Docket Files

MARCH 27 1979

Docket No. 50-334

Mr. C. N. Dunn, Vice President
Operations Division
Duquesne Light Company
435 Sixth Avenue
Pittsburgh, Pennsylvania 15219

Dear Mr. Dunn:

The Commission has issued the enclosed Amendment No. 17 to Facility Operating License No. DPR-66 for the Beaver Valley Power Station Unit No. 1. The amendment consists of changes in the Technical Specifications in response to your letter dated October 10, 1977 and December 22, 1977 (in response to our letter dated October 20, 1977) on pressurizer heatup rate, your letter dated March 20, 1978 on alignment of the low head safety injection pump, your letter dated May 23, 1978 on accumulator levels and pressures, your letter dated May 2, 1977 on the radiation protection manager and your letter dated October 27, 1978, supplemented by your letter dated December 14, 1978 which included the axial power distribution monitoring system, part length rods, and the augmented startup program. The remaining proposed changes in your October 10, 1977 and October 27, 1978 letter are still under review.

The changes include (1) correction of the pressurizer heat-up rate error; (2) documentation of the one time change based on your March 20, 1978 request, which granted extra time to realign the 1A Low Head Safety Injection Pump; (3) new accumulator water level and nitrogen pressure limits to properly account for instrument inaccuracies in the ECCS analysis; (4) clarification of requirements for a radiation protection manager; (5) deletion of a surveillance requirement (4.2.2.2, item g) to be consistent with the Amendment 9 deletion of other requirements for the axial power distribution monitoring system (APDMS); (6) correction of typographical errors to Table 3.7-4 on snubbers; (7) elimination of part length rod requirement (these rods will be removed from the core), and (8) deletion of reference to the completed Augmented Startup Test Program.

DP 1

7904260042 60

OFFICE						
SURNAME						
DATE						

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Original Signed By

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment No. 17 to DPR-66
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
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TACS 7830
7627
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See clipped change.

AB 3/27/79

OFFICE	DOR:ORB-1	DOR:ORB-1	OELD	DOR:STS	DOR:ORB-1	DOR:ORB-1/SED
SURNAME	CParrish	DWigginton:esp	S. Soltinski	DBrinkman	ASchwencer	R. Vollmer
DATE	3/13/79	3/17/79	3/26/79	3/20/79	3/27/79	3/23/79

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

March 27, 1979

Docket No. 50-334

Mr. C. N. Dunn, Vice President
Operations Division
Duquesne Light Company
435 Sixth Avenue
Pittsburgh, Pennsylvania 15219

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The changes include (1) correction of the pressurizer heat-up rate error; (2) documentation of the one time change based on your March 20, 1978 request, which granted extra time to realign the 1A Low Head Safety Injection Pump; (3) new accumulator water level and nitrogen pressure limits to properly account for instrument inaccuracies in the ECCS analysis; (4) clarification of requirements for a radiation protection manager; (5) deletion of a surveillance requirement (4.2.2.2, item g) to be consistent with the Amendment 9 deletion of other requirements for the axial power distribution monitoring system (APDMS); (6) correction of typographical errors to Table 3.7-4 on snubbers; (7) elimination of part length rod requirement (these rods will be removed from the core), and (8) deletion of reference to the completed Augmented Startup Test Program.

7904260042

Copies of the Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Enclosures:
1. Amendment No. 17 to DPR-66
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
See next page

cc
Gerald Charnoff, Esquire
Jay E. Silberg, Esquire
Shaw, Pittman, Potts & Trowbridge
1800 M Street, N.W.
Washington, D.C. 20036

Karin Carter, Esquire
Special Assistant Attorney General
Bureau of Administrative Enforcement
5th Floor, Executive House
Harrisburg, Pennsylvania 17120

Marvin Fein
Utility Counsel
City of Pittsburgh
313 City-County Building
Pittsburgh, Pennsylvania 15219

Mr. J. M. Cumiskey
Stone & Webster Engineering
Corporation
P. O. Box 2325
Boston, Massachusetts 02107

Mr. J. D. Woodward
R&D Center
Westinghouse Electric Corporation
Building 7-303
Pittsburgh, Pennsylvania 15230

Department of Environmental Resources
ATTN: Director, Office of
Radiological Health
Post Office Box 2063
Harrisburgh, Pennsylvania 17105

Mr. Thomas J. Czerpah
Mayor of the Burrough of Shippingport
P. O. Box 26
Shippingport, Pennsylvania 15077

Mr. Jack Carey
Technical Assistant
Duquesne Light Company
P. O. Box 4
Shippingport, Pennsylvania 15077

Ohio Edison Company
c/o Chief Nuclear QA Engineer
76 South Main Street
Akron, Ohio 44308

Pennsylvania Power Company
Ray E. Semmler, President
One E Washington Street
New Castle, Pennsylvania 16103

John W. Cashman, M.D.
Director of Health
450 East Town Street
Columbus, Ohio 43216

Ohio Environmental Protection Agency
Division of Planning
Environmental Assessment Section
P. O. Box 1049
Columbus, Ohio 43216

Office of the Governor
State of West Virginia
Charleston, West Virginia 25305

Mr. Carl Frasure
Committee of State Officials on
Suggested State Legislation
Department of Political Science
West Virginia University
Morgantown, West Virginia 26505

Mr. Joseph H. Mills, Acting Commissioner
State of West Virginia Department of
Labor
1900 Washington Street
East Charleston, West Virginia 25305

Mr. R. E. Martin
Duquesne Light Company
435 Sixth Avenue
Pittsburgh, Pennsylvania 15219

B. F. Jones Memorial Library
663 Franklin Avenue
Aliquippa, Pennsylvania 15001

cc
N. H. Dyer, M. D.
State Director of Health
State Department of Health
State Office Building No. 1
1800 Washington Street, East
Charleston, West Virginia 25305

Director, Technical Assessment Division
Office of Radiation Programs (AW-459)
U. S. Environmental Protection Agency
Crystal Mall #2
Arlington, Virginia 20460

U. S. Environmental Protection
Agency
Region III Office
ATTN: EIS COORDINATOR
Curtis Building (Sixth Floor)
Philadelphia, Pennsylvania 19106

Mr. James A. Werling
Plant Superintendent
Beaver Valley Power Station
P. O. Box 4
Shippingport, Pennsylvania 15077

Governor's Office of State Planning
and Development
ATTN: Coordinator, Pennsylvania
State Clearinghouse
P. O. Box 1323
Harrisburg, Pennsylvania 17120

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

DUQUESNE LIGHT COMPANY

OHIO EDISON COMPANY

PENNSYLVANIA POWER COMPANY

DOCKET NO. 50-334

BEAVER VALLEY POWER STATION UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 17
License No. DPR-66

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

   A. The applications for amendment by Duquesne Light Company, filed on behalf of itself, Ohio Edison Company, and Pennsylvania Power Company (the licensees) dated October 10, 1977 and December 22, 1977 (in response to the Commission's letter dated October 20, 1977), March 20, 1978, May 23, 1978, May 2, 1977, October 27, 1978 and December 14, 1978, 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 applications, 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.

7904260046

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-66 is hereby amended to read as follows:

   "(2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 17, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications."

