power level for changing to the Run position, the safety margins are more than adequate without a reactor scram.

6. Main Steam Line Isolation Valve-Closure

The main steam line isolation valve closure trip was provided to limit the amount of fission product release for certain postulated events. The MSIV's are closed automatically from measured parameters such as high steam flow, low reactor water level, high steam tunnel temperature and low steam line pressure. The MSIV's closure scram anticipates the pressure and flux transients which could follow MSIV closure and thereby protects reactor vessel pressure and fuel thermal/hydraulic Safety Limits.

7. Deleted

3/4.3 INSTRUMENTATION

3/4.3.1 REACTOR PROTECTION SYSTEM INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.1 As a minimum, the reactor protection system instrumentation channels shown in Table 3.3.1-1 shall be OPERABLE with the REACTOR PROTECTION SYSTEM RESPONSE TIME as shown in Table 3.3.1-2.

APPLICABILITY: As shown in Table 3.3.1-1.

ACTION:

a. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for one trip system, place the inoperable channel(s) and/or that trip system in the tripped condition* within 1 hour. The provisions of Specification 3.0.4 are not applicable.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip System requirement for both trip systems, place at least one trip system** in the tripped condition within one hour and take the ACTION required by Table 3.3.1-1.

SURVEILLANCE REQUIREMENTS

4.3.1.1 Each reactor protection system instrumentation channel shall be demonstrated OPERABLE by the performance of the CHANNEL CHECK, CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION operations for the OPERATIONAL CONDITIONS and at the frequencies shown in Table 4.3.1.1-1.

4.3.1.2 LOGIC SYSTEM FUNCTIONAL TESTS and simulated automatic operation of all channels shall be performed at least once per 18 months.

4.3.1.3 The REACTOR PROTECTION SYSTEM RESPONSE TIME of each reactor trip functional unit shown in Table 3.3.1-2 shall be demonstrated to be within its limit at least once per 18 months. Each test shall include at least one channel per trip system such that all channels are tested at least once every N times 18 months where N is the total number of redundant channels in a specific reactor trip system.

4.3.1.4 The provisions of Specification 4.0.4 are not applicable to the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION surveillances for the Intermediate Range Monitors for entry into their applicable OPERATIONAL CONDITIONS (as shown in Table 4.3.1.1-1) from OPERATIONAL CONDITION 1, provided the surveillances are performed within 12 hours after such entry.

---

*An inoperable channel need not be placed in the tripped condition where this would cause the Trip Function to occur. In these cases, the inoperable channel shall be restored to OPERABLE status within 2 hours or the ACTION required by Table 3.3.1-1 for that Trip Function shall be taken.

**The trip system need not be placed in the tripped condition if this would cause the Trip Function to occur. When a trip system can be placed in the tripped condition without causing the Trip Function to occur, place the trip system with the most inoperable channels in the tripped condition; if both systems have the same number of inoperable channels, place either trip system in the tripped condition.

TABLE 3.3.1-1
REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
1. Intermediate Range Monitors:			
a. Neutron Flux - High	2 3, 4 5[b]	3 3 3	1 2 3
b. Inoperative	2 3, 4 5	3 3 3	1 2 3
2. Average Power Range Monitor [c]:			
a. Neutron Flux - High, Setdown	2 3 5[b]	3 3 3	1 2 3
b. Flow Biased Simulated Thermal Power - High	1	3	4
c. Neutron Flux - High	1	3	4
d. Inoperative	1, 2 3 5	3 3 3	1 2 3
3. Reactor Vessel Steam Dome Pressure - High	1, 2[d]	2	1
4. Reactor Vessel Water Level - Low, Level 3	1, 2	2	1
5. Reactor Vessel Water Level - High, Level 8	1[e]	2	4
6. Main Steam Line Isolation Valve - Closure	1[e]	4	4
7. Deleted			
8. Drywell Pressure - High	1, 2[f]	2	[illegible]

