March 23, 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. 57 TO FACILITY OPERATING LICENSE NO. NPF-58
(TAC NOS. M83833 AND M85719)

The Commission has issued the enclosed Amendment No. 57 to Facility Operating License No. NPF-58 for the Perry Nuclear Power Plant, Unit No. 1. This amendment revises the Technical Specifications (TSs) in response to your application dated June 24, 1992 (PY-CEI/NRR-1510 L), as supplemented by letter dated September 25, 1992 (PY-CEI/NRR-1543 L), and application dated November 16, 1992 (PY-CEI/NRR-1537 L).

This amendment changes various TS limits regarding primary containment pressure and temperature and suppression pool water levels during plant operations based upon a revised containment response analysis. Provisions have also been incorporated into the TSs to permit a reduction in the water level of the upper containment pool during plant operation, provided the suppression pool water level is increased to compensate.

A copy of the Safety Evaluation is also enclosed. Notice of issuance has been forwarded to the Office of Federal Register for publication.

Sincerely,
Original signed by Jon B. Hopkins

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

Enclosures:
1. Amendment No. 57 to License No. NPF-58
2. Safety Evaluation

cc w/enclosures:
See next page

LA/PD33	PM/PD33	BC/SCSB	BC/OTSB	PD/PD33	OGC-WF1
MRushbrook	JHopkins	RBarrett	CGrimes	JHannon	C. Marco
10/6/93*	2/9/94	1/26/94*	2/18/94	2/14/94	2/9/94

JBH JBH #94-032

*See previous concurrence

ORIGINAL OFFICE COPY DOCUMENT NAME: G:\PERRY\PER83833.AMD

JBH 3/17/94

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

March 23, 1994

Docket No. 50-440

Mr. Robert A. Stratman
Vice President Nuclear - Perry
Centerior Service Company
P. O. Box 97, S270
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See next page

Mr. Robert A. Stratman
Centerior Service Company

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. 57
License No. NPF-58

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

   A. The applications 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 November 16, 1992, and June 24, 1992, as supplemented by letter dated September 25, 1992, comply 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. 57 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 July 15, 1994.

FOR THE NUCLEAR REGULATORY COMMISSION

Jon B. Hopkins

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

Attachment:
Changes to the Technical
Specifications

Date of issuance: March 23, 1994

ATTACHMENT TO LICENSE AMENDMENT NO. 57

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
3/4 5-8	3/4 5-8
3/4 5-9	3/4 5-9
3/4 6-1	3/4 6-1
3/4 6-3 through 3/4 6-7	3/4 6-3 through 3/4 6-7
3/4 6-11	3/4 6-11
3/4 6-22 through 3/4 6-24	3/4 6-22 through 3/4 6-24
3/4 6-27	3/4 6-27
3/4 8-11	3/4 8-11
3/4 9-10	3/4 9-10
3/4 9-16	3/4 9-16
3/4 9-17	3/4 9-17
B 3/4 5-2	B 3/4 5-2
B 3/4 6-1	B 3/4 6-1
B 3/4 6-2a	B 3/4 6-2a
B 3/4 6-4	B 3/4 6-4
---	B 3/4 6-4a
---	B 3/4 6-4b
B 3/4 6-5	B 3/4 6-5
B 3/4 9-2	B 3/4 9-2

3/4.5.3 SUPPRESSION POOL

LIMITING CONDITION FOR OPERATION

3.5.3 The suppression pool shall be OPERABLE:

a. In OPERATIONAL CONDITION 1, 2 and 3, with a minimum water level greater than or equal to the limit of Specification 3.6.3.1.

b. In OPERATIONAL CONDITION 4 and 5*, with a water level of at least 16'6", except that the suppression pool level may be less than the limit or may be drained provided that:

   1. No operations are performed that have a potential for draining the reactor vessel,

   2. The reactor mode switch is locked in the Shutdown or Refuel position,

   3. The condensate storage tank contains at least 150,000 available gallons of water, equivalent to a level of 47% (220,000 gallons of water), and

   4. The HPCS system is OPERABLE per Specification 3.5.2 with an OPERABLE flow path capable of taking suction from the condensate storage tank and transferring the water through the spray sparger to the reactor vessel.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, 3, 4 and 5*.

ACTION:

a. In OPERATIONAL CONDITION 1, 2 or 3, with the suppression pool water level less than the above limit, restore the water level to within the limit within 1 hour 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 4 or 5*, with the suppression pool water level less than the above limit or drained and the above required conditions not satisfied, suspend CORE ALTERATIONS and all operations that have a potential for draining the reactor vessel and lock the reactor mode switch in the Shutdown position. Establish PRIMARY CONTAINMENT INTEGRITY within 8 hours.

---

*The suppression pool is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded, the steam dryer storage/reactor well gate is removed and the water level in these upper containment pools is maintained within the limits of Specification 3.9.8 and 3.9.9.

SURVEILLANCE REQUIREMENTS

4.5.2.1 At least the above required ECCS shall be demonstrated OPERABLE per Surveillance Requirement 4.5.1.

4.5.2.2 The HPCS system shall be determined OPERABLE at least once per 12 hours by verifying the condensate storage tank required volume when the condensate storage tank is required to be OPERABLE per Specification 3.5.2.e.

SURVEILLANCE REQUIREMENTS

4.5.3.1 The suppression pool shall be determined OPERABLE by verifying the water level to be greater than or equal to:

a. the minimum water level of Specification 3.6.3.1 at least once per 24 hours in OPERATIONAL CONDITIONS 1, 2 and 3.

b. 16'6" at least once per 12 hours in OPERATIONAL CONDITIONS 4 and 5.

4.5.3.2 With the suppression pool level less than the above limit or drained in OPERATIONAL CONDITION 4 or 5*, at least once per 12 hours:

a. Verify the required conditions of Specification 3.5.3.b to be satisfied, or

b. Verify footnote conditions * to be satisfied.

*The suppression pool is not required to be OPERABLE provided that the reactor vessel head is removed, the cavity is flooded, the steam dryer storage/reactor well gate is removed and the water level in these upper containment pools is maintained within the limits of Specification 3.9.8 and 3.9.9.

3/4.6 CONTAINMENT SYSTEMS

3/4.6.1 PRIMARY CONTAINMENT

PRIMARY CONTAINMENT INTEGRITY - OPERATING

LIMITING CONDITION FOR OPERATION

3.6.1.1.1 PRIMARY CONTAINMENT INTEGRITY shall be maintained.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, restore PRIMARY CONTAINMENT INTEGRITY within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.1.1 PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

a. After each closing of each penetration subject to Type B testing, except the primary containment air locks, if opened following Type A or B test, by leak rate testing the seals with gas at P_a and verifying that when the measured leakage rate for these seals is added to the leakage rates determined pursuant to Surveillance Requirement 4.6.1.2.d for all other Type B and C penetrations, the combined leakage rate is less than or equal to 0.60 L_a.

b. At least once per 31 days by verifying that all primary containment penetrations** not capable of being closed by OPERABLE primary containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except for valves that may be opened as permitted by Specification 3.6.4.

c. By verifying each primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.

d. By verifying the suppression pool is in compliance with the requirements of Specification 3.6.3.1.

*See Special Test Exception 3.10.1.

**Except valves, blind flanges, and deactivated automatic valves which are located inside the primary containment, drywell, or the steam tunnel portion of the auxiliary building, and are locked, sealed, or otherwise secured in the closed position. These penetrations shall be verified closed during each COLD SHUTDOWN except such verification need not be performed more often than once per 92 days.

3/4.6.1 PRIMARY CONTAINMENT

PRIMARY CONTAINMENT INTEGRITY - SHUTDOWN

LIMITING CONDITION FOR OPERATION

3.6.1.1.2 PRIMARY CONTAINMENT INTEGRITY* shall be maintained.#

APPLICABILITY:

When irradiated fuel is being handled in the primary containment, and during CORE ALTERATIONS, and operations with a potential for draining the reactor vessel. Under these conditions, the requirements of PRIMARY CONTAINMENT INTEGRITY do not apply to normal operation of the inclined fuel transfer system.

ACTION:

Without PRIMARY CONTAINMENT INTEGRITY, suspend handling of irradiated fuel in the primary containment, CORE ALTERATIONS, and operations with a potential for draining the reactor vessel.

SURVEILLANCE REQUIREMENTS

4.6.1.1.2 PRIMARY CONTAINMENT INTEGRITY shall be demonstrated:

a. At least once per 31 days by verifying that all primary containment penetrations not capable of being closed by OPERABLE primary containment automatic isolation valves and required to be closed during accident conditions are closed by valves, blind flanges, or deactivated automatic valves secured in position, except for valves that may be opened as permitted by Specification 3.6.4.#

b. By verifying each primary containment air lock is in compliance with the requirements of Specification 3.6.1.3.

---

*The primary containment leakage rates in accordance with Specification 3.6.1.2 are not applicable.

#Except that six (6) 3/4" vent and drain line pathways may be opened for the purpose of performing containment isolation valve leak rate testing provided the plant has been subcritical for at least seven (7) days.

LIMITING CONDITION FOR OPERATION

3.6.1.2 Primary containment leakage rates shall be limited to:

a. An overall integrated leakage rate of less than or equal to 0.75 L_a, 0.20 percent by weight of the primary containment air per 24 hours at P_a.

b. A combined leakage rate of less than or equal to 0.60 L_a for all penetrations and all valves, except for main steam line isolation valves and valves which are hydrostatically leak tested, subject to Type B and C tests when pressurized to P_a.

c. Less than or equal to 25 scf per hour for any one main steam line through the isolation valves when tested at P_a.

d. A combined leakage rate of less than or equal to 0.0504 L_a for all penetrations that are secondary containment bypass leakage paths when pressurized to the required test pressure.

e. A combined leakage rate of less than or equal to 1 gpm times the total number of containment isolation valves in hydrostatically tested lines which penetrate the primary containment, when tested at 1.10 P_a.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2* and 3.