3. This license amendment is effective as of the date of its issuance, except for the Technical Specification change relating to the one-time extension in outage time for the 1A Low Head Safety Injection Pump which was effective on March 20, 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: March 27, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. ___

FACILITY OPERATING LICENSE NO. DPR-66

DOCKET NO. 50-334

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages listed in Group A below upon receipt of this License Amendment. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. The corresponding overleaf pages are also provided to maintain document completeness.

Group A

Remove Pages	Insert Pages
3/4 2-1	3/4 2-1
3/4 2-6a	3/4 2-6a
3/4 4-27	3/4 4-27
3/4 5-1	3/4 5-1
3/4 5-2	3/4 5-2
3/4 5-3	3/4 5-3
3/4 7-30	3/4 7-30
6-5	6-5

Replace the following pages of the Appendix "A" Technical Specifications with the enclosed pages listed in Group B below upon completion of the removal of the part length rods from the core. The revised pages are identified by Amendment number and contain vertical lines indicating the area of change. Corresponding overleaf pages are also provided to maintain document completeness.

Group B

Remove Pages	Insert Pages
III	III
1-3	1-3
3/4 1-18	3/4 1-18
3/4 1-19	3/4 1-19
3/4 1-20	3/4 1-20
3/4 1-26	3/4 1-26
3/4 2-3	3/4 2-3
3/4 2-10	3/4 2-10
3/4 2-11	3/4 2-11
3/4 9-1	3/4 9-1
3/4 10-1	3/4 10-1
3/4 10-2	3/4 10-2
3/4 10-6	3/4 10-6
B 3/4 1-4	B 3/4 1-4
B 3/4 2-1	B 3/4 2-1
B 3/4 2-4	B 3/4 2-4
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CHANNEL FUNCTIONAL TEST

1.11 A CHANNEL FUNCTIONAL TEST shall be the injection of a simulated signal into the channel as close to the primary sensor as practicable to verify OPERABILITY including alarm and/or trip functions.

CORE ALTERATION

1.12 CORE ALTERATION shall be the movement or manipulation of any component within the reactor pressure vessel with the vessel head removed and fuel in the vessel. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe conservative position.

SHUTDOWN MARGIN

1.13 SHUTDOWN MARGIN shall be the instantaneous amount of reactivity by which the reactor is or would be subcritical from its present condition assuming all full lenth rod cluster assemblies (shutdown and control) are fully inserted except for the single rod cluster assembly of highest reactivity worth which is assumed to be fully withdrawn.

IDENTIFIED LEAKAGE

1.14 IDENTIFIED LEAKAGE shall be:

a. Leakage (except CONTROLLED LEAKAGE) into closed systems, such as pump seal or valve packing leaks that are captured and conducted to a sump or collecting tank, or

b. Leakage into the containment atmosphere from sources that are both specifically located and known either not to interfere with the operation of leakage detection systems or not to be PRESSURE BOUNDARY LEAKAGE, or

c. Reactor coolant system leakage through a steam generator to the secondary system.

UNIDENTIFIED LEAKAGE

1.15 UNIDENTIFIED LEAKAGE shall be all leakage which is not IDENTIFIED LEAKAGE or CONTROLLED LEAKAGE.

PRESSURE BOUNDARY LEAKAGE

1.16 PRESSURE BOUNDARY LEAKAGE shall be leakage (except steam generator tube leakage) through a non-isolable fault in a Reactor Coolant System component body, pipe wall or vessel wall.

CONTROLLED LEAKAGE

1.17 CONTROLLED LEAKAGE shall be that seal water flow supplied to the reactor coolant pump seals.

QUADRANT POWER TILT RATIO

1.18 QUADRANT POWER TILT RATIO shall be the ratio of the maximum upper excore detector calibrated output to the average of the upper excore detector calibrated outputs, or the ratio of the maximum lower excore detector calibrated output to the average of the lower excore detector calibrated outputs, whichever is greater. With one excore detector inoperable, the remaining three detectors shall be used for computing the average.

DOSE EQUIVALENT I-131

1.19 DOSE EQUIVALENT I-131 shall be that concentration of I-131 (μCi/gram) which alone would produce the same thyroid dose as the quantity and isotopic mixture of I-131, I-132, I-133, I-134, and I-135 actually present. The thyroid dose conversion factors used for this calculation shall be those listed in Table III of TID-14844.

STAGGERED TEST BASIS

1.20 A STAGGERED TEST BASIS shall consist of:

a. A test schedule for n systems, subsystems, trains or other designated components obtained by dividing the specified test interval into n equal subintervals,

b. The testing of one system, subsystem, train or other designated component at the beginning of each subinterval.

FREQUENCY NOTATION

1.21 The FREQUENCY NOTATION specified for the performance of Surveillance Requirements shall correspond to the intervals defined in Table 1.2.

SURVEILLANCE REQUIREMENTS (Continued)

a. At least once per 7 days by:

   1. Verifying the boron concentration in each water source,

   2. Verifying the water level in each water source, and

   3. Verifying the boric acid storage system solution temperature.

b. At least once per 24 hours by verifying the RWST temperature when the RWST ambient air temperature is < 43°F.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

GROUP HEIGHT

LIMITING CONDITION FOR OPERATION

3.1.3.1 All full length (shutdown and control) rods which are inserted in the core shall be OPERABLE and positioned within ± 12 steps (indicated position) of their group step counter demand position.

APPLICABILITY: MODES 1* and 2*

ACTION:

a. With one or more full length rods inoperable due to being immovable as a result of excessive friction or mechanical interference or known to be untrippable, determine that the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied within 1 hour and be in HOT STANDBY within 6 hours.

b. With more than one full length rod inoperable or misaligned from any other rod in its group by more than ± 12 steps (indicated position), be in HOT STANDBY within 6 hours.

c. With one full length rod inoperable or misaligned from its group step counter demand height by more than ± 12 steps (indicated position), POWER OPERATION may continue provided that within one hour either:

   1. The rod is restored to OPERABLE status within the above alignment requirements, or

   2. The rod is declared inoperable and the SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is satisfied. POWER OPERATION may then continue provided that:

      a) An analysis of the potential ejected rod worth is performed within 3 days and the rod worth is determined to be ≤ 0.95% Δk at zero power and ≤ 0.20% Δk at RATED THERMAL POWER for the remainder of the fuel cycle, and

      b) The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 is determined at least once per 12 hours, and

*See Special Test Exceptions 3.10.2 and 3.10.4.

LIMITING CONDITION FOR OPERATION (Continued)

c) The THERMAL POWER level is reduced to ≤ 75% of RATED THERMAL POWER within one hour and within the next 4 hours the high neutron flux trip setpoint is reduced to ≤ 85% of RATED THERMAL POWER, or

d) The remainder of the rods in the group with the inoperable rod are aligned to within ± 12 steps of the inoperable rod within one hour while maintaining the rod sequence and insertion limits of Figures 3.1-1 and 3.1-2; the THERMAL POWER level shall be restricted pursuant to Specification 3.1.3.5 during subsequent operation.

SURVEILLANCE REQUIREMENTS

4.1.3.1.1 The position of each full length rod shall be determined to be within the group demand limit by verifying the individual rod positions at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then verify the group positions at least once per 4 hours.

4.1.3.1.2 Each full length rod not fully inserted shall be determined to be OPERABLE by movement of at least 10 steps in any one direction at least once per 31 days.