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

FUNCTIONAL UNIT	APPLICABLE OPERATIONAL CONDITIONS	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	ACTION
9. Scram Discharge Volume Water Level - High			
a. Level Transmitter	1, 2	2	1
	$5^{(g)}$	2	3
b. Float Switches	1, 2	2	1
	$5^{(g)}$	2	3
10. Turbine Stop Valve - Closure	$1^{(h)}$	4	6
11. Turbine Control Valve Fast Closure, Valve Trip System Oil Pressure - Low	$1^{(h)}$	2	6
12. Reactor Mode Switch Shutdown Position	1, 2	2	1
	3, 4	2	7
	5	2	3
13. Manual Scram	1, 2	2	1
	3, 4	2	8
	5	2	9

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

ACTION

ACTION 1 - Be in at least HOT SHUTDOWN within 12 hours.

ACTION 2 - Verify all insertable control rods to be inserted in the core and lock the reactor mode swithch in the Shutdown position within one hour.

ACTION 3 - Suspend all operations involving CORE ALTERATIONS* and insert all insertable control rods within one hour.

ACTION 4 - Be in at least STARTUP within 6 hours.

ACTION 5 - Deleted.

ACTION 6 - Initiate a reduction in THERMAL POWER within 15 minutes and reduce turbine first stage pressure to less than the automatic bypass setpoint within 2 hours.

ACTION 7 - Verify all insertable control rods to be inserted within one hour.

ACTION 8 - Lock the reactor mode switch in the Shutdown position within one hour.

ACTION 9 - Suspend all operations involving CORE ALTERATIONS* and insert all insertable control rods and lock the reactor mode switch in the Shutdown position within one hour.

*Except replacement of LPRM strings provided SRM instrumentation is OPERABLE per Specification 3.9.2.

TABLE 3.3.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION

TABLE NOTATIONS

(a) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one OPERABLE channel in the same trip system is monitoring that parameter.

(b) Unless adequate shutdown margin has been demonstrated per Specification 3.1.1 and the "one-rod-out" Refuel position interlock has been demonstrated OPERABLE per Specification 3.9.1, the shorting links shall be removed from the RPS circuitry prior to and during the time any control rod is withdrawn.*

(c) An APRM channel is inoperable if there are less than 2 LPRM inputs per level or less than 14 LPRM inputs to an APRM channel.

(d) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.

(e) This function shall be automatically bypassed when the reactor mode switch is not in the Run position.

(f) This function is not required to be OPERABLE when DRYWELL INTEGRITY is not required.

(g) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(h) This function is automatically bypassed when turbine first stage pressure is less than the value of turbine first stage pressure corresponding to 40%** of RATED THERMAL POWER.

*Not required for control rods removed per Specification 3.9.10.1 or 3.9.10.2.

**The Turbine First Stage Pressure Bypass Setpoints and corresponding Allowable Values are adjusted based on Feedwater temperatures (see 3/4.2.2 for definition of ΔT). The Setpoints and Allowable Values for various ΔTs are as follows:

T(°F)	Setpoint (psig)	Allowable Value (psig)
$0 = T$	≤ 212	≤ 218
$0 < \Delta T \leq 50$	≤ 190	≤ 196
$50 < \Delta T \leq 100$	≤ 168	≤ 174
$100 < \Delta T \leq 170$	≤ 146	≤ 152

TABLE 3.3.1-2

REACTOR PROTECTION SYSTEM RESPONSE TIMES

FUNCTIONAL UNIT	RESPONSE TIME (Seconds)
1. Intermediate Range Monitors:	
a. Neutron Flux - High	NA
b. Inoperative	NA
2. Average Power Range Monitor*:	
a. Neutron Flux - High, Setdown	NA
b. Flow Biased Simulated Thermal Power - High	≤ 0.09**
c. Neutron Flux - High	≤ 0.09
d. Inoperative	NA
3. Reactor Vessel Steam Dome Pressure - High	≤ 0.35
4. Reactor Vessel Water Level - Low, Level 3	≤ 1.05
5. Reactor Vessel Water Level - High, Level 8	≤ 1.05
6. Main Steam Line Isolation Valve - Closure	≤ 0.06
7. Deleted	
8. Drywell Pressure - High	NA
9. Scram Discharge Volume Water Level - High	NA
10. Turbine Stop Valve - Closure	≤ 0.06
11. Turbine Control Valve Fast Closure, Valve Trip System Oil Pressure - Low	≤ 0.07#
12. Reactor Mode Switch Shutdown Position	NA
13. Manual Scram	NA

*Neutron detectors are exempt from response time testing. Response time shall be measured from the detector output or from the input of the first electronic component in the channel.