ACTION:

With:

a. The measured overall integrated primary containment leakage rate exceeding 0.75 L_a, or

b. The measured combined leakage rate for all penetrations and all valves except for main steam line isolation valves and valves which are hydrostatically leak tested, subject to Type B and C tests exceeding 0.60 L_a, or

c. The measured leakage rate exceeding 25 scf per hour for any one main steam line through the isolation valves, or

d. The combined leakage rate for all penetrations that are secondary containment bypass leakage paths exceeding 0.0504 L_a, or

e. The measured combined leakage rate for all containment isolation valves in hydrostatically tested lines which penetrate the primary containment exceeding 1 gpm times the total number of such valves:

*See Special Test Exception 3.10.1.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

restore:

a. The overall integrated leakage rate(s) to less than or equal to 0.75 L_a, and

b. The combined leakage rate for all penetrations and all valves, except for main steam line isolation valves and valves which are hydrostatically leak tested, subject to Type B and C tests to less than or equal to 0.60 L_a, and

c. The leakage rate to less than 25 scf per hour for any one main steam line through the isolation valves, and

d. The combined leakage rate for all penetrations that are secondary containment bypass leakage paths to less than or equal to 0.0504 L_a, and

e. The combined leakage rate for all containment isolation valves in hydrostatically tested lines which penetrate the primary containment to less than or equal to 1 gpm times the total number of such valves,

prior to increasing reactor coolant system temperature above 200°F.

SURVEILLANCE REQUIREMENTS

4.6.1.2 The primary containment leakage rates shall be demonstrated at the following test schedule and shall be determined in conformance with the criteria specified in Appendix J of 10 CFR Part 50 using the methods and provisions of ANSI N45.4-1972 and BN-TOP-1; test results shall also be reported based on the Mass Point Methodology described in ANSI/ANS N56.8-1981:

a. Three Type A Overall Integrated Containment Leakage Rate tests shall be conducted at 40 ± 10 month intervals during shutdown at P_a during each 10-year service period. The third test of each set shall be conducted during the shutdown for the 10-year plant inservice inspection.

b. If any periodic Type A test fails to meet 0.75 L_a, the test schedule for subsequent Type A tests shall be reviewed and approved by the Commission. If two consecutive Type A tests fail to meet 0.75 L_a, a Type A test shall be performed at least every 18 months until two consecutive Type A tests meet 0.75 L_a, at which time the above test schedule may be resumed.

c. The accuracy of each Type A test shall be verified by a supplemental test which:

SURVEILLANCE REQUIREMENTS (Continued)

i. Purge supply and exhaust isolation valves with resilient material seals shall be tested and demonstrated OPERABLE per Surveillance Requirements 4.6.1.8.3. and 4.6.1.8.4.

j. The provisions of Specification 4.0.2 are not applicable to Specifications 4.6.1.2.a, 4.6.1.2.b, 4.6.1.2.c, and 4.6.1.2.d.

(Next page is 3/4 6-6.)

SURVEILLANCE REQUIREMENTS (Continued)

1. Confirms the accuracy of the test by verifying that the difference between the supplemental data and the Type A test data is within 0.25 L_a. The formula to be used is:

   $[L_o + L_{am} - 0.25\ L_a] \leq L_c \leq [L_o + L_{am} + 0.25L_a]$ where

   L_c = supplemental test result; L_o = superimposed leakage;

   L_{am} = measured Type A leakage.

2. Has duration sufficient to establish accurately the change in leakage rate between the Type A test and the supplemental test.

3. Requires the quantity of gas injected into the primary containment or bled from the primary containment during the supplemental test to be between 0.75 L_a and 1.25 L_a.

d. Type B and C tests shall be conducted with gas at P_a* at intervals no greater than 24 months except for tests involving:

   1. Air locks,

   2. Main steam line isolation valves,

   3. Valves pressurized with fluid from a seal system,

   4. All containment isolation valves in hydrostatically tested lines which penetrate the primary containment, and

   5. Purge supply and exhaust isolation valves with resilient material seals.

e. Air locks shall be tested and demonstrated OPERABLE per Surveillance Requirement 4.6.1.3.

f. Main steam line isolation valves shall be leak tested at least once per 18 months.

g. Leakage from isolation valves that are sealed with fluid from a seal system may be excluded, subject to the provisions of Appendix J of 10 CFR 50 Section III.C.3, when determining the combined leakage rate provided the seal system and valves are pressurized to at least 1.10 P_a and the seal system capacity is adequate to maintain system pressure for at least 30 days.

h. All containment isolation valves in hydrostatically tested lines which penetrate the primary containment shall be leak tested at least once per 18 months.

*Unless a hydrostatic test is required.

LIMITING CONDITION FOR OPERATION (Continued)

SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

Otherwise, 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 3.0.4 are not applicable.

b. With a primary containment air lock inoperable in OPERATIONAL CONDITIONS 1, 2 or 3, except as a result of an inoperable air lock door and/or interlock mechanism, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With a primary containment air lock inoperable, in OPERATIONAL CONDITION #, except as a result of an inoperable air lock door and/or interlock mechanism, maintain at least one air lock door closed; restore the inoperable air lock to OPERABLE status within 24 hours or suspend all operations involving handling of irradiated fuel in the primary containment, CORE ALTERATIONS, and operations with a potential for draining the reactor vessel.

LIMITING CONDITION FOR OPERATION

3.6.1.3 Each primary containment air lock shall be OPERABLE with:

a. Both doors closed except when the air lock is being used for normal transit entry and exit through the containment, then at least one air lock door shall be closed, and

b. An overall air lock leakage rate of less than or equal to 2.5 scf per hour at P_a.

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

ACTION:

a. With one or both air locks having:

1. an inoperable interlock mechanism, for each affected air lock,

a) Maintain at least one OPERABLE air lock door closed* and within 24 hours lock one OPERABLE air lock door closed.

b) Operation may then continue provided that at least once per 31 days, one OPERABLE air lock door is verified to be locked closed*.

2. one inoperable air lock door, or, both one inoperable door and an inoperable interlock mechanism, for each affected air lock,

a) Maintain at least the OPERABLE air lock door closed** and within 24 hours lock the OPERABLE air lock door closed.

b) Operation may then continue until performance of the next required overall air lock leakage test provided that at least once per 31 days the OPERABLE air lock door is verified to be locked closed**.

Otherwise, in OPERATIONAL CONDITION 1, 2, or 3, be in at least HOT

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

* Entry into and exit from the air lock(s) or primary containment, including through a "locked closed" door, is permitted under administrative controls.

** If one or both air locks have one inoperable door, entry into and exit from the air lock(s) through the OPERABLE door is permitted under administrative controls to perform repairs of the affected air lock components. Also, if both air locks have one inoperable door, entry into and exit from primary containment is permitted under administrative controls for 7 days.

SURVEILLANCE REQUIREMENTS

4.6.1.3 Each primary containment air lock shall be demonstrated OPERABLE:

a. By verifying seal leakage rate less than or equal to 2.5 scf per hour when the gap between the door seals is pressurized to P_a:

   1. within 72 hours# following each closing, except when the air lock is being used for multiple entries, then at least once per 72 hours#; and

   2. prior to establishing PRIMARY CONTAINMENT INTEGRITY when the air lock has been used and no maintenance has been performed on the airlock.*

b. By verifying at least once per 7 days that the service and instrument air systems pressure in the header to the primary containment air lock is $\geq$ 90 psig.

c. By conducting an overall air lock leakage test at P_a and verifying that the overall air lock leakage rate is within its limit:

   1. At least once per 6 months#,

   2. Prior to establishing PRIMARY CONTAINMENT INTEGRITY when maintenance has been performed on the air lock that could affect the air lock sealing capability.*

d. At least once per 6 months by verifying that only one door in each air lock can be opened at a time.

e. By verifying the door inflatable seal system OPERABLE at least once per 18 months by conducting a seal pneumatic system leak test and verifying that system pressure does not decay more than 1.5 psig from 90 psig within 24 hours.

#The provisions of Specification 4.0.2 are not applicable.

*Exemption to Appendix J of 10 CFR 50.

MSIV LEAKAGE CONTROL SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.1.4 Two independent MSIV leakage control system (LCS) subsystems shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2, and 3.

ACTION:

With one MSIV leakage control 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

4.6.1.4 Each MSIV leakage control system subsystem shall be demonstrated OPERABLE:

a. At least once per 31 days by verifying:

   1. Blower OPERABILITY by starting the blower(s) from the control room and operating the blower(s) for at least 15 minutes.

   2. Inboard heater OPERABILITY by demonstrating electrical continuity of the heating element circuitry by verifying the inboard heater draws 8.28 ± 10% amperes per phase.

b. During each COLD SHUTDOWN, if not performed within the previous 92 days, by cycling each motor operated valve, including the main steam stop valves, through at least one complete cycle of full travel.

c. At least once per 18 months by:

   1. Performance of a functional test which includes simulated actuation of the subsystem throughout its operating sequence, and verifying that each automatic valve actuates to its correct position, and the blower(s) start(s).

   2. Verifying that the blower(s) develop(s) at least the below required vacuum at the rated capacity:

      a) Inboard system, 15" H_2O at $\geq$ 100 scfm.

      b) Outboard system, 15" H_2O at $\geq$ 200 scfm.

d. By verifying the inboard flow and inboard and outboard pressure instrumentation to be OPERABLE by performance of a:

   1. CHANNEL FUNCTIONAL TEST at least once per 31 days, and
   2. CHANNEL CALIBRATION at least once per 18 months.

PRIMARY CONTAINMENT AVERAGE AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.1.7 Primary containment average air temperature shall not exceed 95°F.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With the primary containment average air temperature greater than 95°F, reduce the average air temperature to within the limit within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.1.7 The primary containment average air temperature shall be the arithmetical average* of the temperatures at the following locations and shall be determined to be within the limit at least once per 24 hours:

	Elevation	Azimuth
a.	720'-6"	280°
b.	720'-6"	100°
c.	689'-4"	40°
d.	689'-4"	210°
e.	647'-0"	54°
f.	645'-6"	251°
g.	613'-0"	69°
h.	613'-0"	251°

*At least one reading from each elevation for an arithmetical average. However, all available instruments should be used in calculating the arithmetical average.