POSITION INDICATOR CHANNELS

LIMITING CONDITION FOR OPERATION

3.1.3.2 All shutdown and control rod position indicator channels and the demand position indication system shall be OPERABLE and capable of determining the control rod positions within ± 12 steps.

APPLICABILITY: MODES 1 and 2.

ACTION:

a. With a maximum of one rod position indicator channel per group inoperable either:

   1. Determine the position of the non-indicating rod(s) indirectly by the movable incore detectors at least once per 8 hours and immediately after any motion of the non-indicating rod which exceeds 24 steps in one direction since the last determination of the rod's position, or

   2. Reduce THERMAL POWER TO < 50% of RATED THERMAL POWER within 8 hours.

b. With a maximum of one demand position indicator per bank inoperable either:

   1. Verify that all rod position indicators for the affected bank are OPERABLE and that the most withdrawn rod and the least withdrawn rod of the bank are within a maximum of 12 steps of each other at least once per 8 hours, or

   2. Reduce THERMAL POWER to < 50% of RATED THERMAL POWER within 8 hours.

SURVEILLANCE REQUIREMENTS

4.1.3.2 Each rod position indicator channel shall be determined to be OPERABLE by verifying the demand position indication system and the rod position indicator channels agree within 12 steps at least once per 12 hours except during time intervals when the Rod Position Deviation Monitor is inoperable, then compare the demand position indication system and the rod position indicator channels at least once per 4 hours.

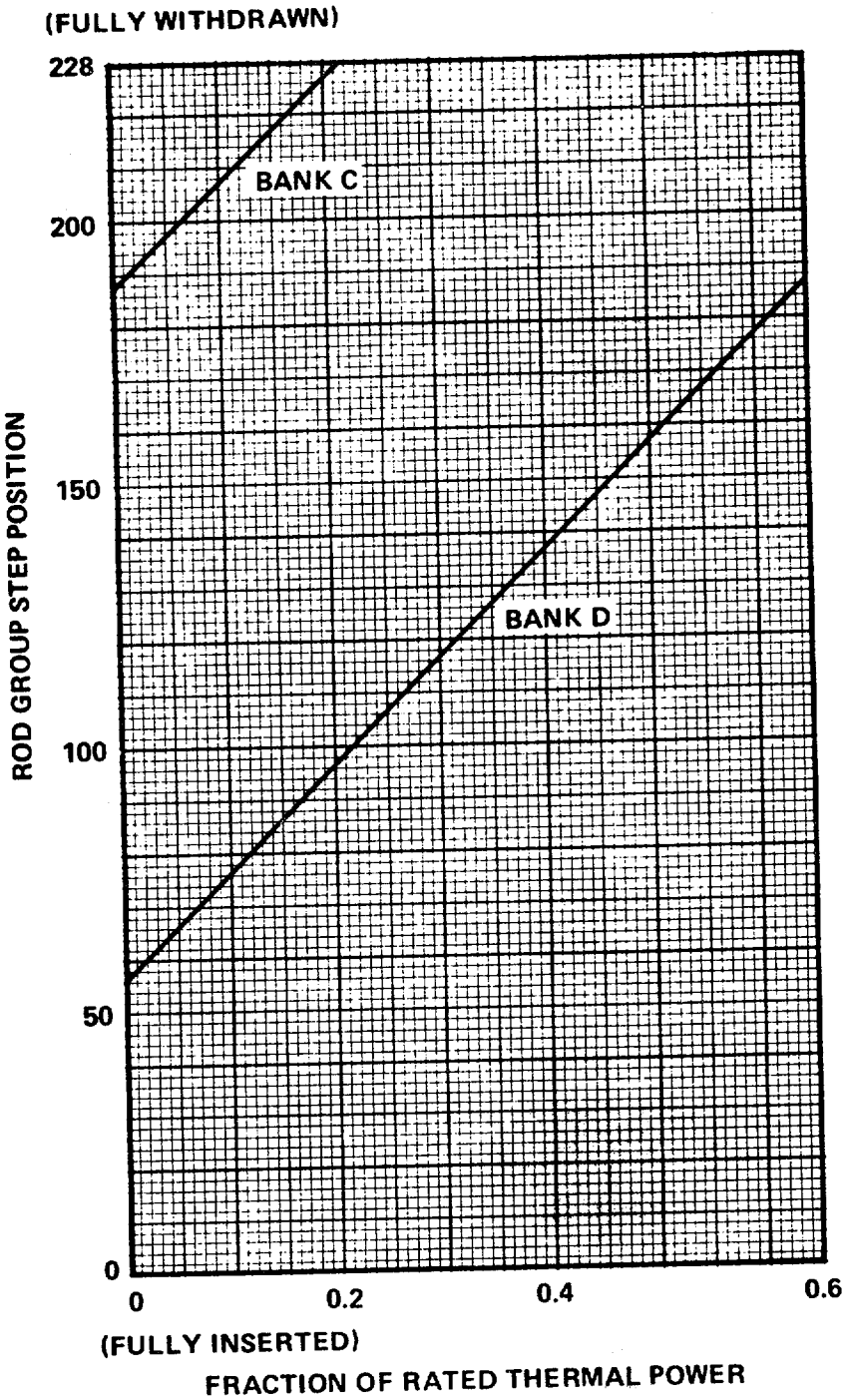

FIGURE 3.1-2 ROD GROUP INSERTION LIMITS VERSUS THERMAL POWER TWO LOOP OPERATION

PART LENGTH ROD INSERTION LIMITS

LIMITING CONDITION FOR OPERATION

This specification has been DELETED

SURVEILLANCE REQUIREMENTS

This specification has been DELETED

3/4.2 POWER DISTRIBUTION LIMITS

AXIAL FLUX DIFFERENCE (AFD)

LIMITING CONDITION FOR OPERATION

3.2.1 The indicated AXIAL FLUX DIFFERENCE (AFD) shall be maintained within a ± 7% target band (flux difference units) about the target flux difference.

APPLICABILITY: MODE 1 ABOVE 50% RATED THERMAL POWER*

ACTION:

a. With the indicated AXIAL FLUX DIFFERENCE outside of the ± 7% target band about the target flux difference and with THERMAL POWER:

1. Above 90% of RATED THERMAL POWER, within 15 minutes:

a) Either restore the indicated AFD to within the target band limits, or

b) Reduce THERMAL POWER to less than 90% of RATED THERMAL POWER.

2. Between 50% and 90% of RATED THERMAL POWER:

a) POWER OPERATION may continue provided:

1) The indicated AFD has not been outside of the ± 7% target band for more than 1 hour penalty deviation cumulative during the previous 24 hours, and

2) The indicated AFD is within the limits shown on Figure 3.2-1. Otherwise, reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 30 minutes and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours.

b) Surveillance testing of the Power Range Neutron Flux Channels may be performed pursuant to Specification 4.3.1.1.1 provided the indicated AFD is maintained within the limits of Figure 3.2-1. A total of 16 hours operation may be accumulated with the AFD outside of the target band during this testing without penalty deviation.

* See Special Test Exception 3.10.2

LIMITING CONDITION FOR OPERATION (Continued)

b. THERMAL POWER shall not be increased above 90% of RATED THERMAL POWER unless the indicated AFD is within the ± 7% target band and ACTION 2.a) 1), above has been satisfied.

c. THERMAL POWER shall not be increased above 50% of RATED THERMAL POWER unless the indicated AFD has not been outside of the ± 7% target band for more than 1 hour penalty deviation cumulative during the previous 24 hours.

