Docket File 50-440

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

September 21, 1994

Mr. Robert A. Stratman
Vice President Nuclear - Perry
Cleveland Electric Illuminating Company
P. O. Box 97, A200
Perry, OH 44081

SUBJECT: AMENDMENT NO. 66 TO FACILITY OPERATING LICENSE NO. NPF-58- PERRY NUCLEAR POWER PLANT, UNIT NO. 1 (TAC NO. M83050)

Dear Mr. Stratman:

The Commission has issued the enclosed Amendment No. 66 to Facility Operating License No. NPF-58 for the Perry Nuclear Power Plant (PNPP), Unit No. 1. This amendment consists of changes to the Technical Specifications (TS) in response to your application dated March 19, 1992.

This amendment incorporates the following ten types of clarifications or administrative changes:

(1) clarifies the applicability of TS 3.0.4 to three separate specifications,

(2) eliminates a potentially misleading clarification of core alteration,

(3) clarifies the operability requirements for reactor vessel level accident monitoring to address both types of monitors,

(4) adds Operational Condition 2 to the requirements for evaluating the effect on structural integrity of the reactor coolant system if chemistry is out of limits,

(5) corrects a typographical error associated with the feedwater leakage control system specification,

(6) clarifies the acceptable range for containment humidity,

(7) conservatively changes the setpoint for secondary containment vacuum requirement,

(8) clarifies what actions are required if both Division 1 and 2 diesel generators are inoperable and then one diesel generator is restored to operability,

(9) clarifies the minimum instrumentation required for monitoring operation of the inclined fuel transfer system

(10) elimination of reporting requirements that conflict with 10 CFR 50.4.

During preparation of this amendment several typographical and grammatical changes were incorporated. These changes are discussed in the Safety Evaluation.

DF01

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, 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. 66 to License No. NPF-58
2. Safety Evaluation

cc w/encl: See attached list

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OFFICE	LA:PDIII-3	APM:PDIII-3	PM:PDIII-3	OTSB	PD:PDIII-3	OGC
NAME	MRushbrook	LGundrum: JBH for	JHopkins*	CGrimes*	JHannon*	SChidakel*
DATE	9/12/94	9/12/94	8/18/94	8/29/94	8/18/94	9/8/94

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Mr. Robert A. Stratman
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Perry Nuclear Power Plant
Unit Nos. 1 and 2

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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. 66
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 March 19, 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. 66 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, Sr. 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: September 21, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 66

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 contains vertical lines indicating the area of change.

Remove	Insert
xiii	xiii
xv	xv
xx	xx
xxiv	xxiv
xxv	xxv
xxvi	xxvi
3/4 1-6	3/4 1-6
3/4 3-1	3/4 3-1
3/4 3-4	3/4 3-4
3/4 3-9	3/4 3-9
3/4 3-31	3/4 3-31
3/4 3-55	3/4 3-55
3/4 3-78	3/4 3-78
3/4 3-79	3/4 3-79
3/4 3-80	3/4 4-80
3/4 4-13	3/4 4-13
3/4 6-14	3/4 6-14
3/4 6-43	3/4 6-43
3/4 6-45	3/4 6-45
3/4 7-10	3/4 7-10
3/4 8-2	3/4 8-2
3/4 8-21	3/4 8-21
3/4 9-18	3/4 9-18
3/4 9-19	3/4 9-19
B 3/4 6-1	B 3/4 6-1
6-15	6-15
6-17	6-17
6-21	6-21
6-21a	6-21a

SECTION | PAGE

3.1.3.2 The maximum scram insertion time of each control rod from the fully withdrawn position, based on de-energization of the scram pilot valve solenoids as time zero, shall not exceed the following limits:

Reactor Vessel Dome Pressure (psig)*	Maximum Insertion Times to Notch Position (Seconds)		
	43	29	13
950	0.31	0.81	1.44
1050	0.32	0.86	1.57

APPLICABILITY: OPERATIONAL CONDITIONS 1 AND 2.

ACTION:

a. With the maximum scram insertion time of one or more control rods exceeding the maximum scram insertion time limits of Specification 3.1.3.2 as determined by Specification 4.1.3.2.a or b, operation may continue provided that:

1. For all "slow" control rods, i.e., those which exceed the limits of Specification 3.1.3.2, the individual scram insertion times do not exceed the following limits:

Reactor Vessel Dome Pressure (psig)*	Maximum Insertion Times to Notch Position (Seconds)		
	43	29	13
950	0.38	1.09	2.09
1050	0.39	1.14	2.22

Or the requirements of ACTION b are satisfied.

2. For "fast" control rods, i.e., those which satisfy the limits of Specification 3.1.3.2, the average scram insertion times do not exceed the following limits:

Reactor Vessel Dome Pressure (psig)*	Maximum Average Insertion Times to Notch Position (Seconds)		
	43	29	13
950	0.30	0.78	1.40
1050	0.31	0.84	1.53

3. The total number of "slow" control rods does not exceed 7.

4. No "slow" control rod or otherwise inoperable control rod occupies an adjacent location in any direction, including the diagonal, to another such control rod.

Otherwise, be in at least HOT SHUTDOWN within 12 hours.

*For intermediate reactor vessel dome pressure, the scram time criteria is determined by linear interpolation at each notch position.

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.

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 (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 switch 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.

3/4.3.2 ISOLATION ACTUATION INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.2 The isolation actuation instrumentation channels shown in Table 3.3.2-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.2-2 and with ISOLATION SYSTEM RESPONSE TIME as shown in Table 3.3.2-3.

APPLICABILITY: As shown in Table 3.3.2-1.

ACTION:

a. With an isolation actuation instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.2-2, declare the channel inoperable until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. 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 one hour.

c. 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 3.3.2-1.

*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 3.3.2-1 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.

ACTION

ACTION 30 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement:
a. With one channel inoperable, place the inoperable channel in the tripped condition within one hour or declare the associated ADS trip system or ECCS inoperable.
b. With more than one channel inoperable, declare the associated ADS trip system or ECCS inoperable.

ACTION 31 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, declare the associated ADS trip system or ECCS inoperable.

ACTION 32 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within one hour.

ACTION 33 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, restore the inoperable channel to OPERABLE status within 8 hours or declare the associated ADS trip system or ECCS inoperable.

ACTION 34 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel(s) in the tripped condition within one hour or declare the HPCS system inoperable.

ACTION 35 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour, or align the HPCS system to take suction from the suppression pool, or declare the HPCS system inoperable.