LIMITING CONDITION FOR OPERATION

3.6.1.8 The drywell and containment purge 42-inch outboard (1M14-F040, F090) supply and exhaust isolation valves and the 18-inch supply and exhaust isolation valves (1M14-F190, F195, F200, F205) shall be OPERABLE and:

a. Each 42-inch inboard purge valve (1M14-F045, F085) shall be sealed closed.

b. Each 42-inch outboard purge valve (1M14-F040, F090) may be open limited to an opening angle of 50° or less for purge system operation* with such operation limited to 1000 hours per 365 days for reducing airborne activity and pressure control.

c. Each 24-inch (1M14-F055A, B and F060A, B) and 36-inch (1M14-F065, F070) drywell purge valve shall be sealed closed.

d. Each 2-inch (1M51-F090 and F110) backup hydrogen purge system isolation valves may be open for controlling drywell pressure.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With a 42-inch inboard drywell and containment purge supply and/or exhaust isolation valve(s) open or not sealed closed, within 4 hours close and/or seal the 42-inch valve(s) or otherwise isolate the penetration or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With a 18-inch or 42-inch outboard drywell and containment purge supply and/or exhaust isolation valves inoperable or open for more than 3000 hours per 365 days for purge system operation*, within four hours close the open 18- or 42-inch valve(s) or otherwise isolate the penetration(s) or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With a 24- or 36-inch drywell purge supply and/or exhaust isolation valve(s) open or not sealed closed, within 4 hours close and/or seal close the 24- or 36-inch valve(s) or otherwise isolate the penetration, or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

d. With a drywell and containment purge supply and/or exhaust isolation valve(s) with resilient material seals having a measured leakage rate exceeding the limit of Surveillance Requirement 4.6.1.8.3 and/or

*Purge system operation shall be defined as any time that both 18-inch and the 42-inch outboard purge valves are open concurrently in either the supply or exhaust line.

3/4.6.3 DEPRESSURIZATION SYSTEMS

SUPPRESSION POOL

LIMITING CONDITION FOR OPERATION

3.6.3.1 The suppression pool shall be OPERABLE with the pool water:

a. Level less than or equal to 18'-6", and greater than or equal to 17'-9.5" plus the level adjustment factor obtained from the Suppression Pool Level Adjustment Graph.

b. Maximum average temperature of 95°F except that the maximum average temperature may be permitted to increase to:

1. 105°F during testing which adds heat to the suppression pool.

2. 110°F with THERMAL POWER less than or equal to 1% of RATED THERMAL POWER.

3. 120°F with the main steam line isolation valves closed following a scram.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With the suppression pool water level outside the above limits, restore the water level to within the limits within 1 hour or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the suppression pool average water temperature greater than 95°F, restore the average temperature to less than or equal to 95°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours, except, as permitted above:

1. With the suppression pool average water temperature greater than 105°F during testing which adds heat to the suppression pool, stop all testing which adds heat to the suppression pool and restore the average temperature to less than 95°F within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

2. With the suppression pool average water temperature greater than:

a) 95°F for more than 24 hours and THERMAL POWER greater than 1% of RATED THERMAL POWER, be in at least HOT SHUTDOWN within 12 hours and in COLD SHUTDOWN within the next 24 hours.

b) 110°F, place the reactor mode switch in the Shutdown position and operate at least one residual heat removal loop in the suppression pool cooling mode.

DRYWELL AVERAGE AIR TEMPERATURE

LIMITING CONDITION FOR OPERATION

3.6.2.6 Drywell average air temperature shall not exceed 145°F.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

With the drywell average air temperature greater than 145°F, reduce the average air temperature to within the limit within 8 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.2.6 The drywell average air temperature shall be the arithmetical average* of the temperatures at the following elevations# and shall be determined to be within the limit at least once per 24 hours:

	Elevation	Azimuth
a.	653'-8"	315°, 220°, 135°, 34°
b.	634'-0" - 640'-0"	340°, 308°, 215°, 145°, 30°, 20°
c.	604'-6" - 609'-8"	310°, 308°, 253°, 212°, 150°, 140°, 80°

*At least one reading from each elevation for an arithmetical average.

#The temperature at each elevation shall be the arithmetical average of the temperatures obtained from all available instruments at that elevation.

Correction Letter of 7-28-88

SURVEILLANCE REQUIREMENTS (Continued)

c. By verifying at least 16 suppression pool water temperature instrumentation channels, at least two channels in each suppression pool sector, OPERABLE by performance of a:

1. CHANNEL CHECK at least once per 24 hours,

2. CHANNEL FUNCTIONAL TEST at least once per 31 days, and

3. CHANNEL CALIBRATION at least once per 18 months,

with the water high temperature alarm setpoint for $\leq$95°F.

LIMITING CONDITION FOR OPERATION (Continued)

ACTION (Continued)

c) 120°F, depressurize the reactor pressure vessel to less than 200 psig within 12 hours.

c. With only one suppression pool water level indicator OPERABLE and/or with less than eight suppression pool water temperature indicators, one in each of the eight locations OPERABLE, restore the inoperable indicator(s) to OPERABLE status within 7 days or verify suppression pool water level and/or temperature to be within the limits at least once per 12 hours.

d. With no suppression pool water level indicators OPERABLE and/or with less than seven suppression pool water temperature indicators covering at least seven locations OPERABLE, restore at least one water level indicator and at least seven water temperature indicators 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.

SURVEILLANCE REQUIREMENTS

4.6.3.1 The suppression pool shall be demonstrated OPERABLE:

a. By verifying the suppression pool water level to be within the limits at least once per 24 hours.

b. At least once per 24 hours in OPERATIONAL CONDITION 1 or 2 by verifying the suppression pool average water temperature to be less than or equal to 95°F, except:

1. At least once per 5 minutes during testing which adds heat to the suppression pool, by verifying the suppression pool average water temperature less than or equal to 105°F.

2. At least once per hour when suppression pool average water temperature is greater than or equal to 95°F, by verifying suppression pool average water temperature to be less than or equal to 110°F, and THERMAL POWER to be less than or equal to 1% of RATED THERMAL POWER after suppression pool average water temperature has exceeded 95°F for more than 24 hours.

3. At least once per 30 minutes following a scram with suppression pool average water temperature greater than or equal to 95°F, by verifying suppression pool average water temperature less than or equal to 120°F.

SUPPRESSION POOL MAKEUP SYSTEM

LIMITING CONDITION FOR OPERATION

3.6.3.4 The suppression pool makeup system shall be OPERABLE.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With one suppression pool makeup line inoperable, restore the inoperable makeup line 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.

b. With the upper containment pool water level less than the limit specified in Surveillance Requirement 4.6.3.4.a.1.a, within 4 hours either:

   1. Restore the water level to within the limit, or

   2. Maintain the upper containment pool water level greater than the limit specified in Surveillance Requirement 4.6.3.4.a.1.b, and raise the suppression pool water level in accordance with Surveillance Requirement 4.6.3.4.a.1.b.

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

c. With upper containment pool water temperature greater than the limit, restore the upper containment pool water temperature to within the limit within 24 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

SURVEILLANCE REQUIREMENTS

4.6.3.4 The suppression pool makeup system shall be demonstrated OPERABLE:

a. At least once per 24 hours by verifying the upper containment pool water:

   1. Level to be greater than or equal to:

      a) 22'-9" above the reactor pressure vessel flange, or

      b) 22'-5" above the reactor pressure vessel flange, and a minimum suppression pool water level greater than or equal to the minimum value specified in LCO 3.6.3.1.a plus 2.20 inches,

   and

   2. Temperature to be less than or equal to 110°F.

b. At least once per 31 days by verifying that:

   1. All upper containment pool gates are removed (except the fuel transfer pool gate may be installed).

   2. Each valve, manual, power operated or automatic, in the flow path that is not locked, sealed, or otherwise secured in position, is in its correct position.

c. At least once per 18 months by performing a system functional test which includes simulated automatic actuation of the system throughout its emergency operating sequence and verifying that each automatic valve in the flow path actuates to its correct position. Actual makeup of water to the suppression pool may be excluded from this test.

3/4.6.4 CONTAINMENT ISOLATION VALVES

LIMITING CONDITION FOR OPERATION

3.6.4 Each containment isolation valve shall be OPERABLE.#

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

ACTION:

a. With one or more of the containment isolation valves inoperable, maintain at least one isolation valve OPERABLE in each affected penetration that is open and within 4 hours either:

   1. Restore the inoperable valve(s) to OPERABLE status, or

   2. Isolate each affected penetration by use of at least one deactivated automatic valve secured in the isolated position,* or

   3. Isolate each affected penetration by use of at least one closed manual valve or blind flange.*

   The provisions of Specification 3.0.4 are not applicable provided that the affected penetration is isolated in accordance with ACTION a.2 or a.3 above, and provided that the associated system, if applicable, is declared inoperable and the appropriate ACTION statements for that system are performed.

   Otherwise, 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.

   Otherwise, in OPERATIONAL CONDITION**, suspend all operations involving CORE ALTERATIONS, handling of irradiated fuel in the primary containment and operations with a potential for draining the reactor vessel. The provisions of Specification 3.0.3 are not applicable.

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*Isolation valves closed to satisfy these requirements may be reopened on an intermittent basis under administrative controls.

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

#The Containment Vessel and Drywell Purge system 42-inch inboard purge valves 1M14-F045 and -F085 are not required to be OPERABLE in OPERATIONAL CONDITIONS 1, 2 and 3. The RCIC system containment isolation valves are not required to be OPERABLE in OPERATIONAL CONDITION **. The Fire Protection system manual hose reel containment isolation valves 1P54-F726 and -F727 may be opened as necessary to supply fire mains in OPERATIONAL CONDITION **. Locked or sealed closed isolation valves may be opened on an intermittent basis under administrative controls.

LIMITING CONDITION FOR OPERATION

3.8.1.2 As a minimum, the following A.C. electrical power sources shall be OPERABLE:

a. One circuit between the offsite transmission network and the onsite Class 1E distribution system, and

b. Diesel generator Div 1 or Div 2, and diesel generator Div 3 when the HPCS system is required to be OPERABLE, with each diesel generator having:

   1. A day tank containing a minimum of 225 gallons of fuel for Div 1 and Div 2 and 204 gallons of fuel for Div 3.

   2. A fuel storage system containing a minimum of 73,700 gallons of fuel for Div 1 and Div 2 and 36,100 gallons of fuel for Div 3.

   3. A fuel transfer pump.

APPLICABILITY: OPERATIONAL CONDITIONS 4, 5 and *.