SURVEILLANCE REQUIREMENTS

4.2.1.1 The indicated AXIAL FLUX DIFFERENCE shall be determined to be within its limits during POWER OPERATION above 15% of RATED THERMAL POWER by:

a. Monitoring the indicated AFD for each OPERABLE excore channel:

1. At least once per 7 days when the AFD Monitor Alarm is OPERABLE, and

2. At least once per hour for the first 24 hours after restoring the AFD Monitor Alarm to OPERABLE status.

b. Monitoring and logging the indicated AXIAL FLUX DIFFERENCE for each OPERABLE excore channel at least once per hour for the first 24 hours and at least once per 30 minutes thereafter, when the AXIAL FLUX DIFFERENCE Monitor Alarm is inoperable. The logged values of the indicated AXIAL FLUX DIFFERENCE shall be assumed to exist during the interval preceding each logging.

4.2.1.2 The indicated AFD shall be considered outside of its ± 7% target band when at least 2 of 4 or 2 of 3 OPERABLE excore channels are indicating the AFD to be outside the target band. POWER OPERATION outside of the ± 7% target band shall be accumulated on a time basis of:

a. One minute penalty deviation for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels equal to or above 50% of RATED THERMAL POWER, and

b. One-half minute penalty deviation for each one minute of POWER OPERATION outside of the target band at THERMAL POWER levels below 50% of RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS (Continued)

4.2.1.3 The target flux difference of each OPERABLE excore channel shall be determined by measurement at least once per 92 Effective Full Power Days. The provisions of Specification 4.0.4 are not applicable.

4.2.1.4 The target flux difference shall be updated at least once per 31 Effective Full Power Days by either determining the target flux difference pursuant to 4.2.1.3 above or by linear interpolation between the most recently measured value and 0 percent at the end of the cycle life. The provisions of Specification 4.0.4 are not applicable.

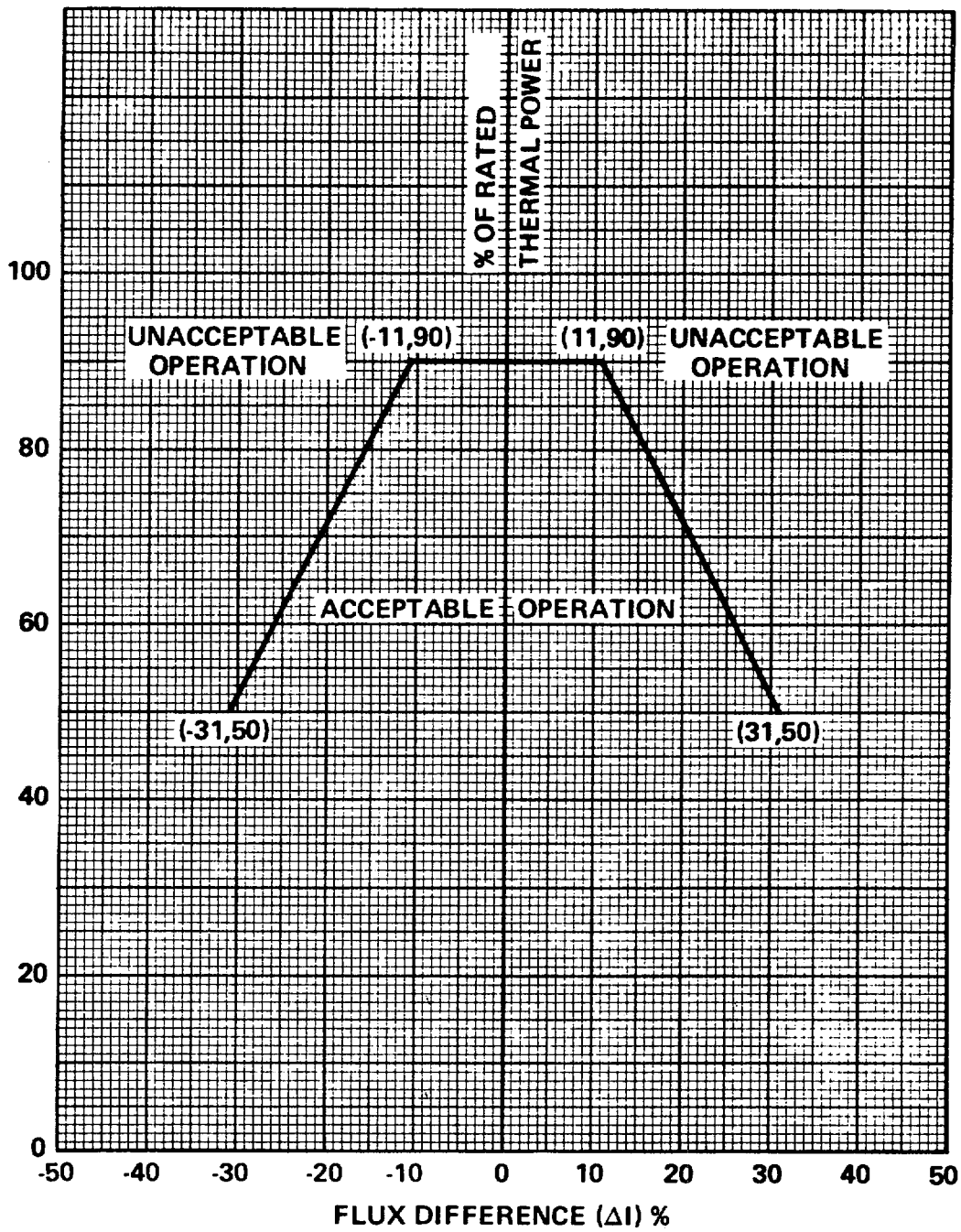

FIGURE 3.2-1 AXIAL FLUX DIFFERENCE LIMITS AS A FUNCTION OF RATED THERMAL POWER

SURVEILLANCE REQUIREMENTS

4.2.3.1 $F^N_{\Delta H}$ shall be determined to be within its limit by using the movable incore detectors to obtain a power distribution map:

a. Prior to operation above 75% of RATED THERMAL POWER after each fuel loading, and

b. At least once per 31 Effective Full Power Days.

4.2.3.2 The measured $F^N_{\Delta H}$ of 4.2.3.1 above, shall be increased by 4% for measurement uncertainty.

QUADRANT POWER TILT RATIO

LIMITING CONDITION FOR OPERATION

3.2.4 THE QUADRANT POWER TILT RATIO shall not exceed 1.02.

APPLICABILITY: MODE 1 ABOVE 50% OF RATED THERMAL POWER*

ACTION:

a. With the QUADRANT POWER TILT RATIO determined to exceed 1.02 but ≤ 1.09:

1. Within 2 hours:

a) Either reduce the QUADRANT POWER TILT RATIO to within its limit, or

b) Reduce THERMAL POWER at least 3% for each 1% of indicated QUADRANT POWER TILT RATIO in excess of 1.0 and similarly reduce the Power Range Neutron Flux-High Trip Setpoints within the next 4 hours.

2. Verify that the QUADRANT POWER TILT RATIO is within its limit within 24 hours after exceeding the limit or reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High Trip setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours.