ACTION 36 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place at least one inoperable channel in the tripped condition within one hour or declare the HPCS system inoperable.

ACTION 37 - With the number of OPERABLE channels less than the Total Number of Channels, declare the associated emergency diesel generator inoperable and take the ACTION required by Specification 3.8.1.1 or 3.8.1.2, as appropriate.

ACTION 38 - With the number of OPERABLE channels less than the Total Number of Channels, place the inoperable channel in the tripped condition within 1 hour*; operation may then continue until performance of the next required CHANNEL FUNCTIONAL TEST.

ACTION 39 - With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, place the inoperable channel in the tripped condition within one hour. Restore the inoperable channel to OPERABLE status within 7 days or declare the associated system inoperable.

*The provisions of Specification 3.0.4 are not applicable.

3/4.3.6 CONTROL ROD BLOCK INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.3.6. The control rod block instrumentation channels shown in Table 3.3.6-1 shall be OPERABLE with their trip setpoints set consistent with the values shown in the Trip Setpoint column of Table 3.3.6-2.

APPLICABILITY: As shown in Table 3.3.6-1.

ACTION:

a. With a control rod block instrumentation channel trip setpoint less conservative than the value shown in the Allowable Values column of Table 3.3.6-2, declare the channel inoperable* until the channel is restored to OPERABLE status with its trip setpoint adjusted consistent with the Trip Setpoint value.

b. With the number of OPERABLE channels less than required by the Minimum OPERABLE Channels per Trip Function requirement, take the ACTION required by Table 3.3.6-1.

SURVEILLANCE REQUIREMENTS

4.3.6.1 Each of the above required control rod block trip systems and instrumentation channels 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 4.3.6-1.

4.3.6.2 The provisions of Specification 4.0.4 are not applicable to the CHANNEL FUNCTIONAL TEST and CHANNEL CALIBRATION surveillances for the Intermediate Range Monitors and Source Range Monitors for entry into their applicable OPERATIONAL CONDITIONS (as shown in Table 4.3.6-1) from OPERATIONAL CONDITION 1 provided the surveillances are performed within 12 hours after such entry.

*The APRM flow biased instrumentation need not be declared inoperable upon entering single recirculation loop operation provided the setpoints are adjusted within 8 hours per Specification 3.4.1.1.

TABLE 3.3.7.5-1

ACCIDENT MONITORING INSTRUMENTATION

	INSTRUMENT	REQUIRED NUMBER OF CHANNELS	MINIMUM CHANNELS OPERABLE	APPLICABLE OPERATIONAL CONDITIONS	ACTION
1.	Reactor Vessel Pressure	2	1	1,2,3	80
2.	Reactor Vessel Water Level				
	a. Fuel Zone	2	1	1,2,3	80
	b. Wide Range	2	1	1,2,3	80
3.	Suppression Pool Water Level	2	1	1,2,3	80
4.	Suppression Pool Water Temperature	16, 2/sector	8, 1/sector	1,2,3	80
5.	Primary Containment Pressure	2	1	1,2,3	80
6.	Primary Containment Air Temperature	2	1	1,2,3	80
7.	Drywell Pressure	2	1	1,2,3	80
8.	Drywell Air Temperature	2	1	1,2,3	80
9.	Primary Containment and Drywell Hydrogen Concentration Analyzer and Monitor	2	1	1,2 3	80
10.	Safety/Relief Valve Position Indicators**	2/valve	1/valve	1,2,3	80
11.	Primary Containment/Drywell Area Gross Gamma Radiation Monitors	2*	1*	1,2,3	81
12.	Offgas Ventilation Exhaust Monitor#	1	1	1,2,3	81
13.	Turbine Building/Heater Bay Ventilation Exhaust Monitor#	1	1	1,2,3	81
14.	Unit 1 Vent Monitor#	1	1	1,2,3	81
15.	Unit 2 Vent Monitor#	1	1	1,2,3	81
16.	Neutron Flux				
	a. Average Power Range	2	1	1,2,3	80
	b. Intermediate Range	2	1	1,2,3	80
	c. Source Range	2	1	1,2,3	80
17.	Primary Containment Isolation Valve Position***	2/valve	1/valve	1,2,3	82

*Each for primary containment and drywell.

**One channel consists of a pressure switch on the SRV discharge pipe, the other channel consists of a temperature sensor on the SRV discharge pipe.

***One channel consists of the open limit switch, and the other channel consists of the closed limit switch for each automatic containment isolation valve.

#High and intermediate range D19 system noble gas monitors.

Table 3.3.7.5-1 (Continued)

ACCIDENT MONITORING INSTRUMENTATIONS

ACTION STATEMENTS

ACTION 80 -

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Required Number of Channels shown in Table 3.3.7.5-1, restore the inoperable channel(s) to OPERABLE status within 7 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE requirements of Table 3.3.7.5-1, restore the inoperable channel(s) to OPERABLE status within 48 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. The provisions of Specification 3.0.4 are not applicable.

ACTION 81 - With the number of OPERABLE Channels less than required by the Minimum Channels OPERABLE requirement, either restore the inoperable Channel(s) to OPERABLE status within 72 hours, or:

a. Initiate the preplanned alternate method of monitoring the appropriate parameter(s), and

b. Prepare and submit a Special Report to the Commission pursuant to Specification 6.9.2 within 14 days following the event outlining the action taken, the cause of the inoperability and the plans and schedule for restoring the system to OPERABLE status.

ACTION 82 -

a. With the number of OPERABLE accident monitoring instrumentation channels less than the Minimum Channels OPERABLE requirements of Table 3.3.7.5-1, verify the valve(s) position by use of alternate indication methods; restore the inoperable channel(s) to OPERABLE status at the next time the valve is required to be demonstrated OPERABLE pursuant to Specification 4.0.5.