ACTION:

a. With less than the offsite circuits and/or diesel generators Div 1 or Div 2 of the above required A.C. electrical power sources OPERABLE, suspend CORE ALTERATIONS, handling of irradiated fuel in the primary containment and Fuel Handling Building, operations with a potential for draining the reactor vessel and crane operations over the spent fuel storage pool when fuel assemblies are therein. In addition, when in OPERATIONAL CONDITION 5 with the water level less than 22 feet 9 inches above the reactor pressure vessel flange, immediately initiate corrective action to restore the required power sources to OPERABLE status as soon as practical.

b. With diesel generator Div 3 of the above required A.C. electrical power sources inoperable, restore the inoperable diesel generator Div 3 to OPERABLE status within 72 hours or declare the HPCS system inoperable and take the ACTION required by Specifications 3.5.2 and 3.5.3.

c. With the fuel oil contained in the storage tank not meeting the properties specified in TS 4.8.1.1.2.d.2 or 4.8.1.1.2.e, the fuel oil shall be brought back within the specified limits within 7 days or the associated diesel generator shall be declared inoperable.

d. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.8.1.2 At least the above required A.C. electrical power sources shall be demonstrated OPERABLE per Surveillance Requirements 4.8.1.1.1, 4.8.1.1.2 (except for the requirement of 4.8.1.1.2.a.5), and 4.8.1.1.3.

*When handling irradiated fuel in the Fuel Handling Building or primary containment.

3/4.8.2 D.C. SOURCES

D.C. SOURCES - OPERATING

LIMITING CONDITION FOR OPERATION

3.8.2.1 As a minimum, the following D.C. electrical power sources shall be OPERABLE:

a. Division 1, consisting of:
   1. 125 volt battery 1R42-S002 or 2R42-S002.
   2. 125 volt full capacity charger 1R42-S006 or 0R42-S007.

b. Division 2, consisting of:
   1. 125 volt battery 1R42-S003 or 2R42-S003.
   2. 125 volt full capacity charger 1R42-S008 or 0R42-S009.

c. Division 3, consisting of:
   1. 125 volt battery 1E22-S005 or 2E22-S005.
   2. 125 volt full capacity charger 1E22-S006 or 0R42-S011.

APPLICABILITY: OPERATIONAL CONDITIONS 1, 2 and 3.

ACTION:

a. With the Unit 1 and Unit 2 Division 1 batteries and/or both chargers of the above required Division 1 D.C. electrical power sources inoperable, restore an inoperable Division 1 battery and charger to OPERABLE status within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

b. With the Unit 1 and Unit 2 Division 2 batteries and/or both chargers of the above required Division 2 D.C. electrical power sources inoperable, restore an inoperable Division 2 battery and charger to OPERABLE status within 2 hours or be in at least HOT SHUTDOWN within the next 12 hours and in COLD SHUTDOWN within the following 24 hours.

c. With the Unit 1 and Unit 2 Division 3 batteries and/or both chargers of the above required Division 3 D.C. electrical power sources inoperable, declare the HPCS system inoperable and take the ACTION required by Specification 3.5.1.

SURVEILLANCE REQUIREMENTS

4.8.2.1 Each of the above required 125 volt batteries and chargers shall be demonstrated OPERABLE:

a. At least once per 7 days by verifying that:
   1. The parameters in Table 4.8.2.1-1 meet the Category A limits, and
   2. Total battery terminal voltage is greater than or equal to 129 volts on float charge.

3/4.9.8 WATER LEVEL- REACTOR VESSEL

LIMITING CONDITION FOR OPERATION

3.9.8 At least 22 feet 9 inches of water shall be maintained over the top of the reactor pressure vessel flange.

APPLICABILITY: During handling of fuel assemblies or control rods within the reactor pressure vessel while in OPERATIONAL CONDITION 5 when the fuel assemblies being handled are irradiated or the fuel assemblies seated within the reactor vessel are irradiated.

ACTION:

With the requirements of the above specification not satisfied, suspend all operations involving handling of fuel assemblies or control rods within the reactor pressure vessel after placing all fuel assemblies and control rods in a safe condition.

SURVEILLANCE REQUIREMENTS

4.9.8 The reactor vessel water level shall be determined to be at least its minimum required depth within 2 hours prior to the start of and at least once per 24 hours during handling of fuel assemblies or control rods within the reactor pressure vessel.

3/4.9.7 CRANE TRAVEL-SPENT FUEL STORAGE POOL, NEW FUEL STORAGE VAULTS, AND UPPER CONTAINMENT FUEL POOL

LIMITING CONDITION FOR OPERATION

3.9.7 Loads which would result in excess of 4000 foot pounds of impact energy if dropped shall be prohibited from travel over fuel assemblies in the spent fuel storage pool racks, new fuel storage vaults, or upper containment fuel pool racks.

APPLICABILITY: With fuel assemblies in the spent fuel storage pool racks, new fuel storage vaults, or upper containment fuel pool racks.

ACTION:

With the requirements of the above specification not satisfied, place the crane load in a safe condition. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.7 Loads, other than fuel assemblies or control rods, shall be verified to result in less than or equal to 4000 foot pounds of impact energy if dropped before travel over fuel assemblies in the spent fuel storage pool racks, new fuel storage vaults, or the upper containment fuel pool racks.

3/4.9.11 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

HIGH WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.11.1 At least one shutdown cooling mode loop of the residual heat removal (RHR) system shall be OPERABLE and in operation with at least:

a. One OPERABLE RHR pump, and

b. Two OPERABLE RHR heat exchangers.

APPLICABILITY: OPERATIONAL CONDITION 5, when irradiated fuel is in the reactor vessel and the water level is greater than or equal to 22 feet 9 inches above the top of the reactor pressure vessel flange and heat losses to the ambient* are not sufficient to maintain OPERATIONAL CONDITION 5.

ACTION:

With no RHR shutdown cooling mode loop OPERABLE, within one hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal. Otherwise, suspend all operations involving an increase in the reactor decay heat load and establish PRIMARY CONTAINMENT INTEGRITY within 4 hours.

SURVEILLANCE REQUIREMENTS

4.9.11.1 At least once per 12 hours verify at least one RHR shutdown cooling mode loop is capable of taking suction from the reactor vessel and discharging back to the reactor vessel through an RHR heat exchanger with available cooling water.

*Ambient losses must be such that no increase in reactor vessel water temperature will occur (even though REFUELING conditions are being maintained).

SURVEILLANCE REQUIREMENTS

4.9.10.2.1 Within 4 hours prior to the start of removal of control rods and/or control rod drive mechanisms from the core and/or reactor pressure vessel and at least once per 24 hours thereafter until all control rods and control rod drive mechanisms are reinstalled and all control rods are inserted in the core, verify that:

a. The reactor mode switch is OPERABLE per Surveillance Requirement 4.3.1.1 or 4.9.1.2, as applicable, and locked in the Shutdown position or in the Refuel position per Specification 3.9.1.

b. The SRM channels are OPERABLE per Specification 3.9.2.

c. The SHUTDOWN MARGIN requirements of Specification 3.1.1 are satisfied.

d. All other control rods are either inserted or have the surrounding four fuel assemblies removed from the core cell.

e. The four fuel assemblies surrounding each control rod and/or control rod drive mechanism to be removed from the core and/or reactor vessel are removed from the core cell.

4.9.10.2.2 Following replacement of all control rods and/or control rod drive mechanisms removed in accordance with this specification, perform a functional test of the "one-rod-out" Refuel position interlock, if this function had been bypassed.

LOW WATER LEVEL

LIMITING CONDITION FOR OPERATION

3.9.11.2 Two shutdown cooling mode loops of the residual heat removal (RHR) system shall be OPERABLE and at least one loop shall be in operation,* with each loop consisting of at least:

a. One OPERABLE RHR pump, and

b. Two OPERABLE RHR heat exchangers.

APPLICABILITY: OPERATIONAL CONDITION 5, when irradiated fuel is in the reactor vessel and the water level is less than 22 feet 9 inches above the top of the reactor pressure vessel flange and heat losses to the ambient are not sufficient** to maintain OPERATIONAL CONDITION 5.

ACTION:

a. With less than the above required shutdown cooling loops of the RHR system OPERABLE, within one hour and at least once per 24 hours thereafter, demonstrate the operability of at least one alternate method capable of decay heat removal for each inoperable RHR shutdown cooling mode loop.

b. With no RHR shutdown cooling mode loop in operation, within one hour establish reactor coolant circulation by an alternate method and monitor reactor coolant temperature at least once per hour.

SURVEILLANCE REQUIREMENTS

4.9.11.2 At least one shutdown cooling mode loop of the residual heat removal system or alternate method shall be verified to be in operation and circulating reactor coolant at least once per 12 hours.

*The shutdown cooling pump may be removed from operation for up to 2 hours per 8-hour period.
**Ambient losses must be such that no increase in reactor vessel water temperature will occur (even though REFUELING conditions are being maintained).

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 of the twelve proximity sensors and at least one liquid level sensor is OPERABLE.

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

ECCS-OPERATING AND SHUTDOWN (Continued)

analysis, will automatically provide makeup at reactor operating pressures on a reactor low water level condition. The HPCS out-of-service period of 14 days is based on the demonstrated OPERABILITY of redundant and diversified low pressure core cooling systems.

The surveillance requirements provide adequate assurance that the HPCS system will be OPERABLE when required. Flow and total developed head values for surveillance testing include system losses to ensure design requirements are met. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test with reactor vessel injection requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage and to provide cooling at the earliest moment.

Upon failure of the HPCS system to function properly after a small break loss-of-coolant accident, the automatic depressurization system (ADS) automatically causes selected safety-relief valves to open, depressurizing the reactor so that flow from the low pressure core cooling systems can enter the core in time to limit fuel cladding temperature to less than 2200°F. ADS is conservatively required to be OPERABLE whenever reactor vessel pressure exceeds 100 psig even though LPCS flow is 6110 gpm rated flow at 128 psid, and LPCI flow is 7100 gpm rated flow at 24 psid.

ADS automatically controls eight selected safety-relief valves although the safety analysis only takes credit for seven valves. It is therefore appropriate to permit one valve to be out-of-service for up to 14 days without materially reducing system reliability. In the event that the ADS safety related instrument air header(s) low pressure alarm system instrumentation channel(s) is inoperable, alternate indication is provided.