3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL power may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour until verified acceptable at 95% or greater RATED THERMAL POWER.

b. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to misalignment of either a shutdown or control rod:

1. Reduce THERMAL POWER at least 3% for each 1% of indicated QUADRANT POWER TILT RATIO in exuess of 1.0, within 30 minutes.

2. Verify that the QUADRANT POWER TILT RATIO is within its limit within 2 hours after exceeding the limit or

*See Special Test Exception 3.10.2.

b) At least once per 31 EFPD, whichever occurs first.

2. When the F_{xy}^{C} is less than or equal to the F_{xy}^{RTP} limit for the appropriate measured core plane, additional power distribution maps shall be taken and F_{xy}^{C} compared to F_{xy}^{RTP} and F_{xy}^{L} at least once per 31 EFPD.

e. The F_{xy} limits for RATED THERMAL POWER within specific core planes shall be:

1. $F_{xy}^{RTP} \leq 1.71$ for all core planes containing bank "D" control rods, and

2. $F_{xy}^{RTP} \leq 1.55$ for all unrodded core planes.

f. The F_{xy} limits of e, above, are not applicable in the following core plane regions as measured in percent of core height from the bottom of the fuel:

1. Lower core region from 0 to 15%, inclusive.
2. Upper core region from 85 to 100% inclusive.
3. Grid plane regions at 17.8 ± 2%, 32.1 ± 2%, 46.4 ± 2%, 60.6 ± 2% and 74.9 ± 2%, inclusive.
4. Core plane regions within ± 2% of core height (± 2.88 inches) about the bank demand position of the bank "D" control rods.

g. With F_{xy}^{C} exceeding F_{xy}^{L}, the effects of F_{xy} on F_Q (Z) shall be evaluated to determine if F_Q (Z) is within its limit.

4.2.2.3 When $F_Q(Z)$ is measured pursuant to Specification 4.10.2.2, an overall measured $F_Q(Z)$ shall be obtained from a power distribution map and increased by 3% to account for manufacturing tolerances and further increased by 5% to account for measurement uncertainty.

LIMITING CONDITION FOR OPERATION (Continued)

reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within the next 2 hours and reduce the Power Range Neutron Flux-High trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours.

3. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour until verified acceptable at 95% or greater RATED THERMAL POWER.

c. With the QUADRANT POWER TILT RATIO determined to exceed 1.09 due to causes other than the misalignment of either a shutdown or control rod:

1. Reduce THERMAL POWER to less than 50% of RATED THERMAL POWER within 2 hours and reduce the Power Range Neutron Flux-High Trip Setpoints to ≤ 55% of RATED THERMAL POWER within the next 4 hours.

2. Identify and correct the cause of the out of limit condition prior to increasing THERMAL POWER; subsequent POWER OPERATION above 50% of RATED THERMAL POWER may proceed provided that the QUADRANT POWER TILT RATIO is verified within its limit at least once per hour until verified at 95% or greater RATED THERMAL POWER.

SURVEILLANCE REQUIREMENTS

4.2.4 The QUADRANT POWER TILT RATIO shall be determined to be within the limit above 50% of RATED THERMAL POWER by:

a. Calculating the ratio at least once per 7 days when the alarm is OPERABLE.

b. Calculating the ratio at least once per 12 hours during steady state operation when the alarm is inoperable.

c. Using the movable detectors to determine the QUADRANT POWER TILT RATIO at least once per 12 hours when one Power Range Channel is inoperable and THERMAL POWER is > 75 percent of RATED THERMAL POWER.

DNB PARAMETERS

LIMITING CONDITION FOR OPERATION

3.2.5 The following DNB related parameters shall be maintained within the limits shown on Table 3.2-1:

a. Reactor Coolant System T_{avg}.

b. Pressurizer Pressure

c. Reactor Coolant System Total Flow Rate

APPLICABILITY: MODE 1

ACTION:

With any of the above parameters exceeding its limit, restore the parameter to within its limit within 2 hours or reduce THERMAL POWER to less than 5% of RATED THERMAL POWER within the next 4 hours.

SURVEILLANCE REQUIREMENTS

4.2.5.1 Each of the parameters of Table 3.2-1 shall be verified to be indicating within their limits at least once per 12 hours.

4.2.5.2 The Reactor Coolant System total flow rate shall be determined to be within its limit by measurement at least once per 18 months.

PRESSURIZER

LIMITING CONDITION FOR OPERATION

3.4.9.2 The pressurizer temperature shall be limited to:

a. A maximum heatup of 100°F in any one hour period,

b. A maximum cooldown of 200°F in any one hour period, and

c. A maximum spray water temperature differential of 320°F.

APPLICABILITY: At all times.

ACTION:

With the pressurizer temperature limits in excess of any of the above limits, restore the temperature to within the limits within 30 minutes; perform an analysis to determine the effects of the out-of-limit condition on the fracture toughness properties of the pressurizer; determine that the pressurizer remains acceptable for continued operation or be in at least HOT STANDBY within the next 6 hours and reduce the pressurizer pressure to less than 500 psig within the following 30 hours.

SURVEILLANCE REQUIREMENTS

4.4.9.2 The pressurizer temperatures shall be determined to be within the limits at least once per 30 minutes during system heatup or cooldown. The spray water temperature differential shall be determined to be within the limit at least once per 12 hours during steady state operation.

3.4.10 STRUCTURAL INTEGRITY

REACTOR PRESSURE VESSEL AND REACTOR COOLANT SYSTEM BOUNDARY

LIMITING CONDITION FOR OPERATION

3.4.10.1 The structural integrity of the reactor pressure vessel and reactor coolant system boundary (except steam generator tubes) shall be maintained at a level consistent with the acceptance criteria in Specifications 4.4.10.1.1 and 4.4.10.1.2.

APPLICABILITY: MODES 1, 2, 3 and 4.

ACTION:

With the structural integrity of any of the above components not conforming to the above requirements, restore the structural integrity of the affected component to within its limit or isolate the affected component prior to increasing the Reactor Coolant System temperature more than 50°F above the minimum temperature required by NDT considerations. The provisions of Specification 3.0.4 are not applicable.

SURVEILLANCE REQUIREMENTS

4.4.10.1.1 The following inspection program shall be performed during shutdown:

a. Inservice Inspections The structural integrity of the above components shall be demonstrated by verifying their acceptability when inspected per the applicable requirements of Section XI of the ASME Boiler and Pressure Vessel Code, 1971 Edition, as outlined by the inspection program shown in Table 4.4-4.

3/4.5 EMERGENCY CORE COOLING SYSTEMS (ECCS)

ACCUMULATORS

LIMITING CONDITION FOR OPERATION

3.5.1 Each reactor coolant system accumulator shall be OPERABLE with:

a. The isolation valve open,

b. Between 7664 and 7816 gallons of borated water,

c. Between 1900 and 2100 ppm of boron, and

d. A nitrogen cover-pressure of between 605 and 661 psig.

APPLICABILITY: MODES 1, 2 and 3.*

ACTION:

a. With one accumulator inoperable, except as a result of a closed isolation valve, restore the inoperable accumulator to OPERABLE status within one hour or be in at least HOT STANDBY within the next 6 hours and in HOT SHUTDOWN within the following 6 hours.

b. With one accumulator inoperable due to the isolation valve being closed, either immediately open the isolation valve or be in HOT STANDBY within one hour and be in HOT SHUTDOWN within the next 12 hours.