**Not including the simulated thermal power time constant specified in the COLR.

#Measured from start of turbine control valve fast closure.

TABLE 4.3.1.1-1

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION(a)	OPERATIONAL CONDITIONS IN WHICH SURVEILLANCE REQUIRED
1. Intermediate Range Monitors:				
a. Neutron Flux - High	S/U,S,(b) S	W W	R R	2 3, 4, 5
b. Inoperative	NA	W	NA	2, 3, 4, 5
2. Average Power Range Monitor:(f)				
a. Neutron Flux - High, Setdown	S/U,S,(b) S	W W	SA SA	2 3, 5
b. Flow Biased Simulated Thermal Power - High	S,D(h)	W	W(d)(e), SA(m), R(i)	1
c. Neutron Flux - High	S	W	W(d), SA	1
d. Inoperative	NA	W	NA	1, 2, 3, 5
3. Reactor Vessel Steam Dome Pressure - High	S	M	R(g)	1, 2(j)
4. Reactor Vessel Water Level - Low, Level 3	S	M	R(g)	1, 2
5. Reactor Vessel Water Level - High, Level 8	S	M	R(g)	1
6. Main Steam Line Isolation Valve - Closure	NA	M	R	1
7. Deleted				
8. Drywell Pressure - High	S	M	R(g)	1, 2(l)
9. Scram Discharge Volume Water Level - High				
a. Level Transmitter	S	M	R(g)	1, 2, 5(k)
b. Float Switches	NA	M	R	1, 2, 5(k)

TABLE 4.3.1.1-1 (Continued)

REACTOR PROTECTION SYSTEM INSTRUMENTATION SURVEILLANCE REQUIREMENTS

FUNCTIONAL UNIT	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION	OPERATIONAL CONDITIONS FOR WHICH SURVEILLANCE REQUIRED
10. Turbine Stop Valve - Closure	NA	M	R	1
11. Turbine Control Valve Fast Closure Valve Trip System Oil Pressure - Low	NA	M	R	1
12. Reactor Mode Switch Shutdown Position	NA	R	NA	1, 2, 3, 4, 5
13. Manual Scram	NA	M	NA	1, 2, 3, 4, 5

(a) Neutron detectors may be excluded from CHANNEL CALIBRATION.

(b) The IRM and SRM channels shall be determined to overlap for at least 1/2 decades during each startup after entering OPERATIONAL CONDITION 2 and the IRM and APRM channels shall be determined to overlap for at least 1/2 decades during each controlled shutdown, if not performed within the previous 7 days.

(c) Deleted

(d) This calibration shall consist of the adjustment of the APRM channel to conform to the power values calculated by a heat balance during OPERATIONAL CONDITION 1 when THERMAL POWER > 25% of RATED THERMAL POWER. Adjust the APRM channel if the absolute difference is greater than 2% of RATED THERMAL POWER. The provisions of Specification 4.0.4 are not applicable provided the surveillance is performed within 12 hours after reaching 25% of RATED THERMAL POWER.

(e) This calibration shall consist of the adjustment of the APRM flow biased channel to conform to a calibrated flow signal.

(f) The LPRMs shall be calibrated at least once per 1000 MWD/T using the TIP system.

(g) Calibrate trip unit setpoint at least once per 31 days.

(h) Verify measured core flow (total core flow) to be greater than or equal to established core flow at the existing loop flow (APRM % flow).