3/4.5.3 SUPPRESSION POOL

The suppression pool is required to be OPERABLE as part of the ECCS to ensure that a sufficient supply of water is available to the HPCS, LPCS and LPCI systems in the event of a LOCA. This limit on suppression pool minimum water volume ensures that sufficient water is available to permit recirculation cooling flow to the core. The OPERABILITY of the suppression pool in OPERATIONAL CONDITIONS 1, 2, or 3 is required by Specification 3.6.3.1. See that Specification for a detailed discussion of the Suppression Pool temperature and level limits.

Repair work might require making the suppression pool inoperable. This specification will permit those repairs to be made and at the same time give assurance that the irradiated fuel has an adequate cooling water supply when the suppression pool must be made inoperable, including draining, in OPERATIONAL CONDITIONS 4 or 5.

In OPERATIONAL CONDITION 4 and 5 the suppression pool minimum required water volume (106,508 cubic feet) is reduced because the reactor coolant is maintained at or below 200°F. Since pressure suppression is not required below 212°F, the minimum required water volume is based on NPSH, recirculation volume, and vortex prevention plus a safety margin for conservatism.

3/4.5.1 and 3/4.5.2 ECCS - OPERATING and SHUTDOWN

ECCS division 1 consists of the low pressure core spray system and low pressure coolant injection subsystem "A" of the RHR system and the automatic depressurization system (ADS) as actuated by ADS trip system "A". ECCS division 2 consists of low pressure coolant injection subsystems "B" and "C" of the RHR system and the automatic depressurization system (ADS) as actuated by ADS trip system "B".

The low pressure core spray (LPCS) system and the low pressure coolant injection (LPCI) system is provided to assure that the core is adequately cooled following a loss-of-coolant accident and provides adequate core cooling capacity for all break sizes up to and including the double-ended reactor recirculation line break, and for smaller breaks following depressurization by the ADS.

The LPCS and LPCI are sources of emergency core cooling after the reactor vessel is depressurized and a source for flooding of the core in case of accidental draining.

The surveillance requirements provide adequate assurance that the LPCS and LPCI systems will be OPERABLE when required. Although all active components are testable and full flow can be demonstrated by recirculation through a test loop during reactor operation, a complete functional test requires reactor shutdown. The pump discharge piping is maintained full to prevent water hammer damage to piping and to start cooling at the earliest moment.

ECCS division 3 consists of the high pressure core spray system. The high pressure core spray (HPCS) system is provided to assure that the reactor core is adequately cooled to limit fuel clad temperature in the event of a small break in the reactor coolant system and loss of coolant which does not result in rapid depressurization of the reactor vessel. The HPCS system permits the reactor to be shut down while maintaining sufficient reactor vessel water level inventory until the vessel is depressurized. The HPCS system operates over a range of 1177 psid, differential pressure between the reactor vessel and HPCS suction source, to 0 psid.

The capacity of the system is selected to provide the required core cooling. The HPCS pump is designed to deliver greater than or equal to 517/1550/6110 gpm at differential pressures of 1177/1147/200 psid. Initially, water from the condensate storage tank is used instead of injecting water from the suppression pool into the reactor, but no credit is taken in the safety analyses for the condensate storage tank water.

With the HPCS system inoperable, adequate core cooling is assured by the OPERABILITY of the redundant and diversified automatic depressurization system and both the LPCS and LPCI systems. In addition, the reactor core isolation cooling (RCIC) system, a system for which no credit is taken in the safety

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 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 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.

3/4.6.1 CONTAINMENT (Continued)

3/4.6.1.2 CONTAINMENT LEAKAGE (Continued)

The surveillance testing for measuring leakage rates is consistent with the requirements of Appendix J to 10 CFR 50 with the exception of exemptions granted for testing the air locks after each opening.

3/4.6.1.3 CONTAINMENT AIR LOCKS

The limitations on closure and leak rate for the containment air locks are required to meet the restrictions on PRIMARY CONTAINMENT INTEGRITY and the containment leakage rate given in Specifications 3.6.1.1 and 3.6.1.2. The specification makes allowances for the fact that there may be long periods of time when the air locks will be in a closed and secured position during reactor operation. Only one closed door in each air lock is required to maintain the integrity of the containment.

An allowance has been provided within Action a.1 for access into or through the containment air locks when an interlock mechanism in one or both air locks is inoperable. Action a.1 requires that at least one of the two OPERABLE doors for each affected air lock be maintained closed, and if the interlock mechanism has not been restored to OPERABLE status within 24 hours, one door must be locked closed. The provisions of footnote * may be utilized for entries and exits. The administrative controls of footnote * allow the unlocking and use of the air lock provided that an individual is stationed at the air lock, dedicated to assuring that at least one OPERABLE air lock door remains closed at all times. This allowance is provided to address those situations when the use of an air lock with only an inoperable interlock mechanism may be preferred over the use of the other air lock, such as when the other air lock has an inoperable door.

An allowance has also been provided in Action a.2 for access into or through the containment air locks when one air lock door in one or both air locks is inoperable. The first sentence of footnote ** provides that entry and exit through the OPERABLE door on one or both air locks is permissible under administrative controls for the performance of repairs of the affected air lock components. The second sentence of footnote ** provides for entry into and exit from the containment for activities other than just the repairs of affected air lock components under administrative controls, but only permits these entries when both air locks have an inoperable door, and limits such use to a 7 day period. The administrative controls for the second sentence shall define limits on entry and exit, in order to minimize openings of the OPERABLE door.

3/4.6.1.3 CONTAINMENT AIR LOCKS (Continued)

The administrative controls for both sentences of footnote ** include provisions that after each entry and exit, the OPERABLE door must be promptly closed. The allowances of footnote ** are acceptable because of the low probability of an event that could pressurize the containment during the short time that the OPERABLE door will be open for entry into and exit from the containment.

The air supply to the containment air lock and seal system is the service and instrument air system. The system consists of two 100% capacity air compressors per unit and can be cross-connected. This system is redundant and extremely reliable and provides system pressure indication in the control room.

3/4.6.1.4 MSIV LEAKAGE CONTROL SYSTEM

Calculated doses resulting from the maximum leakage allowance for the main steam line isolation valves in the postulated LOCA situations would be a small fraction of the 10 CFR 100 guidelines, provided the main steam line system from the isolation valves up to and including the turbine condenser remains intact. Operating experience has indicated that degradation has occasionally occurred in the leak tightness of the MSIV's such that the specified leakage requirements have not always been maintained continuously. The requirement for the leakage control system will reduce the untreated leakage from the MSIV's when isolation of the primary system and containment is required.

3/4.6.1.5 CONTAINMENT STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the containment will be maintained comparable to the original design standards for the life of the unit. Structural integrity is required to ensure that the containment will withstand the maximum pressure of 15 psig in the event of a LOCA. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.1.6 CONTAINMENT INTERNAL PRESSURE

The limitations on primary containment to secondary containment differential pressure ensure that the primary containment peak pressure of 7.80 psig does not exceed the design pressure of 15.0 psig during LOCA conditions or that the external pressure differential does not exceed the design maximum external pressure differential of +0.8 psid. The limit of -0.1 to +1.0 psid for initial positive primary containment to secondary containment pressure will limit the primary containment pressure to 7.80 psig which is less than the design pressure and is consistent with the safety analysis.

3/4.6.1.7 CONTAINMENT AVERAGE AIR TEMPERATURE

The limitation on containment average air temperature ensures that the containment peak air temperature does not exceed the design temperature of 185°F during LOCA conditions and is consistent with the safety analysis.

3/4.6.1.8 DRYWELL AND CONTAINMENT PURGE SYSTEM

The use of the drywell and containment purge lines is restricted to the 42-inch outboard and 18-inch purge supply and exhaust isolation valves. These valves will close during a LOCA or steam line break accident and therefore the site boundary dose guidelines of 10 CFR Part 100 would not be exceeded in the event of an accident during purging operations. The term sealed closed as used in this context means that the valve is secured in its closed position by deactivating the valve motor operator, and does not pertain to injecting seal water between the isolation valves by a seal water system.

3/4.6.2.4 DRYWELL STRUCTURAL INTEGRITY

This limitation ensures that the structural integrity of the drywell will be maintained comparable to the original design specification for the life of the unit. A visual inspection in conjunction with Type A leakage tests is sufficient to demonstrate this capability.

3/4.6.2.5 DRYWELL INTERNAL PRESSURE

The limitations on drywell-to-containment differential pressure ensure that the drywell peak calculated pressure of 21.8 psig does not exceed the design pressure of 30.0 psig and that the containment peak pressure of 7.80 psig does not exceed the design pressure of 15.0 psig during LOCA conditions. The maximum external drywell pressure differential is limited to +0.5 psid, well below the 2.4 psid at which suppression pool water will be forced over the weir wall and into the drywell. The limit of 2.0 psid for initial positive drywell to containment pressure will limit the drywell pressure to 21.8 psig which is less than the design pressure and is consistent with the safety analysis.

3/4.6.2.6 DRYWELL AVERAGE AIR TEMPERATURE

The drywell average temperature is an input parameter to the containment/ drywell response analyses as the result of a DBA-LOCA. Furthermore, the drywell average temperature is important in drywell equipment qualification considerations.

3/4.6.3 DEPRESSURIZATION SYSTEMS

The specifications of this section ensure that the drywell and containment pressure will not exceed the design pressure of 30 psig and 15 psig, respectively, during primary system blowdown from full operating pressure.

The suppression pool water volume must absorb the associated decay and structural sensible heat released during a reactor blowdown from 1045 psig. Using conservative parameter inputs, the maximum calculated containment pressure during and following a design basis accident is below the containment design pressure of 15 psig. Similarly the drywell pressure remains below the design pressure of 30 psig.

The 18'-6" and 17'-9.5" maximum and minimum water levels in OPERATIONAL CONDITIONS 1, 2 and 3 (corresponding to the HWL water volume of 118,548 cubic feet and the LWL water volume of 113,675 cubic feet respectively), are nominal values assuming a differential pressure of zero across the drywell wall. The minimum allowed water level of 16'-6" in OPERATIONAL CONDITIONS 4 and 5 (see Specification 3.5.3) is also based on an assumed differential pressure of zero across the drywell wall. These values include the water volume of the containment portion of the pool, the horizontal vents, and the weir annulus (including encroachments).