SURVEILLANCE REQUIREMENTS

4.5.1 Each accumulator shall be demonstrated OPERABLE:

a. At least once per 12 hours by:

1. Verifying, by the absence of alarms, the contained borated water volume and nitrogen cover-pressure in the tanks, and

2. Verifying that each accumulator isolation valve is open.

*Pressurizer Pressure above 1000 psig.

SURVEILLANCE REQUIREMENTS (Continued)

b. At least once per 31 days and within 6 hours after each solution volume increase of greater than or equal to 1% of tank volume by verifying the boron concentration of the accumulator solution.

c. At least once per 31 days when the RCS pressure is above 2000 psig be verifying that power to the isolation valve operator control circuit is disconnected by removal of the plug in the lock out jack from the circuit.

d. Verifying at least once per 18 months that each accumulator isolation valve opens automatically under each of the following conditions:

1. When the RCS pressure exceeds 2000 psig,

2. Upon receipt of a Safety Injection test signal.

4.5.1.2 Each accumulator water level and pressure alarm channel shall be demonstrated OPERABLE:

a. At least once per 31 days by the performance of a CHANNEL FUNCTIONAL TEST.

b. At least once per 18 months by the performance of a CHANNEL CALIBRATION.

ECCS SUBSYSTEMS - $T_{avg} \geq 350°F$

LIMITING CONDITION FOR OPERATION

3.5.2 Two separate and independent ECCS subsystems shall be OPERABLE with each subsystem comprised of:

a. One OPERABLE centrifugal charging pump,

b. One OPERABLE low head safety injection pump, and

c. An OPERABLE flow path capable of taking suction from the refueling water storage tank on a safety injection signal and transferring suction to the containment sump during the recirculation phase of operation.

APPLICABILITY: MODES 1, 2 and 3.

ACTION:

a. With one ECCS subsystem inoperable, restore the inoperable subsystem to OPERABLE status within 72 hours or be in HOT SHUTDOWN within the next 12 hours.*

b. In the event the ECCS is actuated and injects water into the Reactor Coolant System, a Special Report shall be prepared and submitted to the Commission pursuant to Specification 6.9.2 within 90 days describing the circumstances of the actuation and the total accumulated actuation cycles to date.

*An additional 24 hours is permitted to align the drive motor of 1A Low Head Safety Injection pump on a one time basis to be completed on March 21, 1978.

SURVEILLANCE REQUIREMENTS

4.5.2 Each ECCS subsystem shall be demonstrated OPERABLE:

a. At least once per 12 hours by verifying that the following valves are in the indicated positions with power to the valve operator control circuits disconnected by removal of the plug in the lock out circuit from each circuit:

	Valve Number	Valve Function	Valve Position
a.	MOV SI 890 A	LHSI to hot leg	CLOSED
b.	MOV SI 890 B	LHSI to hot leg	CLOSED
c.	MOV SI 890 C	LHSI to cold leg	OPEN
d.	MOV SI 869 A	Ch Pmp to hot leg	CLOSED
e.	MOV SI 869 B	Ch Pmp to hot leg	CLOSED

b. At least once per 31 days on a STAGGERED TEST BASIS by:

1. Verifying that each centrifugal charging pump:

   a) Starts (unless already operating) from the control room.

   b) Develops a discharge pressure of $\geq$ 2402 psig on recirculation flow.

   c) Operates for at least 15 minutes.

2. Verifying that each low head safety injection pump:

   a) Starts (unless already operating) from the control room.

   b) Develops a discharge pressure $\geq$ 159 psig on recirculation flow.

   c) Operates for at least 15 minutes.

TABLE 3.7-4 (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION				ACCESSIBLE OR INACCESSIBLE	HIGH RADIATION ZONE	ESPECIALLY DIFFICULT TO REMOVE
RC-HC-9B	RC	RCP	Cub. B	766'10"	I	Yes	Yes
RC-HC-10B	"	"	"	766'10"	"	"	"
RC-HC-11B	"	"	"	766'10"	"	"	"
RC-HC-12B	"	"	"	766'10"	"	"	"
RC-HC-1C	RC	RCP	Cub. C	731'2"	I	Yes	Yes
RC-HC-2C	"	"	"	736'2"	"	"	"
RC-HC-3C	"	"	"	728'2"	"	"	"
RC-HC-4C	"	"	"	736'6"	"	"	"
RC-HC-5C	"	"	"	736'6"	"	"	"
RC-HC-6C	"	"	"	731'2"	"	"	"
RC-HC-7C	"	"	"	728'2"	"	"	"
RC-HC-8C	"	"	"	736'6"	"	"	"
RC-HC-9C	"	"	"	766'10"	"	"	"
RC-HC-10C	"	"	"	766'10"	"	"	"
RC-HC-11C	"	"	"	766'10"	"	"	"
RC-HC-12C	"	"	"	766'10"	"	"	"
RC-HSS-5	RC REA. CONT. BLDG.			716'	I	Yes	Yes
RC-HSS-6	"			716'	"	"	"
RC-HSS-7	"			716'	"	"	"
RC-HSS-8	"			716'	"	"	"

TABLE 3.7-4 (Continued)

SAFETY RELATED HYDRAULIC SNUBBERS*

SNUBBER NO.	SYSTEM SNUBBER INSTALLED ON, LOCATION AND ELEVATION			ACCESSIBLE OR INACCESSIBLE	HIGH RADIATION ZONE	ESPECIALLY DIFFICULT TO REMOVE
RC-HSS-9	RC	Reac. Cont. Bldg.	716'	I	Yes	Yes
RC-HSS-10	"	"	716'	I	Yes	Yes
RC-HSS-123	RHR	"	699'	A	No	No
RC-HSS-124	RHR	"	699'	A	No	No
SI-HSS-110	SI	"	708'	A	No	No
SI-HSS-111	"	"	708'	A	No	No
SI-HSS-119	"	"	697'	A	No	No
SI-HSS-120	"	"	697'	A	No	No
SI-HSS-108	"	"	708'	A	No	No
SI-HSS-103	"	"	695'	A	No	No
SI-HSS-101	"	"	705'	A	No	No
SI-HSS-104	"	"	705'	A	No	No
RH-HSS-106	RHR	"	704"	A	Yes	No
RH-HSS-109	RHR	"	704'	A	Yes	No
RH-HSS-110	RHR	"	704'	A	Yes	No
RH-HSS-112	RHR	"	704'	A	Yes	No
RC-HSS-11	RC	RCP Cub. A	723'	I	Yes	Yes
RC-HSS-12	RC	" "	723'	I	Yes	Yes
RC-HSS-13	"	" "	735'	I	Yes	Yes
RC-HSS-14	"	" "	735'	I	Yes	Yes

3/4.9 REFUELING OPERATIONS

BORON CONCENTRATION

LIMITING CONDITION FOR OPERATION

3.9.1 With the reactor vessel head unbolted or removed, the boron concentration of all filled portions of the Reactor Coolant System and the refueling canal shall be maintained uniform and sufficient to ensure that the more restrictive of the following reactivity conditions is met:

a. Either a K_{eff} of 0.95 or less, which includes a 1% $\Delta k/k$ conservative allowance for uncertainties, or

b. A boron concentration of 2000 ppm, which includes a 50 ppm conservative allowance for uncertainties.