TABLE 4.3.7.5-1

ACCIDENT MONITORING INSTRUMENTATION SURVEILLANCE REQUIREMENTS

INSTRUMENT	CHANNEL CHECK	CHANNEL CALIBRATION	APPLICABLE OPERATIONAL CONDITIONS
1. Reactor Vessel Pressure	M	R	1, 2, 3
2. Reactor Vessel Water Level			
a. Fuel Zone	M	R	1, 2, 3
b. Wide Range	M	R	1, 2, 3
3. Suppression Pool Water Level	M	R	1, 2, 3
4. Suppression Pool Water Temperature	M	R	1, 2, 3
5. Primary Containment Pressure	M	R	1, 2, 3
6. Primary Containment Air Temperature	M	R	1, 2, 3
7. Drywell Pressure	M	R	1, 2, 3
8. Drywell Air Temperature	M	R	1, 2, 3
9. Primary Containment and Drywell Hydrogen Concentration Analyzer and Monitor	NA	Q*	1, 2, 3
10. Safety/Relief Valve Position Indicators	M	R	1, 2, 3
11. Primary Containment/Drywell Area Gross Gamma Radiation Monitors	M	R**	1, 2, 3
12. Offgas Ventilation Exhaust Monitor#	M	R	1, 2, 3
13. Turbine Building/Heater Bay Ventilation Exhaust Monitor#	M	R	1, 2, 3
14. Unit 1 Vent Monitor#	M	R	1, 2, 3
15. Unit 2 Vent Monitor#	M	R	1, 2, 3
16. Neutron Flux			
a. Average Power Range	M	R	1, 2, 3
b. Intermediate Range	M	R	1, 2, 3
c. Source Range	M	R	1, 2, 3
17. Primary Containment Isolation Valve Position	M	R	1, 2, 3

*Using sample gas containing:
a. One volume percent hydrogen, balance nitrogen.
b. Four volume percent hydrogen, balance nitrogen.

**The CHANNEL CALIBRATION shall consist of an electronic calibration of the channel, not including the detector, for range decades above 10 R/hr and a one point calibration check of the detector below 10 R/hr with an installed or portable gamma source.

#High and intermediate range D19 system noble gas monitors.

3/4.4.4 CHEMISTRY

LIMITING CONDITION FOR OPERATION

3.4.4 The chemistry of the reactor coolant system shall be maintained within the limits specified in Table 3.4.4-1.

APPLICABILITY: At all times.

ACTION:

a. In OPERATIONAL CONDITION 1:

1. With the conductivity, chloride concentration or pH exceeding the limit specified in Table 3.4.4-1 for less than 72 hours during one continuous time interval and, for conductivity and chloride concentration, for less than 336 hours per year, but with the conductivity less than 10 μmho/cm at 25°C and with the chloride concentration less than 0.5 ppm, this need not be reported to the Commission.

2. With the conductivity, chloride concentration or pH exceeding the limit specified in Table 3.4.4-1 for more than 72 hours during one continuous time interval or with the conductivity and chloride concentration exceeding the limit specified in Table 3.4.4-1 for more than 336 hours per year, be in at least STARTUP within the next 6 hours.

3. With the conductivity exceeding 10 μmho/cm at 25°C or chloride concentration exceeding 0.5 ppm, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b. In OPERATIONAL CONDITION 2 and 3, with the conductivity, chloride concentration or pH exceeding the limit specified in Table 3.4.4-1 for more than 48 hours during one continuous time interval, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. At all other times:

1. With the:

a) Conductivity or pH exceeding the limit specified in Table 3.4.4-1, restore the conductivity and pH to within the limit within 72 hours, or

b) Chloride concentration exceeding the limit specified in Table 3.4.4-1, restore the chloride concentration to within the limit within 24 hours, or

perform an engineering evaluation to determine the effects of the out-of-limit condition on the structural integrity of the reactor coolant system. Determine that the structural integrity of the reactor coolant system remains acceptable for continued operation prior to proceeding to OPERATIONAL CONDITION 2 or 3.

2. The provisions of Specification 3.0.3 are not applicable.

CONTAINMENT SYSTEMS

FEEDWATER LEAKAGE CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.9 Two independent feedwater leakage control (FWLC) system subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

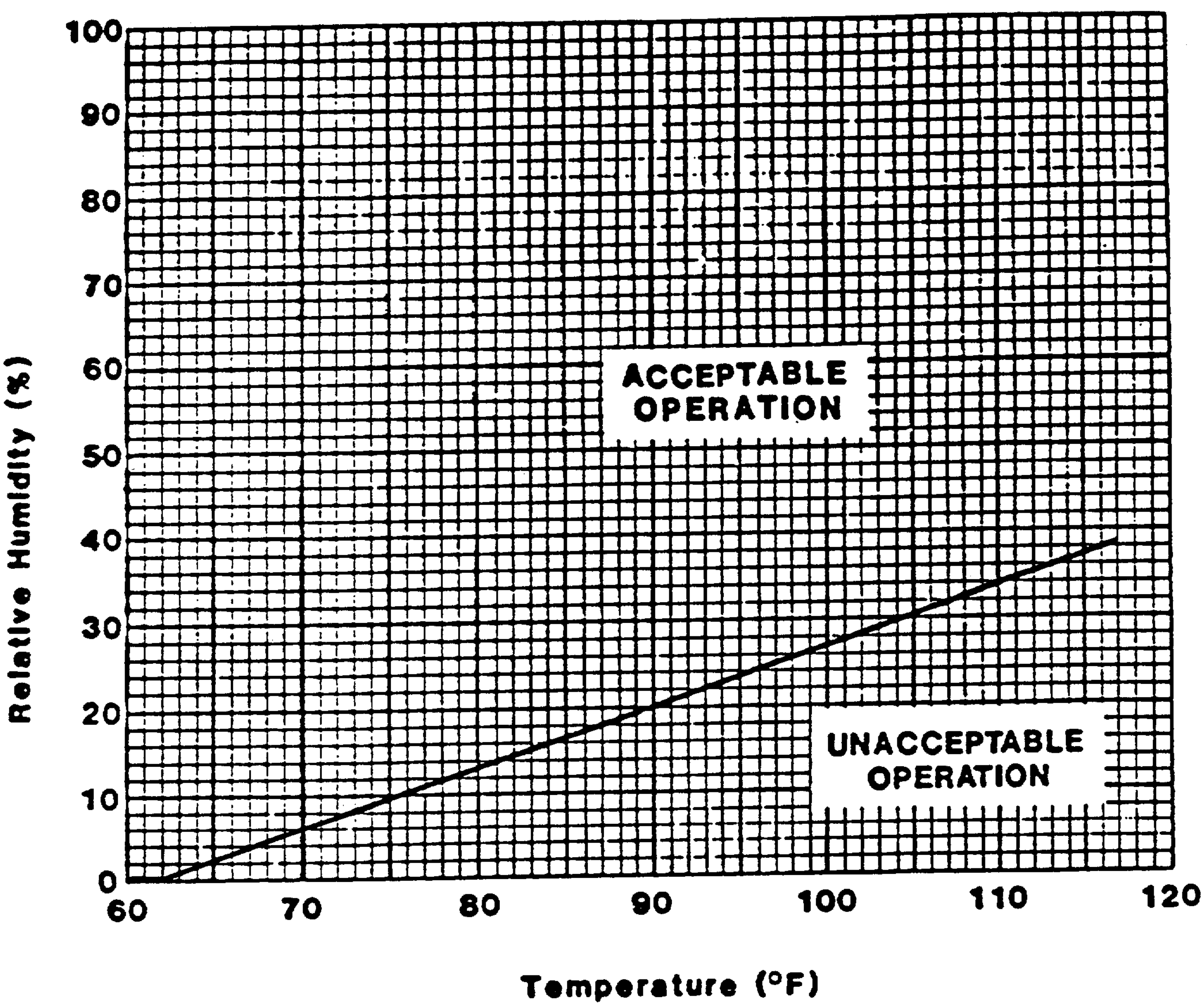
With one FWLC system subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 30 days or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.9 Each FWLC system subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by observing proper operation of the associated ECCS water leg pump.