(i) This calibration shall consist of verifying that the simulated thermal power time constant is within the limits specified in the COLR.

(j) This function is not required to be OPERABLE when the reactor pressure vessel head is removed per Specification 3.10.1.

(k) With any control rod withdrawn. Not applicable to control rods removed per Specification 3.9.10.1 or 3.9.10.2.

(l) This function is not required to be OPERABLE when Drywell Integrity is not required.

(m) The CHANNEL CALIBRATION shall exclude the flow reference transmitters, these transmitters shall be calibrated at least once per 18 months.

TABLE 3.3.2-1

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	APPLICABLE OPERATIONAL CONDITION	ACTION
1. PRIMARY CONTAINMENT ISOLATION			
a. Reactor Vessel Water Level - Low, Level 2 (Division 1 & 2)	2	1, 2, 3 and #	20
b. Reactor Vessel Water Level - Low, Level 2 (Division 3)	4[d]	1, 2, 3 and #	28
c. Drywell Pressure - High (Division 1 & 2)	2	1, 2, 3	20
d. Drywell Pressure - High (Division 3)	4[d]	1, 2, 3	28
e. Containment and Drywell Purge Exhaust Plenum Radiation - High	2[b]	1, 2, 3 and *	21
f. Reactor Vessel Water Level - Low, Level 1	2	1, 2, 3 and #	20
g. Manual Initiation (Division 1 & 2)	2[c]	1, 2, 3 and *	22
h. Manual Initiation (Division 3)	1[e]	1, 2, 3 and *	28
2. MAIN STEAM ISOLATION			
a. Reactor Vessel Water Level - Low, Level 1	2	1, 2, 3	20
b. Main Steam Line Radiation - High	2[f]	***	29
c. Main Steam Line Pressure - Low	2	1	24
d. Main Steam Line Flow - High	2/line	1, 2, 3	23
e. Condenser Vacuum - Low	2	1, 2**, 3**	23
f. Main Steam Line Tunnel Temperature - High	2	1, 2, 3	23
g. Main Steam Line Tunnel Δ Temperature - High	2	1, 2, 3	23
h. Turbine Building Main Steam Line Temperature - High	2	1, 2, 3	23
i. Manual Initiation	2	1, 2, 3	22

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION

TRIP FUNCTION	MINIMUM OPERABLE CHANNELS PER TRIP SYSTEM (a)	APPLICABLE OPERATIONAL CONDITION	ACTION
3. SECONDARY CONTAINMENT ISOLATION			
a. Reactor Vessel Water Level - Low, Level 2	2	1, 2, 3 and #	25
b. Drywell Pressure - High	2	1, 2, 3	25
c. Manual Initiation	2	1, 2, 3	22
	2	*	25
4. REACTOR WATER CLEANUP SYSTEM ISOLATION			
a. Δ Flow - High	1	1, 2, 3	27
b. Δ Flow Timer	1	1, 2, 3	27
c. Equipment Area Temperature - High	1	1, 2, 3	27
d. Equipment Area Δ Temperature - High	1	1, 2, 3	27
e. Reactor Vessel Water Level - Low, Level 2	1	1, 2, 3	27
f. Main Steam Line Tunnel Ambient Temperature - High	2	1, 2, 3	27
g. Main Steam Line Tunnel Δ Temperature - High	1	1, 2, 3	27
h. SLCS Initiation	1	1, 2, 3	27
i. Manual Initiation	1	1, 2, 3	27
	2	1, 2, 3	26

ACTION

ACTION 20 - In OPERATIONAL CONDITION 1, 2 or 3, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours. In OPERATIONAL CONDITION #, suspend CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

ACTION 21 - Close the affected system isolation valve(s) within one hour or:

a. In OPERATIONAL CONDITION 1, 2 or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In Operational Condition *, suspend CORE ALTERATIONS, handling of irradiated fuel in the primary containment and operations with a potential for draining the reactor vessel.