APPLICABILITY: MODE 6*

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes and initiate and continue boration at $\geq$ 30 gpm of 7000 ppm boric acid solution or its equivalent until K_{eff} is reduced to $\leq$ 0.95 or the boron concentration is restored to $\geq$ 2000 ppm, whichever is the more restrictive. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.1.1 The more restrictive of the above two reactivity conditions shall be determined prior to:

a. Removing or unbolting the reactor vessel head, and

b. Withdrawal of any full length control rod in excess of 3 feet from its fully inserted position.

4.9.1.2 The boron concentration of the reactor coolant system and the refueling canal shall be determined by chemical analysis at least 3 times per 7 days with a maximum time interval between samples of 72 hours.

* The reactor shall be maintained in MODE 6 when the reactor vessel head is unbolted or removed.

INSTRUMENTATION

LIMITING CONDITION FOR OPERATION

3.9.2 As a minimum, two source range neutron flux monitors shall be operating, each with continuous visual indication in the control room and one with audible indication in the containment and control room.

APPLICABILITY: MODE 6.

ACTION:

With the requirements of the above specification not satisfied, immediately suspend all operations involving CORE ALTERATIONS or positive reactivity changes. The provisions of Specification 3.0.3 are not applicable.

SURVEILLANCE REQUIREMENTS

4.9.2 Each source range neutron flux monitor shall be demonstrated OPERABLE by performance of:

a. A CHANNEL FUNCTIONAL TEST at least once per 7 days, and

b. A CHANNEL FUNCTIONAL TEST within 8 hours prior to the initial start of CORE ALTERATIONS, and

c. A CHANNEL CHECK at least once per 12 hours during CORE ALTERATIONS.

3/4.10 SPECIAL TEST EXCEPTIONS

SHUTDOWN MARGIN

LIMITING CONDITION FOR OPERATION

3.10.1 The SHUTDOWN MARGIN requirement of Specification 3.1.1.1 may be suspended for measurement of control rod worth and shutdown margin provided the reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion from OPERABLE control rod(s) and

APPLICABILITY: MODE 2.

ACTION:

a. With the reactor critical ($K_{eff} \geq 1.0$) and with less than the above reactivity equivalent available for trip insertion, immediately initiate and continue boration at $\geq$ 30 gpm of 7000 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

b. With the reactor subcritical ($K_{eff} < 1.0$) by less than the above reactivity equivalent, immediately initiate and continue boration at $\geq$ 30 gpm of 7000 ppm boric acid solution or its equivalent until the SHUTDOWN MARGIN required by Specification 3.1.1.1 is restored.

SURVEILLANCE REQUIREMENTS

4.10.1.1 The position of each full length rod either partially or fully withdrawn shall be determined at least once per 2 hours.

4.10.1.2 Each full length rod not fully inserted shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the SHUTDOWN MARGIN to less than the limits of Specification 3.1.1.1.

GROUP HEIGHT, INSERTION AND POWER DISTRIBUTION LIMITS

LIMITING CONDITION FOR OPERATION

3.10.2 The group height, insertion and power distribution limits of Specifications 3.1.3.1, 3.1.3.4, 3.1.3.5 3.2.1 and 3.2.4 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER is maintained ≤ 85% of RATED THERMAL POWER, and

b. The limits of Specifications 3.2.2 and 3.2.3 are maintained and determined at the frequencies specified in Specification 4.10.2.2 below.

APPLICABILITY: MODE 1

ACTION:

With any of the limits of Specifications 3.2.2 or 3.2.3 being exceeded while the requirements of Specifications 3.1.3.1, 3.1.3.4, 3.1.3.5, 3.2.1, and 3.2.4 are suspended, either:

a. Reduce THERMAL POWER sufficient to satisfy the ACTION requirements of Specifications 3.2.2 and 3.2.3, or

b. Be in HOT STANDBY within 6 hours.

SURVEILLANCE REQUIREMENTS

4.10.2.1 The THERMAL POWER shall be determined to be ≤ 85% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

SURVEILLANCE REQUIREMENTS (Continued)

4.10.3.3 Each Intermediate and Power Range Nuclear Channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating low temperature PHYSICS TESTS.

SPECIAL TEST EXCEPTIONS

PHYSICS TESTS

LIMITING CONDITION FOR OPERATION

3.10.4 The limitations of Specifications 3.1.1.4, 3.1.3.1, 3.1.3.4, and 3.1.3.5 may be suspended during the performance of PHYSICS TESTS provided:

a. The THERMAL POWER does not exceed 5% of RATED THERMAL POWER, and

b. The reactor trip setpoints on the OPERABLE Intermediate and Power Range Nuclear Channels are set at $\leq$ 25% of RATED THERMAL POWER.

APPLICABILITY: MODE 2.

ACTION:

With the THERMAL POWER > 5% of RATED THERMAL POWER, immediately open the reactor trip breakers.

SURVEILLANCE REQUIREMENTS

4.10.4.1 The THERMAL POWER shall be determined to be $\leq$ 5% of RATED THERMAL POWER at least once per hour during PHYSICS TESTS.

4.10.4.2 Each Intermediate and Power Range Channel shall be subjected to a CHANNEL FUNCTIONAL TEST within 12 hours prior to initiating PHYSICS TESTS.

3/4.1.2 BORATION SYSTEMS (Continued)

The boration capability of either system is sufficient to provide a SHUTDOWN MARGIN from all operating conditions of 1.0% Δk/k after xenon decay and cooldown to 200°F. The maximum boration capability requirement occurs at EOL from full power equilibrium xenon conditions and requires 11,336 gallons of 7000 ppm borated water from the boric acid storage tanks or 49,917 gallons of 2000 ppm borated water from the refueling water storage tank.

With the RCS temperature below 200°F, one injection system is acceptable without single failure consideration on the basis of the stable reactivity condition of the reactor and the additional restrictions prohibiting CORE ALTERATIONS and positive reactivity change in the event the single injection system becomes inoperable.

The boron capability required below 200°F is sufficient to provide a SHUTDOWN MARGIN of 1% Δk/k after xenon decay and cooldown from 200°F to 140°F. This condition requires either 5000 gallons of 7000 ppm borated water from the boric acid storage tanks or 175,000 gallons of 2000 ppm borated water from the refueling water storage tank.

3/4.1.3 MOVABLE CONTROL ASSEMBLIES

The specifications of this section ensure that (1) acceptable power distribution limits are maintained, (2) the minimum SHUTDOWN MARGIN is maintained, and (3) limit the potential effects of a rod ejection accident. OPERABILITY of the control rod position indicators is required to determine control rod positions and thereby ensure compliance with the control rod alignment and insertion limits.

The ACTION statements which permit limited variations from the basic requirements are accompanied by additional restrictions which ensure that the original criteria are met. Misalignment of a rod requires measurement of peaking factors or a restriction in THERMAL POWER; either of these restrictions provide assurance of fuel rod integrity during continued operation. The reactivity worth of a misaligned rod is limited for the remainder of the fuel cycle to prevent exceeding the assumptions used in the accident analysis for a rod ejection accident.

3/4.1.1 MOVABLE CONTROL ASSEMBLIES (Continued)

The maximum rod drop time restriction is consistent with the assumed rod drop time used in the accident analyses. Measurement with $T_{avg} \geq 541°F$ and with all reactor coolant pumps operating ensures that the measured drop times will be representative of insertion times experienced during a reactor trip at operating conditions.