b. At least once per 18 months by cycling each valve in the flow path not testable during POWER OPERATION through at least one complete cycle of full travel.



CONTAINMENT AVERAGE TEMPERATURE VS RELATIVE HUMIDITY

Figure 3.6.5.2-1

3/4.6.6 SECONDARY CONTAINMENT

SECONDARY CONTAINMENT INTEGRITY

LIMITING CONDITION FOR OPERATION

3.6.6.1 SECONDARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3 and *.

ACTION:

Without SECONDARY CONTAINMENT INTEGRITY:

a. In OPERATIONAL CONDITION 1, 2 or 3, restore SECONDARY CONTAINMENT INTEGRITY within 4 hours or 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 handling of irradiated fuel in the primary containment, CORE ALTERATIONS and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.6.6.1 SECONDARY CONTAINMENT INTEGRITY shall be demonstrated by:

a. Verifying at least once per 24 hours that the vacuum within the secondary containment is greater than or equal to 0.66 inches of vacuum water gauge.

b. Verifying at least once per 31 days that:

   1. The primary containment equipment hatch is closed and sealed and the shield blocks are installed adjacent to the shield building.

   2. The door in each access to the secondary containment is closed, except for routine entry and exit.

   3. All penetrations terminating in the annulus not capable of being closed by OPERABLE automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position.

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

e. Functional Tests

During the first refueling shutdown and at least once per 18 months thereafter during shutdown, a representative sample of snubbers shall be tested using one of the following sample plans for each type of snubber. The sample plan shall be selected prior to the test period and cannot be changed during the test period. The Nuclear Regulatory Commission shall be notified in writing pursuant to 10 CFR 50.4 of the sample plan selected prior to the test period or the sample plan used in the prior test period shall be implemented:

1) At least 10% of the total of each type of snubber shall be functionally tested either in-place or in a bench test. For each snubber of a type that does not meet the functional test acceptance criteria of Specification 4.7.4.f., an additional 5% of that type of snubber shall be functionally tested until no more failures are found or until all snubbers of that type have been functionally tested; or

2) A representative sample of each type of snubber shall be functionally tested in accordance with Figure 4.7.4-1. "C" is the total number of snubbers of a type found not meeting the acceptance requirements of Specification 4.7.4.f. The cumulative number of snubbers of a type tested is denoted by "N". At the end of each day's testing, the new values of "N" and "C" (previous day's total plus current day's increments) shall be plotted on Figure 4.7.4-1. If at any time the point plotted falls on or above the "Reject" line all snubbers of that type shall be functionally tested. If at any time the point plotted falls on or below the "Accept" line, testing of snubbers of that type may be terminated. When the point plotted lies in the "Continue Testing" region, additional snubbers of that type shall be tested until the point falls in the "Accept" region or the "Reject" region, or all the snubbers of that type have been tested. Testing equipment failure during functional testing may invalidate that day's testing and allow that day's testing to resume anew at a later time, providing all snubbers tested with the failed equipment during the day of equipment failure are retested; or

3) An initial representative sample of 55 snubbers of each type shall be functionally tested. For each snubber type which does not meet the functional test acceptance criteria, another sample of at least one-half the size of the initial sample shall be tested until the total number tested is equal to the initial sample size multiplied by the factor, $1 + C/2$, where "C" is the number of snubbers found which do not meet the functional test acceptance criteria. The results from this sample plan shall be plotted using an "Accept" line which follows the equation $N = 55(1 + C/2)$. Each snubber point should be plotted as soon as the snubber is tested. If the point plotted falls on or below the "Accept" line, testing of that type of snubber may be terminated. If the point plotted fails above the "Accept" line, testing must continue until the point falls on or below the "Accept" line or all the snubbers of that type have been tested.

ACTION (Continued)

restore the diesel generator to OPERABLE status within 72 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With one offsite circuit of the above required A.C. sources and diesel generator Div 1 or Div 2 of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the remaining A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If a diesel generator became inoperable due to any cause other than preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE diesel generators separately for each diesel generator by performing Surveillance Requirements 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 within 8 hours* for each diesel generator which has not been successfully tested within the past 24 hours. Restore at least one of the inoperable A.C. sources to OPERABLE status within 12 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. Restore at least two offsite circuits and diesel generators Div 1 and Div 2 to OPERABLE status within 72 hours from time of initial loss or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With diesel generator Div 3 of the above required A.C. electrical power sources inoperable, demonstrate the OPERABILITY of the offsite A.C. sources by performing Surveillance Requirement 4.8.1.1.1.a within 1 hour and at least once per 8 hours thereafter. If the diesel generator became inoperable due to any cause other than preplanned preventive maintenance or testing, demonstrate the OPERABILITY of the remaining OPERABLE diesel generators separately by performing Surveillance Requirements 4.8.1.1.2.a.4 and 4.8.1.1.2.a.5 within 24 hours*. Restore diesel generator Div 3 to OPERABLE status within 72 hours or declare the HPCS system and the C ESW pump inoperable and take the ACTION required by Specifications 3.5.1. and 3.7.1.1.

e. With diesel generator Div 1 or Div 2 of the above required A.C. electrical power sources inoperable, in addition to ACTION b, c, or g**, as applicable, verify within 2 hours that all required systems, subsystems, trains, components and devices that depend on the remaining OPERABLE diesel generator as a source of emergency power are also OPERABLE; otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

*This test is required to be completed regardless of when the inoperable diesel generator is restored to OPERABILITY. The provisions of Specification 3.0.2 are not applicable.