ACTION 22 - Restore the manual initiation function to OPERABLE status within 48 hours or:

a. In OPERATIONAL CONDITION 1, 2, or 3, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. In OPERATIONAL CONDITION *, suspend CORE ALTERATIONS, operations with a potential for draining the reactor vessel, and handling of irradiated fuel in the primary containment.

ACTION 23 - Be in at least STARTUP with the associated isolation valves closed within 6 hours or be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

ACTION 24 - Be in at least STARTUP within 6 hours.

ACTION 25 - Verify SECONDARY CONTAINMENT INTEGRITY with the annulus exhaust gas treatment system operating within one hour.

ACTION 26 - Restore the manual initiation function to OPERABLE status within 8 hours or close the affected system isolation valves within 1 hour and declare the affected system inoperable.

ACTION 27 - Close the affected system isolation valves within one hour and declare the affected system inoperable.

ACTION 28 - Within one hour lock the affected system isolation valves closed, or verify, by remote indication, that the valve(s) is closed and electrically disarmed, or isolate the penetration(s), and declare the affected system inoperable.

ACTION 29 - Close the associated isolation valves within 6 hours or be in at least HOT SHUTDOWN within 12 hours.

NOTES

* 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 Condenser Low Vacuum Bypass Switch is in the normal position.

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

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

TABLE 3.3.2-1 (Continued)

ISOLATION ACTUATION INSTRUMENTATION
ACTION

NOTES (Continued)

(a) A channel may be placed in an inoperable status for up to 2 hours for required surveillance without placing the trip system in the tripped condition provided at least one other OPERABLE channel in the same trip system is monitoring that parameter.

(b) Containment and Drywell Purge System inboard and outboard isolation valves each use a separate two out of two isolation logic.

(c) There is only one (1) RCIC manual initiation channel for RCIC system containment isolation valves.

(d) Division 3 has only one trip system consisting of four channels logically combined in a one-out-of-two-twice configuration which only closes the HPCS Suppression Pool Test Return Valve (1E22-F023).

(e) Division 3 Manual Initiation consists of a single channel in a single trip system.

(f) This Trip Function no longer isolates the Main Steam Lines. The only isolation is of the mechanical pump lines (valves 1N62-F130A and B), using a single trip system consisting of two channels configured in a one-out-of-two logic.

TABLE 3.3.2-3

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
1. PRIMARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level - Low, Level 2	NA
b. Drywell Pressure - High	NA
c. Containment and Drywell Purge Exhaust Plenum Radiation - High[(b)]	$\leq 10^{(a)}$
d. Reactor Vessel Water Level - Low, Level 1	NA
e. Manual Initiation	NA
2. MAIN STEAM LINE ISOLATION	
a. Reactor Vessel Water Level - Low, Level 1	$\leq 1.0^{*}/\leq 10^{(a)**}$
b. Main Steam Line Radiation - High	NA
c. Main Steam Line Pressure - Low	$\leq 1.0^{*}/\leq 10^{(a)**}$
d. Main Steam Line Flow - High	$\leq 0.5^{*}/\leq 10^{(a)**}$
e. Condenser Vacuum - Low	NA
f. Main Steam Line Tunnel Temperature - High	NA
g. Main Steam Line Tunnel Δ Temperature - High	NA
h. Turbine Building Main Steam Line Temperature - High	NA
i. Manual Initiation	NA
3. SECONDARY CONTAINMENT ISOLATION	
a. Reactor Vessel Water Level - Low, Level 2	NA
b. Drywell Pressure - High	NA
c. Manual Initiation	NA
4. REACTOR WATER CLEANUP SYSTEM ISOLATION	
a. Δ Flow - High	NA
b. Δ Flow Timer	NA
c. Equipment Area Temperature - High	NA
d. Equipment Area Δ Temperature - High	NA
e. Reactor Vessel Water Level - Low, Level 2	NA
f. Main Steam Line Tunnel Ambient Temperature - High	NA
g. Main Steam Line Tunnel Δ Temperature - High	NA
h. SLCS Initiation	NA
i. Manual Initiation	NA