DRYWELL AND CONTAINMENT PURGE SYSTEM (Continued)

Leakage integrity tests with a maximum allowable leakage rate for purge supply and exhaust isolation valves will provide early indication of resilient material seal degradation and will allow the opportunity for repair before gross leakage failure develops. The 0.60 L_a leakage limit shall not be exceeded when the leakage rates determined by the leakage integrity tests of these valves are added to the previously determined total for all valves and penetrations subject to Type B and C tests.

3/4.6.1.9 FEEDWATER LEAKAGE CONTROL SYSTEM

The OPERABILITY of the feedwater leakage control system is required to meet the restrictions on overall containment leak rate assumed in the accident analyses.

3/4.6.2 DRYWELL

3/4.6.2.1 DRYWELL INTEGRITY

Drywell integrity ensures that the steam released for the full spectrum of drywell pipe breaks is condensed inside the primary containment either by the suppression pool or by containment spray. By utilizing the suppression pool as a heat sink, energy released to the containment is minimized and the severity of the transient is reduced.

3/4.6.2.2 DRYWELL BYPASS LEAKAGE

The limitation on drywell bypass leakage rate is based on having containment spray OPERABLE. It ensures that the maximum leakage which could bypass the suppression pool during an accident would not result in the containment exceeding its design pressure of 15.0 psig. The integrated drywell leakage value is limited to 10% of the design drywell leakage rate.

The limiting case accident is a very small reactor coolant system break which will not automatically result in a reactor depressurization. The long term differential pressure created between the drywell and containment will result in a significant pressure buildup in the containment due to this bypass leakage.

3/4.6.2.3 DRYWELL AIR LOCK

The limitations on closure for the drywell air lock is required to meet the restrictions on DRYWELL INTEGRITY and the drywell leakage rate given in Specifications 3.6.2.1 and 3.6.2.2. The specification makes allowances for the fact that there may be long periods of time when the air lock will be in a closed and secured position during reactor operation. Only one closed door in the air lock is required to maintain the integrity of the drywell.

The air supply to the drywell air lock and seal system is the service and instrument air system. This system consists of two 100% capacity air compressors per unit and can be cross-connected. This system is redundant and extremely reliable and provides system pressure indication in the control room.

DEPRESSURIZATION SYSTEMS (Continued)

The suppression pool volume used in the short-term containment response analyses (119,348 cubic feet), corresponds to the suppression pool HWL of 18'-6", and includes the effects of the maximum negative drywell-to-containment differential pressure (-0.5 psid) and primary containment to secondary containment differential pressure (1.0 psid) [which are the respective Technical Specification limits], on the water volume to maximize the drywell pressure and temperature responses following design basis loss of coolant accidents (and transient events) from the analysis power level of 104.2% of rated thermal power.

The suppression pool volume used in the long-term containment response analyses (142,772 cubic feet, which includes the makeup volume from the upper pool of 32,573 cubic feet), corresponds to a suppression pool low water level of 17'-6", and includes the effects of the maximum positive drywell-to-containment differential pressure (2.0 psid - which is the Technical Specification limit). This volume was utilized in the long-term containment response analyses to maximize the containment pressure and temperature responses following design basis loss of coolant accidents (and transient events) from the analysis power level of 104.2% of rated thermal power. Note that both the short-term and long-term analyses were performed at a power level 2.2% higher than the licensing requirement of 102% of rated thermal power for additional conservatism.

The LCO limit on the minimum suppression pool water level was set at 17'-9.5" (at a zero drywell-to-containment differential pressure) in order to satisfy the analysis for maximum drawdown of the suppression pool. In order to account for positive drywell-to-containment differential pressure, the LCO requires the use of a Suppression Pool Level Adjustment Graph. The Suppression Pool Level Adjustment Graph is contained in the Plant Data Book, and it plots the pool level adjustment factor versus the drywell-to-containment differential pressure. This graph is used to modify the nominal minimum suppression pool water level of 17'-9.5" to account for the effect of a positive differential pressure across the drywell wall on the suppression pool water level (volume). Negative pressure differentials were directly accounted for in the short-term analyses and therefore do not need to be adjusted for by the operator.

The suppression pool levels (depths) satisfy criteria or constraints imposed by: (1) maintaining a 2 foot minimum post-LOCA horizontal vent coverage to assure steam condensation/pressure suppression, and to maintain coverage over the RHR A Test Return Line, (2) adequate ECCS pump NPSH, (3) adequate depth for vortex prevention, (4) adequate depth for minimum recirculation volume, and (5) minimizing hydrodynamic loads on submerged structures during SRV and horizontal vent steam discharges.

DEPRESSURIZATION SYSTEMS (Continued)

The suppression pool temperature limits are based on the following:

1. 95°F is an initial condition for the containment response analysis for demonstrating the adequacy to satisfy the post-LOCA long-term peak suppression pool temperature limit of 185°F.

2. 120°F is analytically based and is derived to satisfy the 170°F post-LOCA blowdown peak suppression pool temperature assuming a LOCA when the reactor is isolated.

3. 110°F and 105°F are derived from the analytically based 95°F and 120°F values using engineering judgment; considering operator response time, reactor pressure vessel energy, and pool heat capacity to meet the 170°F limit, and also to avoid unnecessary scrams and/or depressurizations.

Testing in the Mark III Pressure Suppression Test Facility and analysis have assured that the suppression pool temperature will not rise above 185°F for the full range of break sizes.

Should it be necessary to make the suppression pool inoperable, this shall only be done as specified in Specification 3.5.3.

Experimental data indicates that effective steam condensation without excessive load on the containment pool walls will occur with a quencher device and pool temperature below 200°F during relief valve operation. Specifications have been placed on the envelope of reactor operating conditions to assure the bulk pool temperature does not rise above 185°F in compliance with the containment structural design criteria.

In addition to the limits on temperature of the suppression pool water, operating procedures define the action to be taken in the event a safety-relief valve inadvertently opens or sticks open. As a minimum this action shall include: (1) use of all available means to close the valve, (2) initiate suppression pool water cooling, and (3) if other safety-relief valves are used to depressurize the reactor, their discharge shall be separated from that of the stuck-open safety relief valve, where possible, to assure mixing and uniformity of energy insertion to the pool.

The containment spray system consists of two 100% capacity loops, each with three spray rings located at different elevations about the inside circumference of the containment. RHR pump A supplies one loop and RHR pump B supplies the other. RHR pump C cannot supply the spray system. Dispersion of the flow of water is effected by 346 nozzles in loop A and 344 nozzles in loop B, enhancing the condensation of water vapor in the containment volume and preventing overpressurization. Heat rejection is through the RHR heat exchangers. The turbulence caused by the spray system aids in mixing the containment air volume to maintain a homogeneous mixture for H_2 control.

DEPRESSURIZATION SYSTEMS (Continued)

The suppression pool cooling function is a mode of the RHR system and functions as part of the containment heat removal system. The purpose of the system is to ensure containment integrity following a LOCA by preventing excessive containment pressures and temperatures. The suppression pool cooling mode is designed to limit the long term bulk temperature of the pool to 185°F considering all of the post-LOCA energy additions. The suppression pool cooling trains, being an integral part of the RHR system, are redundant, safety-related component systems that are initiated following the recovery of the reactor vessel water level by ECCS flows from the RHR system. Heat rejection to the emergency service water is accomplished in the RHR heat exchangers.

The suppression pool make-up system provides water from the upper containment pool to the suppression pool by gravity flow through two 100% capacity dump lines following a LOCA. The quantity of water provided is sufficient to account for all conceivable post-accident entrapment volumes, ensuring the long term energy sink capabilities of the suppression pool and maintaining the water coverage over the uppermost drywell vents. During refueling, there will be administrative control to ensure the make-up dump valves will not be opened.

The upper containment pool water level may be reduced (for example, for maintenance of the inclined fuel transfer system), provided the minimum required suppression pool level (volume) is raised to compensate. Raising the minimum required suppression pool water level provides the same effective volume of water (by transferring a portion of the upper pool dump volume to the suppression pool) and ensures that after a suppression pool make-up system dump, adequate water coverage over the uppermost drywell horizontal vents and the long-term energy sink capability of the suppression pool is maintained.

3/4.6.4 CONTAINMENT ISOLATION VALVES

The OPERABILITY of the containment isolation valves ensures that the containment atmosphere will be isolated from the outside environment in the event of a release of radioactive material to the containment atmosphere or pressurization of the containment and is consistent with the requirements of GDC 54 through 57 of Appendix A to 10 CFR 50. Containment isolation within the time limits specified for those isolation valves designed to close automatically ensures that the release of radioactive material to the environment will be consistent with the assumptions used in the analyses for a LOCA.

All required Containment Isolation Valves are listed in the PNPP Unit 1 Plant Data Book. The opening of normally locked or sealed closed containment isolation valves under administrative controls in accordance with footnote # includes the following considerations: (1) stationing an operator, who is in constant communication with the control room, at the valve controls, (2) instructing this operator to close these valves in an accident situation, and (3) assuring that environmental conditions will not preclude access to close the valves and that this action will prevent the release of radioactivity outside the containment. The above considerations do not apply to the normally locked closed (LC) Fire Protection system manual hose reel containment isolation valves 1P54-F726 and -F727 when opened as necessary to supply fire mains when handling irradiated fuel in the primary containment, during CORE ALTERATIONS and operations with a potential for draining the reactor vessel.

3/4.6.5 VACUUM RELIEF

3/4.6.5.1 CONTAINMENT VACUUM RELIEF AND 3/4.6.5.2 CONTAINMENT HUMIDITY CONTROL

Vacuum breakers are provided on the containment to prevent an excessive vacuum from developing inside containment during an inadvertent or improper operation of the containment spray. Four vacuum breakers and their associated isolation valves are provided. Any two vacuum breakers provide 100% vacuum relief.

The containment vacuum relief system is designed to prevent an excessive vacuum from being created inside the containment following inadvertent initiation of the containment spray system. By maintaining temperature/relative humidity within the limits for acceptable operation shown on Figure 3.6.5.2-1, the maximum containment vacuum created by actuation of both containment spray loops will be limited to approximately -0.7 psig.