**When either the Div 1 or Div 2 diesel is restored to OPERABILITY.

CONTAINMENT PENETRATION CONDUCTOR OVERCURRENT PROTECTIVE DEVICES

LIMITING CONDITION FOR OPERATION

3.9.3.1 Primary and backup containment penetration conductor overcurrent protective devices associated with each containment electrical penetration circuit shall be OPERABLE. The scope of these protective devices excludes those circuits for which credible fault currents would not exceed the electrical penetration design rating.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one or more of the primary and backup containment penetration conductor overcurrent protective devices inoperable, declare the affected system or component inoperable and apply the appropriate ACTION statement for the affected system and:

   1. For 13.8 kV circuit breakers, de-energize the 13.8 kV circuit(s) by tripping the associated redundant circuit breaker(s) within 72 hours and verify the redundant circuit breaker to be tripped at least once per 7 days thereafter.

   2. For 120-volt circuit breakers, remove the inoperable circuit breaker(s) from service by racking out* the breaker within 72 hours and verify the inoperable breaker(s) to be racked out* at least once per 7 days thereafter.

   Otherwise, be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. The provisions of Specification 3.0.4 are not applicable to overcurrent devices in 13.8 kV circuits which have their redundant circuit breakers tripped or to 120-volt circuits which have the inoperable circuit breaker racked out.*

SURVEILLANCE REQUIREMENTS

4.8.4.1 Each of the primary and backup containment penetration conductor overcurrent protective devices shall be demonstrated OPERABLE:

a. At least once per 18 months:

   1. By verifying that the medium voltage 13.8 kV circuit breakers are OPERABLE by selecting, on a rotating basis, at least 10% of the circuit breakers and performing:

      a) A CHANNEL CALIBRATION of the associated protective relays,
      b) An integrated system functional test which includes simulated automatic actuation of the system and verifying that each relay and associated circuit breakers and overcurrent control circuits function as designed, and

*Racking out may be accomplished by tripping the breaker under administrative control.

3/4.9.12 INCLINED FUEL TRANSFER SYSTEM

LIMITING CONDITION FOR OPERATION

3.9.12 The inclined fuel transfer system (IFTS) may be in operation provided that:

a. The access door and floor plugs of all rooms through which the transfer system penetrates are closed and locked.

b. All access interlocks and palm switches are OPERABLE.

c. The Versa blocking valve located in the Fuel Handling Building IFTS hydraulic power unit is OPERABLE.

d. At least one IFTS carriage position indicator is OPERABLE at each carriage position and at least one liquid level sensor is OPERABLE at each liquid level monitoring position.

e. All keylock switches which provide IFTS access control-transfer system lockout ar OPERABLE.

f. The warning light outside of the access door is OPERABLE.

APPLICABILITY: When the IFTS blank flange is removed.

ACTION:

a. With one or more access interlocks, warning lights, and/or palm switches inoperable, operation of the IFTS may continue provided that entry into the area is prohibited by establishing a continuous watch and conspicuously posting as a high radiation area.

b. With the requirements of the above specification not satisfied, suspend IFTS operation with the IFTS at either terminal point. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.12.1 Within 4 hours prior to the startup of the IFTS, verify that no personnel are in areas immediately adjacent to the IFTS tube and that the access door and floor plugs to rooms through which the IFTS tube penetrates are closed and locked.

SURVEILLANCE REQUIREMENTS (Continued)

4.9.12.2 Within 4 hours prior to the operation of IFTS and at least once per 12 hours thereafter, when the IFTS is in operation verify that:

a. At least one IFTS carriage position indicator is OPERABLE at each carriage position and at least one liquid level sensor is OPERABLE at each liquid level monitoring position.

b. The warning light outside of the access door is OPERABLE.

4.9.12.3 Within 4 hours prior to the operation of IFTS and at least once per 7 days thereafter, when the IFTS is in operation verify that:

a. All access interlocks for the IFTS Valve Room are OPERABLE.

b. The Versa blocking valve in the Fuel Handling Building IFTS hydraulic power unit is OPERABLE.

c. All keylock switches which provide IFTS access control-transfer system lockout are OPERABLE.

4.9.12.4 Within 4 hours prior to installation of the floor plugs, after they have been removed, verify that the access interlocks and palm switches for the shield building annulus room and/or mid-support room, as applicable, are OPERABLE.

3/4.6.1 CONTAINMENT

3/4.6.1.1 PRIMARY CONTAINMENT INTEGRITY

PRIMARY CONTAINMENT INTEGRITY ensures that the release of radioactive materials from the containment atmosphere will be restricted to those leakage paths and associated leak rates assumed in the accident analyses. This restriction, in conjunction with the leakage rate limitation, will limit the site boundary radiation doses to within the limits of 10 CFR Part 100 during accident conditions.

During shutdown when irradiated fuel is being handled in the primary containment, and during CORE ALTERATIONS and operations with a potential for draining the reactor vessel, the # footnote permits the opening of six vent and drain pathways for the purpose of performing containment isolation valve leak rate surveillance testing provided the reactor has been subcritical for at least seven days. Offsite doses were calculated assuming the postulated fuel handling accident inside primary containment after a seven day decay time, and assuming all the airborne activity existing inside containment after a seven day decay time, and assuming all the airborne activity existing inside containment after the accident is immediately discharged directly to the environment (i.e., no containment). Although this analysis would indicate that no restriction on the number of vent and drain pathways was required, the number of open pathways was restricted to six for conservatism.

3/4.6.1.2 PRIMARY CONTAINMENT LEAKAGE

The limitations on containment leakage rates ensure that the total containment leakage volume will not exceed the value assumed in the accident analyses at the peak accident pressure of 7.80 psig, P_a. As an added conservatism, the measured overall integrated leakage rate is further limited to less than or equal to 0.75 L_a during performance of the periodic tests to account for possible degradation of the containment leakage barriers between leakage tests.

Overall integrated leakage rate means the leakage rate which obtains from a summation of leakage through all potential leakage paths. Where a leakage path contains more than one valve, fitting, or component in series, the leakage for that path will be that leakage of the worst leaking valve, fitting, or component and not the summation of the leakage of all valves, fittings, or components in that leakage path.