ISOLATION SYSTEM INSTRUMENTATION RESPONSE TIME

TRIP FUNCTION	RESPONSE TIME (Seconds)#
5. REACTOR CORE ISOLATION COOLING SYSTEM ISOLATION	
a. RCIC Steam Line Flow - High	NA
b. RCIC Steam Supply Pressure - Low	NA
c. RCIC Turbine Exhaust Diaphragm Pressure - High	NA
d. RCIC Equipment Room Ambient Temperature - High	NA
e. RCIC Equipment Room Δ Temperature - High	NA
f. Main Steam Line Tunnel Ambient Temperature - High	NA
g. Main Steam Line Tunnel Δ Temperature - High	NA
h. Main Steam Line Tunnel Temperature Timer	NA
i. RHR Equipment Room Ambient Temperature - High	NA
j. RHR Equipment Room Δ Temperature - High	NA
k. RCIC Steam Line Flow High Timer	NA
l. Drywell Pressure - High	NA
m. Manual Initiation	NA
6. RHR SYSTEM ISOLATION	
a. RHR Equipment Area Ambient Temperature - High	NA
b. RHR Equipment Area Δ Temperature - High	NA
c. RHR/RCIC Steam Line Flow - High	NA
d. Reactor Vessel Water Level - Low, Level 3	NA
e. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	NA
f. Drywell Pressure - High	NA
g. Manual Initiation	NA

(a) Isolation system instrumentation response time specified includes the diesel generator starting and sequence loading delays.

(b) Radiation detectors are exempt from response time testing. Response time shall be measured from detector output or the input of the first electronic component in the channel.

*Isolation system instrumentation response time for MSIVs only. No diesel generator delays assumed.

**Isolation system instrumentation response time for associated valves except MSIVs.

#Isolation system instrumentation response time specified for the Trip Function actuating each containment isolation valve shall be added to the isolation time for each valve to obtain ISOLATION SYSTEM RESPONSE TIME for each valve.

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	M	$R^{(b)}$	1, 2, 3 and #
b. Drywell Pressure - High	S	M	$R^{(b)}$	1, 2, 3
c. Containment and Drywell Purge Exhaust Plenum Radiation - High	S	M	R	1, 2, 3 and *
d. Reactor Vessel Water Level - Low, Level 1	S	M	$R^{(b)}$	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	M	$R^{(b)}$	1, 2, 3
b. Main Steam Line Radiation - High	S	M	R	***
c. Main Steam Line Pressure - Low	S	M	$R^{(b)}$	1
d. Main Steam Line Flow - High	S	M	$R^{(b)}$	1, 2, 3
e. Condenser Vacuum - Low	S	M	$R^{(b)}$	1, 2**, 3**
f. Main Steam Line Tunnel Temperature - High	S	M	R	1, 2, 3
g. Main Steam Line Tunnel Δ Temperature - High	S	M	R	1, 2, 3
h. Turbine Building Main Steam Line Temperature - High	S	M	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	M	R(b)	1, 2, 3 and #
b. Drywell Pressure - High	S	M	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	M	R	1, 2, 3
b. Δ Flow Timer	NA	M	R	1, 2, 3
c. Equipment Area Temperature - High	S	M	R	1, 2, 3
d. Equipment Area Ventilation Δ Temperature - High	S	M	R	1, 2, 3
e. Reactor Vessel Water Level - Low, Level 2	S	M	R(b)	1, 2, 3
f. Main Steam Line Tunnel Ambient Temperature - High	S	M	R	1, 2, 3
g. Main Steam Line Tunnel Δ Temperature - High	S	M	R	1, 2, 3
h. SLCS Initiation	NA	M(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	M	$R^{(b)}$	1, 2, 3
b. RCIC Steam Supply Pressure - Low	S	M	$R^{(b)}$	1, 2, 3
c. RCIC Turbine Exhaust Diaphragm Pressure - High	S	M	$R^{(b)}$	1, 2, 3
d. RCIC Equipment Room Ambient Temperature - High	S	M	R	1, 2, 3
e. RCIC Equipment Room Δ Temperature - High	S	M	R	1, 2, 3
f. Main Steam Line Tunnel Ambient Temperature - High	S	M	R	1, 2, 3
g. Main Steam Line Tunnel Δ Temperature - High	S	M	R	1, 2, 3
h. Main Steam Line Tunnel Temperature Timer	NA	M	R	1, 2, 3
i. RHR Equipment Room Ambient Temperature - High	S	M	R	1, 2, 3
j. RHR Equipment Room Δ Temperature - High	S	M	R	1, 2, 3
k. RCIC Steam Line Flow High Timer	NA	M	R	1, 2, 3
l. Drywell Pressure - High	S	M	$R^{(b)}$	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	M	R	1, 2, 3
b. RHR Equipment Area Δ Temperature - High	S	M	R	1, 2, 3
c. RHR/RCIC Steam Line Flow - High	S	M	R(b)	1, 2, 3
d. Reactor Vessel Water Level - Low, Level 3	S	M	R(b)	1, 2, 3
e. Reactor Vessel (RHR Cut-in Permissive) Pressure - High	S	M	R(b)	1, 2, 3
f. Drywell Pressure - High	S	M	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.