3/4.6.5.3 DRYWELL VACUUM BREAKERS

Drywell vacuum breakers are provided on the drywell to prevent drywell flooding due to differential pressure across the drywell and to equalize pressure between the drywell and containment.

Two drywell vacuum breakers and their associated isolation valves are provided. Any one vacuum breaker can provide full vacuum relief capability.

3/4.6.6 SECONDARY CONTAINMENT

Secondary containment is designed to minimize any ground level release of radioactive material which may result from an accident. The Shield Building provides secondary containment during normal operation when the containment is sealed and in service. At other times, the containment may be open and, when required, secondary containment integrity is specified.

Establishing and maintaining a vacuum in the annulus with the annulus exhaust gas treatment system, along with the surveillance of the doors, hatches, and valves, is adequate to ensure that there are no violations of the integrity of the secondary containment.

The OPERABILITY of the annulus exhaust gas treatment systems ensures that sufficient iodine removal capability will be available in the event of a LOCA. The reduction in containment iodine inventory reduces the resulting site

3/4.9.7 CRANE TRAVEL - SPENT FUEL STORAGE POOL, NEW FUEL STORAGE VAULTS, AND UPPER CONTAINMENT FUEL POOL

The restriction on movement of loads which would result in excess of 4000 foot-pounds of impact energy if dropped over fuel assemblies in the pools ensures that in the event this load is dropped 1) the activity release will be less than that assumed in the safety analysis, and 2) any possible distortion of fuel in the storage racks will not result in a critical array. This assumption is consistent with the activity release assumed in the safety analyses.

3/4.9.8 and 3/4.9.9 WATER LEVEL - REACTOR VESSEL AND WATER LEVEL - SPENT FUEL STORAGE AND UPPER CONTAINMENT FUEL POOLS

The restrictions on minimum water level ensure that sufficient water depth is available to remove 99% of the assumed 10% iodine gap activity released from the rupture of an irradiated fuel assembly. This minimum water depth is consistent with the assumptions of the accident analysis.

3/4.9.10 CONTROL ROD REMOVAL

These specifications ensure that maintenance or repair of control rods or control rod drives will be performed under conditions that limit the probability of inadvertent criticality. The requirements for simultaneous removal of more than one control rod are more stringent since the SHUTDOWN MARGIN specification provides for the core to remain subcritical with only one control rod fully withdrawn.

3/4.9.11 RESIDUAL HEAT REMOVAL AND COOLANT CIRCULATION

The requirement that at least one residual heat removal loop be OPERABLE and in operation or that an alternate method capable of decay heat removal be demonstrated and that an alternate method of coolant mixing be in operation ensures that 1) sufficient cooling capacity is available to remove decay heat and maintain the water in the reactor pressure vessel below 140°F as required during REFUELING, and 2) sufficient coolant circulation would be available through the reactor core to assure accurate temperature indication and to distribute and prevent stratification of the poison in the event it becomes necessary to actuate the standby liquid control system.

The requirement to have two shutdown cooling mode loops OPERABLE when there is less than 22 feet 9 inches feet of water above the reactor vessel flange ensures that a single failure of the operating loop will not result in a complete loss of residual heat removal capability. With the reactor vessel head removed and 22 feet 9 inches of water above the reactor vessel flange, a large heat sink is available for core cooling. Thus, in the event a failure of the operating RHR loop, adequate time is provided to initiate alternate methods capable of decay heat removal or emergency procedures to cool the core.

3/4.9.12 INCLINED FUEL TRANSFER SYSTEM

The purpose of the inclined fuel transfer system specification is to control personnel access to those potentially high radiation areas immediately adjacent to the system and to assure safe operation of the system.

3/4.9.1 REACTOR MODE SWITCH

Locking the OPERABLE reactor mode switch in the Shutdown or Refuel position, as specified, ensures that the restrictions on control rod withdrawal and refueling platform movement during the refueling operations are properly activated. These conditions reinforce the refueling procedures and reduce the probability of inadvertent criticality, damage to reactor internals or fuel assemblies, and exposure of personnel to excessive radioactivity.

3/4.9.2 INSTRUMENTATION

The OPERABILITY of at least two source range monitors ensures that redundant monitoring capability is available to detect changes in the reactivity condition of the core.

3/4.9.3 CONTROL ROD POSITION

The requirement that all control rods be inserted during CORE ALTERATIONS ensures that fuel will not be loaded into a cell without a control rod, athough one rod may be withdrawn under the control of the reactor mode switch refuel position one-rod-out interlock.

3/4.9.4 DECAY TIME

The minimum requirement for reactor subcriticality prior to fuel movement ensures that sufficient time has elapsed to allow the radioactive decay of the short lived fission products. This decay time is consistent with the assumptions used in the accident analyses.

3/4.9.5 COMMUNICATIONS

The requirement for communications capability ensures that refueling station personnel can be promptly informed of significant changes in the facility status or core reactivity condition during movement of fuel within the reactor pressure vessel.

3/4.9.6 REFUELING PLATFORM

The OPERABILITY requirements ensure that (1) the refueling platform will be used for handling control rods and fuel assemblies within the reactor pressure vessel, (2) each crane and hoist has sufficient load capacity for handling fuel assemblies and control rods, and (3) the core internals and pressure vessel are protected from excessive lifting force in the event they are inadvertently engaged during lifting operations.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 57 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 June 24, 1992, as supplemented by letter dated September 25, 1992, the Cleveland Electric Illuminating Company, et al. (the licensee) submitted a request to amend the Technical Specifications (TSs) for the Perry Nuclear Power Plant, Unit No. 1. In particular, the licensee requested revision to a number of primary containment pressure and temperature limits and the suppression pool water level limit based on a revised containment response analysis in order to address the problem of recurring high ambient and lake temperatures during the summer months. The supplemental letter did not affect the notice of Opportunity for Hearing published in the Federal Register on July 1, 1992 (57 FR 29337). By letter dated June 21, 1993, the licensee requested partial issuance of the amendment for the summer of 1993; however, the NRC staff did not act on that request.

By letter dated November 16, 1992, the licensee submitted a request to amend the TSs to permit a reduction in the water level of the upper containment pool water level during plant operations, provided that the suppression pool water level was increased in order to compensate. Due to the similarity of the subject matter, the staff decided to review this amendment request in conjunction with the above mentioned amendment request.

2.0 EVALUATION

The licensee's amendment request is based upon a containment response analysis performed by GE Nuclear Energy (GE), as well as structural design and operational impact reviews performed by Gilbert Associates, Inc. (GAI), and the licensee. The containment analysis comprises the bulk of the licensee's technical justification, and is discussed in detail below.

2.1 CONTAINMENT ANALYSIS

Primary containment temperature and pressure response following a postulated loss-of-coolant accident (LOCA) is of great importance when determining the potential for offsite release of radioactive material, in determining Emergency Core Cooling System (ECCS) pump net positive suction head (NPSH) requirements, and in determining environmental qualification requirements for safety-related equipment located inside the primary containment. As part of the generic BWR power uprate program, GE proposed to update the calculational methods used for determining peak containment temperatures and pressures following a postulated LOCA. In particular, GE proposed to utilize the SHEX

computer code when calculating the peak pressures and temperatures during the long-term portion of containment post-LOCA response, in place of the previously used M3CPT/HXSIZ combination. The staff agreed with the use of SHEX in support of the generic BWR power uprate program; however, at that time the staff did not support the use of SHEX for other uses. In a July 13, 1993, letter from A. Thadani (NRC) to G. Sozzi (GE), the staff expanded the acceptable uses of SHEX, and stated that although SHEX had not been approved by the NRC staff for generic use, the use of SHEX on a plant-specific basis would be permitted. The use of SHEX in the evaluation of long-term containment response is not currently part of the licensing basis at Perry.

The licensee's June 24, 1992, amendment request contained a GE Topical Report entitled, "Perry Technical Specifications Improvement - Containment Response Analysis" (NEDC-31940). This report contains the results of revised containment pressure and temperature analysis performed to evaluate both short-term and long-term response of the containment to postulated LOCAs. GE used the M3CPT computer model to perform the short-term analysis, and the SHEX computer model to perform the long-term analysis.

2.1.1 Input Assumptions

In performing the containment response analysis, GE proposed a number of revised TS limits which would provide greater operational flexibility for the licensee. A comparison of the current limits to the analyzed limits is included in Table 1. (The licensee subsequently revised several of these proposed limits, based upon structural and equipment qualification evaluations performed by GAI and the licensee.) The staff does not object to the use of these limits in the containment response evaluation.

Table 1

COMPARISON OF CURRENT AND ANALYZED TECHNICAL SPECIFICATION LIMITS

Description	Current Limit	Analyzed Limit
Containment Ambient Temperature	90°F	104°F
Suppression Pool Temperature	90°F	95°F
Suppression Pool Lower Water Level (Depth)	18'0"	17'6"
Minimum Upper Pool Depth (above reactor vessel flange)	22'10"	22'9"

Other input assumptions were revised in order to make the results of the analysis more conservative. These revised input assumptions are listed in Table 2.

Table 2

REVISED INPUT ASSUMPTIONS USED IN CONTAINMENT RESPONSE ANALYSIS

Description	FSAR Value	Revised Value
COMMON ASSUMPTIONS		
Reactor Power (MWt)	3650	3729
Initial Suppression Pool Temperature	90°F	95°F
SHORT-TERM ANALYSIS		
Initial drywell/containment differential pressure (psid)	0.0	-0.5
Initial primary/secondary containment differential pressure (psid)	0.0	+1.0
LONG-TERM ANALYSIS		
Initial suppression pool water level	18'0"	17'6"
Initial drywell/containment differential pressure (psid)	0.0	+2.0
Upper pool dump to suppression pool	1800 sec	1800 sec or Rx low-low level
Temperature of upper pool water	100°F	110°F
ESW temperature at inlet to RHR heat exchanger	80°F	85°F

2.1.2 Short-Term Response

When evaluating containment post-LOCA response, the M3CPT code is used to calculate short-term containment temperature and pressure response following a postulated LOCA, while either SHEX or a combination of M3CPT and HXSIZ would

be used to determine the long-term suppression pool temperature. The M3CPT code uses a mechanistic method to model the highly transient conditions in the containment immediately following a LOCA, and is capable of modelling containment long-term response, up to the initiation of containment cooling. M3CPT has been verified against experimental data and has been previously approved by the NRC staff.