Operating experience with the main steam line isolation valves has indicated that degradation has occasionally occurred in the leak tightness of the valves; therefore the special requirement for testing these valves.

ACTIVITIES (Continued)

f. The Plant Security Plan and Emergency Plan, and implementing instructions, shall be reviewed at least once per 12 months. Recommended changes to the Plans and implementing instructions shall be reviewed pursuant to the requirements of Specification 6.5.1.6 and approved by the Plant Manager. NRC approval shall be obtained as appropriate.

6.6 REPORTABLE EVENT ACTION

6.6.1 The following actions shall be taken for REPORTABLE EVENTS:

a. The Commission shall be notified and a report submitted pursuant to the requirements of Section 50.73 to 10 CFR Part 50, and

b. Each REPORTABLE EVENT shall be reviewed by the PORC and the results of the review submitted to the NSRC and the Vice President - Nuclear.

6.7 SAFETY LIMIT VIOLATION

6.7.1 The following actions shall be taken in the event a Safety Limit is violated:

a. The NRC Operations Center shall be notified by telephone as soon as possible and in all cases within 1 hour. The Vice President - Nuclear and the NSRC shall be notified within 24 hours.

b. A Safety Limit Violation Report shall be prepared. The report shall be reviewed by the PORC. This report shall describe (1) applicable circumstances preceding the violation, (2) effects of the violation upon unit components, systems, or structures, and (3) corrective action taken to prevent recurrence.

c. The Safety Limit Violation Report shall be submitted to the Nuclear Regulatory Commission pursuant to 10 CFR 50.4, the NSRC, and the Vice President - Nuclear within 30 days of the violation.

d. Critical operation of the unit shall not be resumed until authorized by the Commission.

6.8 PROCEDURES/INSTRUCTIONS AND PROGRAMS

6.8.1 Written procedures/instructions shall be established, implemented, and maintained covering the activities referenced below:

a. The applicable procedures recommended in Appendix A of Regulatory Guide 1.33, Revision 2, February 1978.

6.8 PROCEDURES/INSTRUCTIONS AND PROGRAMS (Continued)

b. In-Plant Radiation Monitoring

A program which will ensure the capability to accurately determine the airborne iodine concentration in vital areas under accident conditions. This program shall include the following:

1. Training of personnel,
2. Procedures for monitoring, and
3. Provisions for maintenance of sampling and analysis equipment.

c. Post-accident Sampling

A program which will ensure the capability to obtain and analyze reactor coolant, radioactive iodines and particulates in plant gaseous effluents, and containment atmosphere samples under accident conditions. The program shall include the following:

1. Training of personnel,
2. Procedures for sampling and analysis, and
3. Provisions for maintenance of sampling and analysis equipment.

6.9 REPORTING REQUIREMENTS

ROUTINE REPORTS

6.9.1 In addition to the applicable reporting requirements of Title 10, Code of Federal Regulations, the following reports shall be submitted to the Nuclear Regulatory Commission pursuant to 10 CFR 50.4 unless otherwise noted.

STARTUP REPORT

6.9.1.1 A summary report of plant startup and power escalation testing shall be submitted following (1) receipt of an Operating License, (2) amendment to the license involving a planned increase in power level, (3) installation of fuel that has a different design or has been manufactured by a different fuel supplier, and (4) modifications that may have significantly altered the nuclear, thermal, or hydraulic performance of the unit.

6.9.1.2 The startup report shall address each of the tests identified in the Final Safety Analysis Report Subsection 14.2.12.2 and shall include a description of the measured values of the operating conditions or characteristics obtained during the test program and a comparison of these values with design predictions and specifications. Any corrective actions that were required to obtain satisfactory operation shall also be described. Any additional specific details required in license conditions based on other commitments shall be included in this report.

6.9.1.3 Startup reports shall be submitted within (1) 90 days following completion of the startup test program, (2) 90 days following resumption or commencement of commercial power operation, or (3) 9 months following initial criticality, whichever is earliest. If the startup report does not cover all three events, i.e., initial criticality, completion of startup test program, and resumption or commencement of commercial operation supplementary reports shall be submitted at least every 3 months until all three events have been completed.

ANNUAL RADIOACTIVE EFFLUENT RELEASE REPORT (continued)

The Annual Radioactive Effluent Release Report shall include any changes made during the reporting period to the PROCESS CONTROL PROGRAM (PCP) and to the OFFSITE DOSE CALCULATION MANUAL (ODCM), pursuant to Specifications 6.13 and 6.14, respectively, as well as any major change to Liquid, Gaseous, or Solid Radwaste Treatment Systems pursuant to Specification 6.15. It shall also include a listing of new locations for dose calculations and/or environmental monitoring identified by the Land Use Census pursuant to Specification 3.12.2.

The Annual Radioactive Effluent Release Report shall also include the following: an explanation as to why the inoperability of liquid or gaseous effluent monitoring instrumentation was not corrected within the time specified in Specification 3.3.7.9 or 3.3.7.10, respectively; and description of the events leading to liquid holdup tanks exceeding the limits of Specification 3.11.1.4.

MONTHLY OPERATING REPORTS

6.9.1.8 Routine reports of operating statistics and shutdown experience shall be submitted to the Nuclear Regulatory Commission pursuant to 10 CFR 50.4 on a monthly basis, with a copy to the Director, Office of Resource Management, U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, no later than the 15th of each month following the calendar month covered by the report.

CORE OPERATING LIMITS REPORT

6.9.1.9 Core operating limits shall be established and documented in the CORE OPERATING LIMITS REPORT before each reload cycle or any remaining part of a reload cycle for the following:

(1) The Average Planar Linear Heat Generation Rate (APLHGR) for Technical Specification 3.2.1.

(2) The Minimum Critical Power Ratio (MCPR) for Technical Specification 3.2.2.

(3) The Linear Heat Generation Rate (LHGR) for Technical Specification 3.2.3.

(4) The Simulated Thermal Power Time Constant for Technical Specification 3.3.1.

The analytical methods used to determine the core operating limits shall be those previously reviewed and approved by NRC in NEDE-24011-P-A, General Electric Standard Application for Reactor Fuel. (The approved revision at the time reload analyses are performed shall be identified in the COLR.)

The core operating limits shall be determined so that all applicable limits (e.g., fuel thermal-mechanical limits, core thermal-hydraulic limits, ECCS limits, nuclear limits such as SHUTDOWN MARGIN, and transient and accident analysis limits) of the safety analysis are met.