\# 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 31 days.

(b) Calibrate trip unit setpoint at least once per 31 days.

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

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 58 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 September 28, 1992, the Cleveland Electric Illuminating Company, et al. (the licensee), requested an amendment to Facility Operating License No. NPF-58, for the Perry Nuclear Power Plant (PNPP), Unit 1. The licensee proposed to eliminate the reactor protection system trip and the main steam line isolation valve (MSIV) closure requirements associated with the main steam line radiation monitors (MSLRM). This amendment proposed changes to Technical Specification 2.2.1, Reactor Protection System Instrumentation Setpoints, TS 3.3.1, Reactor Protection System Instrumentation, and TS 3.3.2, Isolation Actuation Instrumentation.

2.0 EVALUATION

On May 14, 1991, the NRC published a Safety Evaluation Report (SER) entitled "Acceptance for Referencing of Licensing Topical Report NEDO-31400, 'Safety Evaluation for Eliminating the Boiling Water Reactor Main Steam Line Isolation Valve Closure Function and Scram Function of the Main Steam Line Radiation Monitor (May 1987),'" in response to a Boiling Water Reactors Owners Group (BWROG) submittal of the topical report. The NRC staff has reviewed and accepted Topical Report NEDO-31400 on a generic basis provided the licensee satisfies the following three conditions:

1. The applicant demonstrates that the assumptions with regard to input values (including power per assembly, Chi/Q, and decay times) that are made in the generic analysis bound those for the plant.

2. The applicant includes sufficient evidence (implemented or proposed operating procedures, or equivalent commitments) to provide reasonable assurance that increased significant levels of radioactivity in the main steam lines will be controlled expeditiously to limit both occupational doses and environmental releases.

3. The applicant standardizes the MSLRM and offgas radiation monitor alarm setpoint at 1.5 times the nominal nitrogen-16 (N-16) background dose rate at the monitor locations. The applicant also commits to promptly sample the reactor coolant to determine possible contamination levels in the plant reactor coolant and the need for additional corrective actions, if the MSLRM or offgas radiation monitors or both exceed their alarm setpoints.

In response to the first condition, the licensee used the PNPP short term (accident) diffusion estimates provided in the Updated Safety Analysis Report (USAR), Table 2.3-24, to determine their Chi/Q value. The USAR Chi/Q value of 0.00043 seconds/cubic meter (s/m^3) for the PNPP control rod drop accident (CRDA) was slightly higher than the NEDO-31400 value of 0.0003 s/m^3. The licensee used the PNPP Chi/Q value in combination with PNPP design offgas system retention times and the other NEDO-31400 input parameters to calculate doses for the CRDA assuming that the main steam line isolation valves (MSIVs) do not close. The whole body dose at the PNPP Exclusion Area Boundary (EAB) based on the assessment was calculated to be 0.003 rem. This value is significantly lower than the guidelines established in the NRC Standard Review Plan (SRP) Section 15.4.9. Therefore, this deviation from the NEDO-31400 input values is acceptable.