Short-term containment response is primarily affected by initial drywell and wetwell pressures and the suppression pool water level and temperature. By assuming a negative initial drywell to wetwell differential pressure, the analysis took into account the reduction of drywell free-air space due to an increase in the amount of water in the weir area of the drywell. Decreasing the free-air volume tends to increase the peak drywell pressure calculated by the short-term analysis. An increase in the water level in the weir area can be caused by either a negative drywell to wetwell differential pressure or an increase in overall suppression pool level above normal. GE performed a sensitivity study to ensure that the -0.5 psid drywell to wetwell differential pressure bounded all other normal suppression pool level variations.

The short-term containment response analysis yielded results, listed in Table 3, which are similar to those obtained from the Final Safety Analysis Report (FSAR). The main steam line break within the drywell dominated the short-term analysis, predicting higher peak pressures than the recirculation suction line break. The short-term containment hydrodynamic loads due to LOCA bubble and pool swell loadings did not change significantly from the FSAR. The peak pressures calculated by the short-term analysis fall well below the containment design limits, and are therefore acceptable.

Table 3

SHORT-TERM CONTAINMENT RESPONSE

	Revised	FSAR	Design Limit
Peak drywell pressure (psig)	21.8	22.1	30
Peak drywell/containment differential pressure (psid)	20.4	21.2	30
Peak wetwell pressure (psig)	11.2	9.2	15

2.1.3 Long-Term Response

During the 1970's and early 1980's, GE used the M3CPT/HXSIZ combination to model the long-term response of the containment to a spectrum of LOCAs. The M3CPT code was used to model both the short-term and long-term response to the LOCA from the time of the breakup to the time of initiation of containment

cooling. After initiation of containment cooling, the HXSIZ code was used to model the containment heat exchangers, using input values obtained from M3CPT. By modelling the containment heat exchangers, the suppression pool temperature could be calculated as a function of time.

The SHEX code utilizes more refined models than those used by M3CPT/HXSIZ to determine suppression pool temperature, and is capable of modelling containment responses to more accident scenarios than the HXSIZ code. Many of the models used in SHEX are the same as, or very similar to, those used in M3CPT. SHEX is also capable of modelling all containment auxiliary systems, permitting a more accurate analysis of actual containment conditions following a postulated LOCA.

Containment long-term response is affected by suppression pool volume, initial temperature, and heat-exchanger efficiency, all of which affect the ability of the containment to absorb the heat loads caused by the proposed LOCA. In the long-term containment response analysis performed by GE, the suppression pool volume was decreased (due to an assumed drywell to wetwell differential pressure and lower overall suppression pool and upper containment pool water levels), the initial pool temperature was increased, and the efficiency of the heat exchangers was decreased (by increasing the temperature of the service water at the inlet to the heat exchangers). Decreasing the ability to remove heat from the containment increases the peak pressures and temperatures calculated by the analyses. GE performed a sensitivity analysis of the effect of suppression pool levels on the long-term containment response analysis in order to determine revised high and low water level limits for the suppression pool.

GE, on behalf of the licensee, analyzed a spectrum of LOCA break sizes and events to determine the impact on the long-term containment response. Results of the long-term containment response analysis are summarized in Table 4. Several different accident scenarios were used to define the limiting peak pressures and temperatures; however, results of all analyses remained within the design basis of the containment structure, and are, therefore, acceptable.

2.2 PROPOSED CHANGES TO TECHNICAL SPECIFICATIONS

As a result of the revised containment response analysis, the peak pressure expected to be experienced by the containment has been reduced from 11.31 psig to 7.8 psig. The licensee has proposed to change all references to this peak pressure to the new value, permitting leakage testing of the containment to be performed from a lower initial pressure. Specifically, the licensee has changed references to the peak pressure contained in TSs 4.6.1.1.1, 3.6.1.2, 4.6.1.2, 3.6.1.3, and 4.6.1.3, and the Bases for TS 3/4.6.1.2, 3/4.6.1.6, and 3/4.6.2.5. These changes are consistent with the results of the containment analyses and are therefore acceptable.

The licensee also proposed several changes to the minimum suppression pool and upper containment pool water levels. Specifically, the licensee proposed to change the minimum suppression pool water level described in TS 3.6.3.1 from the present value of 18'0" to 17'9.5" plus a "level adjustment factor." All

Table 4

LONG-TERM CONTAINMENT RESPONSE

	Event	Value	FSAR	Limit
Peak drywell temperature	MSLB	328.7°F	330.0°F	330.0°F
Peak suppression pool temperature	ASD-A	184.7°F	184.6°F	185.0°F
Peak containment pressure (psig)	MSLB	7.8	11.3	15.0
Peak containment temperature	LHS	160.5°F	184.6°F	185.0°F

MSLB -- Main steam line break inside drywell
ASD-A -- Alternate shutdown event A
LHS -- LOCA from hot standby conditions

other explicit references to minimum suppression pool water level have been removed and replaced with a reference to TS 3.6.3.1. The level adjustment factor was developed by the licensee to account for changes in suppression pool volume caused by drywell to wetwell differential pressure. Using the level adjustment factor (which is always zero or positive), the suppression pool water level (and overall suppression pool water volume) would always be maintained above that assumed in the containment long-term response analysis. The licensee stated that although the graph defining the suppression pool adjustment factor is not contained in the TSs, it would be maintained in accordance with other plant procedures, and any changes would be subject to Plant Operations Review Committee (PORC) review. The staff finds this proposed change to be acceptable.

The licensee has proposed to reduce the minimum water level in the upper containment pool from the present limit of 22'10" above the reactor vessel flange to 22'5" above the flange, provided that the suppression pool level is increased by 2.20" to compensate. Since the net volume of the suppression pool (after makeup from the upper pool) will remain the same, the containment long-term analysis will remain valid. Therefore, this change is acceptable.

In addition, the licensee proposed to revise the suppression pool and containment air temperatures described in TSs 3.6.1.7, 3.6.3.1, and 4.6.3.1 from 90°F to 95°F. The staff finds these changes to be acceptable.

The staff has concluded that the containment temperature and pressure response following a postulated LOCA will remain acceptable after implementation of the proposed changes. The staff also concludes that the containment will continue

to meet the requirements for sufficient margin from temperature and pressure limits as described in 10 CFR Part 50, Appendix A, General Design Criterion 50, "Containment design basis." The staff, therefore, considers the proposed changes to the TSs of the Perry Nuclear Power Plant, Unit No. 1, as proposed by the licensee, 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 PUBLIC COMMENTS

By letter dated August 17, 1992, comments regarding the licensee's June 24, 1992, amendment request were received from the Ohio Citizens for Responsible Energy, Inc. (OCRE). The comments focused on two principal issues regarding the licensee's amendment request: (1) the authors assert that the impact of accidents beyond those in the plant design basis have not been addressed by the licensee in the amendment request, and (2) the change in methodology used in the containment analysis completely masks the effect of the proposed TS changes. The staff does not consider these comments to have technical merit for the following reasons:

With respect to OCRE's first concern regarding the licensee's failure to perform an analysis of the impact of the amendment request on those accidents which are beyond the design basis, 10 CFR Part 50 does not require that requests to amend plant TSs address accidents which go beyond the design basis of the plant. The purpose of the Individual Plant Examination (IPE) program was to specifically address the effect of accidents beyond the design basis on plant equipment, and to highlight risk-significant areas for improvement. The NRC intends that the IPE will be maintained by the licensee as a living document, and that plant modifications, including those done under 10 CFR Part 50.59, will be periodically incorporated into the PRA models developed for the plant. However, the concept of risk-significance is specifically not used in the technical review of plant license amendments. Instead, the staff relies on existing NRC rules, as contained in Title 10 of the Code of Federal Regulations, Regulatory Guides, and NRC generic correspondence, as well as guidance provided in the Standard Review Plan (NUREG-0800) when evaluating the acceptability of proposed license amendments. The analyses presented by the licensee in the abovementioned submittals is technically sound, and the proposed TSs do not violate these existing NRC requirements; therefore, the staff has found them to be acceptable.

OCRE's second concern was that the use of a different methodology for the containment analyses effectively masks the effects of the various changes to input assumptions. Direct comparison of the results of the new analyses against the FSAR analyses, even using the same input assumptions, would not have yielded any useful information. One would expect that the "more realistic" methodology used by SHEX would result in lower peak temperatures and pressures than the preceding analysis. Additionally, "taking credit" for

previously unrecognized margin is acceptable to the staff, if done in a cautious manner. The staff has reviewed the input assumptions used in the analyses and has compared these to the input assumptions used in the original FSAR analysis. The licensee revised several input assumptions from the original FSAR analysis to reflect actual plant conditions; since these assumptions tended to produce less favorable results from the analyses, the staff considers this to be an added conservatism from the original analysis. Other assumptions were changed to allow for more flexible operation of the plant; these changes (and the associated changes to the TSs) could prevent unnecessary plant shutdowns and challenges to plant safety equipment. Thus, the staff has no reason to find the proposed changes unacceptable.

The staff compared the results of the new analyses to those obtained using the FSAR methodology for consistency. Unlike calculations performed to show compliance with 10 CFR Part 50, Appendix K, containment response analyses do not need to be performed to a prescribed NRC methodology, using an NRC approved computer code. Although the NRC has not explicitly reviewed the SHEX computer code, the staff did undertake detailed review of the preceding codes M3CPT and HXSIZ, and ultimately granted NRC approval of these codes. The staff has reviewed all aspects of the containment analyses, including the input assumptions, methodologies, and results, compared these against those contained in the FSAR and in applicable NRC regulations, and has found the containment analyses submitted by the licensee to be acceptable.

5.0 ENVIRONMENTAL CONSIDERATION

Pursuant to 10 CFR 51.21, 51.32, and 51.35, an environmental assessment and finding of no significant impact has been prepared and published in the Federal Register on March 15, 1994 (59 FR 12013). Accordingly, based upon the environmental assessment, the Commission has determined that the issuance of this amendment will not have a significant effect on the quality of the human environment.

6.0 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.

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