The CORE OPERATING LIMITS REPORT, including any mid-cycle revisions or supplements thereto, shall be submitted upon issuance for each reload cycle, to the Nuclear Regulatory Commission pursuant to 10 CFR 50.4.

SPECIAL REPORTS

6.9.2 Special reports shall be submitted to the Nuclear Regulatory Commission pursuant to 10 CFR 50.4 within the time period specified for each report.

6.9.3 Safety-relief valve failures will be reported to the Nuclear Regulatory Commission pursuant to 10 CFR 50.4 and 10 CFR 50.73 within 30 days.

6.9.4 Violations of the requirements of the fire protection program described in the Final Safety Analysis Report which would have adversely affected the ability to achieve and maintain safe shutdown in the event of a fire shall be reported to the Nuclear Regulatory Commission pursuant to 10 CFR 50.4 and 10 CFR 50.73 within 30 days.

6.10 RECORD RETENTION

6.10.1 In addition to the applicable record retention requirements of Title 10, Code of Federal Regulations, the following records shall be retained for at least the minimum period indicated.

6.10.2 The following records shall be retained for at least 5 years:

a. Records and logs of unit operation covering the interval at each power level.

b. Records and logs of principal maintenance activities, inspections, repair, and replacement of principal items of equipment related to nuclear safety.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 66 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 March 19, 1992, the Cleveland Electric Illuminating Company, et al. (licensees), proposed changes to the Technical Specifications (TSs) for the Perry Nuclear Power Plant, Unit No. 1. This amendment includes 10 administrative changes. The changes include: clarifying the applicability of TS 3.0.4, eliminating potentially misleading clarification of core alteration, clarifying the operability requirements for reactor vessel level accident monitoring, adding Operational Condition 2 to the requirements for evaluating the effect on structural integrity of the reactor coolant system, if chemistry is out of limits, correcting a typographical error associated with the feedwater leakage control system, clarifying the acceptable range for containment humidity, conservatively changing the setpoint for secondary containment vacuum, clarifying what actions are required when 1 of 2 inoperable diesel generators has been restored to operability, clarifying the required number and location of the proximity sensors and level monitoring instrumentation associated with the inclined fuel transfer system (IFTS), and eliminating reporting requirements that conflict with 10 CFR 50.4. These ten administrative changes will ensure consistency of definitions, actions, and setpoints between various TS sections, the Updated Safety Analysis Report (USAR) and 10 CFR Part 50. In addition to the ten changes requested by the licensee, additional pages were changed, as described in section 2.11 to correct identified errors.

2.0 EVALUATION

Each change will be discussed separately.

2.1 APPLICABILITY OF TS 3.0.4

For TS 3.3.1 ACTION a. and TS 3.3.2 ACTION b. the proposed change is the deletion of statement that TS 3.0.4 does not apply. As a consequence of Generic Letter 87-09, Amendment 30 deleted the requirement to specify in applicable TS when TS 3.0.4 did not apply. However, the two deletions requested for TS 3.3.1 and TS 3.3.2 were not originally submitted with the other changes requested for Amendment 30. TS 3.3.1 Action a. gives specific directions to place inoperable channels of the reactor protection system in one trip system in-trip unless placing the channel in-trip would cause the Trip Function to occur. Similarly TS 3.3.2 ACTION b. gives specific directions to place inoperable channel(s) of one isolation actuation trip system in a tripped condition within one hour unless placing the channel(s) in a tripped condition would cause the Trip Function to occur. Both of these TS are footnoted to specify restoring a channel to operability in two hours or take the actions associated with loss of a trip function. Therefore, the change deletes the statement TS 3.0.4 does not apply. As discussed in GL 87-09, TS 3.0.4 does not apply if the TS permits continued operation for an unlimited period of time, or allows entry into an operational mode or other specified condition of operation. Therefore, these deletions are in accordance with the guidance of GL 87-09.

Another change requested restores the statement, "The provisions of Specification 3.0.4 are not applicable." to Action 38 associated with Table 3.3.3-1, "Emergency Core Cooling System Action Instrumentation Action." This requirement was deleted with Amendment 30. The basis for restoring the statement is that Action statement 38 does not allow an unlimited period of time, but states that one of two channels can be placed in-trip until the next channel functional test. The surveillance requirements for channel functional tests is monthly. However, the purpose of GL 87-09 was not to restrict mode changes if the action requirements provide an acceptable level of safety for continued operation. Therefore, the staff agrees with restoring the exception. The staff notes that the improved Standard Technical Specifications (STS), when approved for Perry, will eliminate the need for the new TS 3.0.4 statement.

2.2 <u>CORE ALTERATION DEFINITION CLARIFICATION</u>

The TS amendment requests deletion of the asterisk and associated footnote for ACTION 3 and ACTION 9 in Table 3.3.1-1, "Reactor Protection System Instrumentation." The footnote explains that the replacement of local power range monitor (LPRM) strings need not be suspended when all operations involving CORE ALTERATIONS are suspended. This footnote leads to confusion since the definition of CORE ALTERATIONS specifically excludes normal movement of LPRMs. Removing this footnote is consistent with TS 3.9.2 which does not specify any additional restrictions on the source range monitors during replacement of LPRM strings. This is strictly an administrative change.

2.3 <u>REACTOR VESSEL WATER LEVEL INSTRUMENTATION</u>

The change requests additional information be added for the BWR Accident Monitoring reactor coolant level instrumentation (Table 3.3.7.5-1) to include both wide range and fuel zone instrumentation operability and surveillance requirements. This change is for clarification and does not change the intent or requirements for the reactor vessel level instrumentation.

2.4 <u>ADDITIONAL OPERATIONAL CONDITION ADDED TO CHEMISTRY REQUIREMENTS</u>

This change is to provide clarification to Specification 3.4.4., ACTION c. the intent of the last sentence of ACTION c. is to ensure that, in the event an engineering evaluation is relied on to justify continued plant startup from Operational Condition 4 or 5, with an out-of-limit conductivity, pH or chloride concentration, that the engineering evaluation is completed and the effects on the structural integrity of the reactor coolant system are determined acceptable for continued operation prior to entering a higher mode of operation. Under the existing wording of ACTION c., a mode change to Operational Condition 3 is not allowed until the engineering evaluation is performed and an acceptable determination obtained. However, the ACTION statement fails to extend this requirement to mode changes to Operational Condition 2. It is typical for a BWR, during the performance of a plant startup, to move from Operational Condition 4 to Operational Condition 2, without entering Operational Condition 3 at any time. Therefore, the proposed change to ACTION c. will make it clear that mode changes into either Operational Condition 2 or 3 are not allowed, until it is determined that the structural integrity of the reactor coolant system remains acceptable for continued operation.