The licensee also compared the PNPP source terms provided in USAR Table 15.4.13 to determine whether the values were consistent with the 18 source term inputs specified in NEDO-31400. Two of the source terms used in NEDO-31400 were smaller than the corresponding PNPP USAR source terms. The other 16 NEDO-31400 source terms were either the same or a larger value than the corresponding PNPP USAR source terms. The General Electric Company (GE) and PNPP reviewed all 18 source terms and determined that the overall NEDO source terms do bound the PNPP USAR source terms. Therefore, this deviation from the NEDO-31400 input values is acceptable.

The licensee also noted that the power/rod value used in the PNPP USAR is 0.122 megawatts/rod (Mw/rod) as compared to the value of 0.12 Mw/rod used in NEDO-31400. The value used in NEDO-31400 is equivalent to the PNPP value rounded off to two significant figures. This deviation does not invalidate the NEDO-31400 analysis because the only use of the power/rod value in NEDO-31400 was in the determination of the fission product source terms used in the radiological analyses which, as discussed above, bound the values used in the PNPP USAR analyses. Therefore, this deviation from the NEDO-31400 input values is acceptable.

In response to the second condition, the licensee reviewed the existing PNPP alarm response instructions (ARIs) and off-normal instructions (ONIs) to verify that proper instructions are provided for the operators to respond to high radiation levels detected by the MSLRM and the off-gas pretreatment radiation monitor. The existing instructions for the response to an alarm from the off-gas pretreatment radiation monitor are considered adequate. The existing instructions for the response to an alarm from the MSLRM will be revised to include a requirement to sample the reactor coolant and to check the off-gas pretreatment radiation monitor for trends in radiation levels.

In response to the third condition, the licensee has committed to adjust the MSLRM alarm setpoint to 1.5 times the nominal 100 percent background reading at the monitor locations. However, with respect to the alarm setpoint for the off-gas pretreatment radiation monitor, the licensee has provided a technical justification for taking an exception to setting the alarm at 1.5 times the N-16 background level. Because of the design of PNPP the N-16 levels at the off-gas pretreatment radiation monitor are extremely low. Other longer lived radioactive isotopes provide the majority of the background radiation levels detected by the monitor. Therefore, the N-16 background level is not an appropriate basis for setting the alarm for detecting fuel failures.

There are presently two alarms that are generated based on signals from the off-gas pretreatment radiation monitor; one alarm generates directly from the monitor and another generates from a recorder in the Control Room. The setpoint for the first alarm is addressed in the TS and is set at less than or equal to 0.358 Curies/second (Ci/sec). NEDO-31400 assumed that a change in the off-gas release rate of 1 to 10 Ci/sec would be promptly alarmed. Therefore, the existing TS alarm setpoint would satisfy the assumption in NEDO-31400. If the TS alarm setpoint is exceeded, the TS requires the operators to take actions to restore the release rate to within the limit or to shut the plant down.

The second alarm is typically set at 0.01 Ci/sec above the background dose rate or, if off-gas release rates exceed 0.075 Ci/sec, at 1.15 times the background release rate. This setpoint will cause annunciation of the alarm due to release rate increases much less than those assumed by the NEDO-31400 document.

In addition, experience at PNPP has shown that the off-gas pretreatment radiation monitor can accurately detect and trend very minor fuel damage. The licensee was able to detect leaks from two fuel rods in the first fuel cycle and from one bundle in the second cycle.

Therefore, because the proposed setpoints for the off-gas pretreatment radiation monitor will provide the operators with indications of fuel failures earlier than NEDO-31400 assumed, these setpoints are acceptable.

3.0 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.

4.0 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 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 this amendment involves no significant hazards consideration and there has been no public comment on such finding (58 FR 598). 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.

5.0 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: Andrew J. Kugler

Date: April 22, 1994