2.5 CORRECT DISCREPANCY IN ACTIONS FOR FEEDWATER LEAKAGE CONTROL SYSTEM

The proposed change in the ACTION statement for Specification 3.6.1.9 will require restoration of an inoperable Feedwater Leakage Control System within 30 days or require being in HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours. Currently the action statement requires restoration within 30 days or be in HOT SHUTDOWN within the next 12 hours or in COLD SHUTDOWN within the following 24 hours. The revised wording requested by the licensee replaces "or in COLD SHUTDOWN..." with "and in COLD SHUTDOWN..." is consistent with the original intent of this specification and with that of other standard TS ACTION statements.

2.6 CONTAINMENT AVERAGE TEMPERATURE AND RELATIVE HUMIDITY RELATIONSHIP

The requested change will extend the line on Figure 3.6.5.2-1, which divides the regions of acceptable versus unacceptable operation for containment relative humidity as it relates to the containment average temperature. Currently, the Figure does not provide any criteria for acceptable versus unacceptable when the average containment air temperature is below 74 °F. Since the containment temperature can go below 74 °F, the extension of the line to 0% relative humidity at or below 62 °F eliminates concern about the need to maintain humidity levels.

The intent of the figure is to show initial relative humidities at various containment temperatures which are acceptable in order to maintain peak vacuum inside containment $\leq$ 0.72 psi (design is $\leq$ 0.80 psi) following initiation of both containment spray loops. Additional engineering evaluations indicate that extending the line is conservative.

The staff notes that the submittal to revise TS to the improved STS, PY-CEI/NRR-1732 L, dated December 16, 1993, requests deleting the figure and including the information in the TS bases.

2.7 INCREASE VACUUM REQUIREMENTS FOR SECONDARY CONTAINMENT

PNPP's design bases and safety analysis require the secondary containment (annulus area) to be maintained at a minimum negative pressure of 0.25 inches water gauge at all times. The licensee requests the value of vacuum in TS 4.6.6.1 be increased to 0.66. As a result of NRC Information Notice 88-76, "Recent Discovery Of a Phenomenon Not Previously Considered In The Design of Secondary Containment Pressure Control," engineering calculations were revised to address the differences between outside air and annulus air and sensor locations being 170 feet below the top of secondary containment. Based on the new calculations, the analytical setpoint required to maintain the minimum negative pressure of 0.25 inch vacuum water gauge was determined to be 0.66 inches water gauge post LOCA and 0.50 inches water gauge during normal minimum design environmental conditions. Calculations to establish new field setpoints for delta-P and for secondary containment air flow values based on the new analytical setpoint were completed and the new field setpoints were incorporated into procedures and the Safety Analysis Report (SAR) was revised. To ensure consistency between the revised analysis, the SAR, and the TS, the values included in the TS will be changed to the more conservative values.

2.8 CLARIFICATION OF ACTIONS FOR RESTORING DIVISION 1 AND DIVISION 2 DIESEL GENERATORS

The change to TS 3.8.1.1 clarifies the actions to take if one diesel generator is declared OPERABLE after both Division 1 and Division 2 diesel generators have been inoperable. ACTION g. contains the requirements and time constraints for returning first one, and then both of the Division 1 and 2 diesel generators. The change clarifies that ACTION e. shall be performed after the first diesel generator is restored. ACTION e. requires that within 2 hours a verification be made for all systems, subsystems, trains, components, and devices that depend on the restored diesel generator as a source of emergency power. This TS change is for clarification and neither adds or removes any requirements.

2.9 IFTS SPECIFICATION AND SURVEILLANCE REQUIREMENTS

The change requested clarifies the LIMITING CONDITION FOR OPERATION (LCO) and Surveillance Requirements for the IFTS proximity sensors and liquid level sensors. The IFTS has 2 proximity sensors at 12 separate carriage positions. The change will ensure that at least 1 of the 2 proximity sensors at each of the 12 carriage positions shall be OPERABLE. Similarly, the clarification will ensure that at least 1 of the 2 liquid (water) level sensors will be OPERABLE at each of the two monitoring locations or positions (the "Tube Full" and the "Tube Empty" positions).

2.10 CONSISTENCY OF TS SECTION 6.9 AND 10 CFR 50.4 REQUIREMENTS

The changes requested remove inconsistencies between the administrative reporting requirements (where to send reports and who to send the reports to) included in the TS and the requirements of 10 CFR 50.4. When 10 CFR 50.4 was revised, it specifically stated that its requirements took precedence over any existing TS requirement. This is an administrative change only.

2.11 MISCELLANEOUS CHANGES

Table 3.6.4-1, deleted in Amendment 44, was deleted from Index page xiii. Table 3.8.4.1-1, deleted in Amendment 44, was deleted from Index page xv. Index page xx was corrected for TS 3/4.6.1 page numbers. The second listing for Figure 5.1.1-1 was removed on Index page xxiv. Entries for TS 6.2.1 and 6.2.3 on page xxv were changed to agree with the TS and page entries corrected. Entry for TS 6.9.2 on page xxvi was corrected. The title "CONTROL ROD SCRAM MAXIMUM INSERTION TIMES," was corrected to read "CONTROL ROD MAXIMUM SCRAM INSERTION TIMES" on page 3/4 1-6. On page 3/4 8-2 in Action c. "with" changed to "within" in last sentence. On page 3/4 9-19 in TS 4.9.12.2 "in in" was changed to "is in" in the second line. Page B 3/4 6-1 was changed to insert the word "primary" before "containment leakage" to agree with the TS title. The footnote on TS page 3/4 8-21 was changed from "**" to a single "*." Page B 3/4 6-1 was changed to correct the title of 3/4.6.1.2 in the BASES section to "Primary Containment Leakage" to agree with the TS. All of these changes are made for clarity and accuracy.

2.12 RESULTS OF REVIEW

The NRC staff has reviewed the ten proposed changes to the TS and based on the evaluations performed, the NRC staff finds the proposed changes to be 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 changes 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 changes 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 (57 FR 30260). The amendment also changes a reporting or recordkeeping requirement. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9) and (c)(10). 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: Linda Gundrum

Date: September 21, 1994