Control rod positions and OPERABILITY of the rod position indicators are required to be verified on a nominal basis of once per 12 hours with more frequent verifications required if an automatic monitoring channel is inoperable. These verification frequencies are adequate for assuring that the applicable LCO's are satisfied.

BASES

The specifications of this section provide assurance of fuel integrity during Condition I (Normal Operation) and II (Incidents of Moderate Frequency) events by: (a) maintaining the minimum DNBR in the core $\geq$ 1.30 during normal operation and in short term transients, and (b) limiting the fission gas release, fuel pellet temperature & cladding mechanical properties to within assumed design criteria. In addition, limiting the peak linear power density during Conditions I events provides assurance that the initial conditions assumed for the LOCA analyses are meet and the ECCS acceptance criteria limit of 2200°F is not exceeded.

The definitions of hot channel factors as used in these specifications are as follows:

$F_Q(Z)$ Heat Flux Hot Channel Factor, is defined as the maximum local heat flux on the surface of a fuel rod at core elevation Z divided by the average fuel rod heat flux, allowing for manufacturing tolerances on fuel pellets and rods.

$F^N_{\Delta H}$ Nuclear Enthalpy Rise Hot Channel Factor, is defined as the ratio of the integral of linear power along the rod with the highest integrated power to the average rod power.

3/4.2.1 AXIAL FLUX DIFFERENCE (AFD)

The limits on AXIAL FLUX DIFFERENCE assure that the $F_Q(Z)$ upper bound envelope of 2.32 times the normalized axial peaking factor is not exceeded during either normal operation or in the event of xenon redistribution following power changes.

Target flux difference is determined at equilibrium xenon conditions. The full length rods may be positioned within the core in accordance with their respective insertion limits and should be inserted near their normal position for steady state operation at high power levels. The value of the target flux difference obtained under these conditions divided by the fraction of RATED THERMAL POWER is the target flux difference at RATED THERMAL POWER for the associated core burnup conditions. Target flux differences for other THERMAL POWER levels are

obtained by multiplying the RATED THERMAL POWER value by the appropriate fractional THERMAL POWER level. The periodic updating of the target flux difference value is necessary to reflect core burnup considerations.

Although it is intended that the plant will be operated with the AXIAL FLUX DIFFERENCE within the ± 7% target band about the target flux difference, during rapid plant THERMAL POWER reductions, control rod motion will cause the AFD to deviate outside of the target band at reduced THERMAL POWER levels. This deviation will not affect the xenon redistribution sufficiently to change the envelope of peaking factors which may be reached on a subsequent return to RATED THERMAL POWER (with the AFD within the target band) provided the time duration of the deviation is limited. Accordingly, a 1 hour penalty deviation limit cumulative during the previous 24 hours is provided for operation outside of the target band but within the limits of Figure 3.2-1 while at THERMAL POWER levels between 50% & 90% of RATED THERMAL POWER. For THERMAL POWER levels between 15% & 50% of RATED THERMAL POWER, deviations of the AFD outside of the target band are less significant. The penalty of 2 hours actual time reflects this reduced significance.

Provisions for monitoring the AFD on an automatic basis are derived from the plant process computer through the AFD Monitor Alarm. The computer determines the one minute average of each of the OPERABLE excore detector outputs and provides an alarm message immediately if the AFD for at least 2 of 4 or 2 of 3 OPERABLE excore channels are outside the target band & the THERMAL POWER is greater than 90% of RATED THERMAL POWER. During operation at THERMAL POWER levels between 50% & 90% & 15% & 50% RATED THERMAL POWER, the computer outputs an alarm message when the penalty deviation accumulates beyond the limits of 1 hour & 2 hours, respectively.

Figure B 3/4 2-1 shows a typical monthly target band near the beginning of core life.

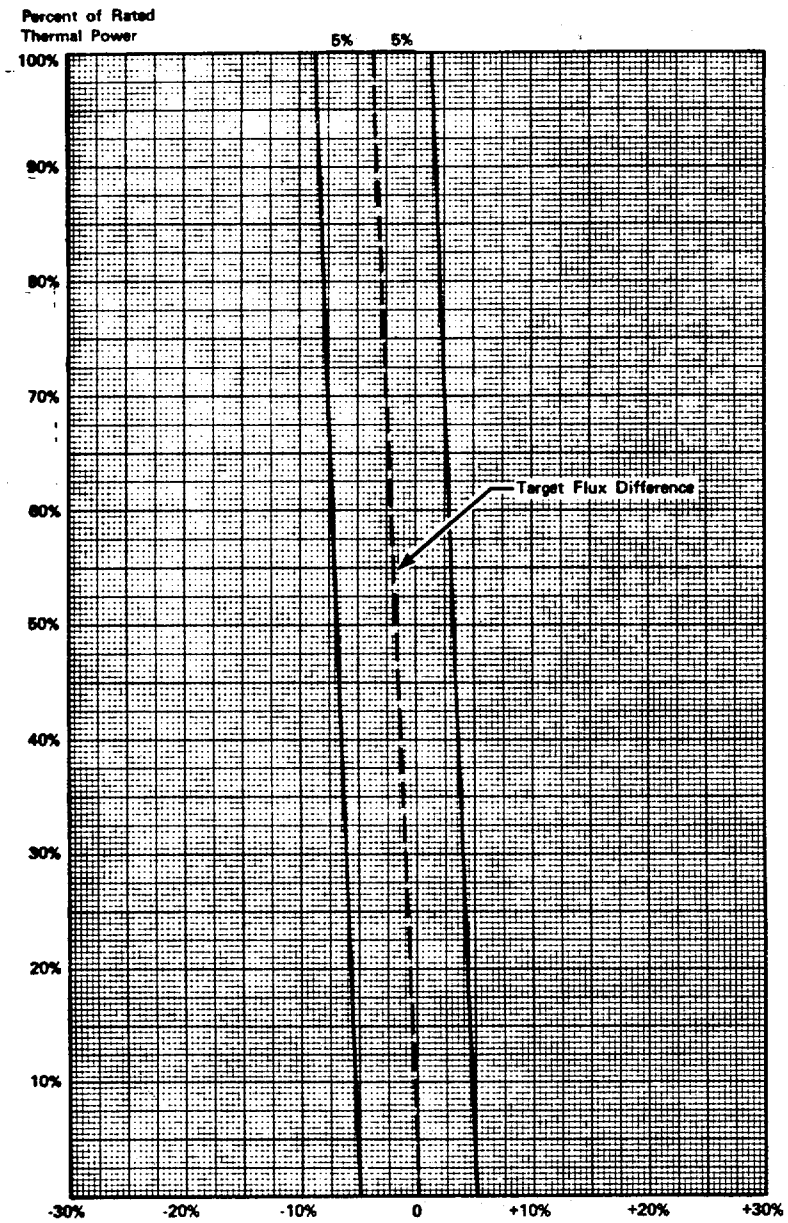

Figure B 3/4 2-1 TYPICAL INDICATED AXIAL FLUX DIFFERENCE VERSUS THERMAL POWER AT BOL

FIGURE B 3/4. 2-1

3/4.2.2 and 3/4.2.3 HEAT FLUX AND NUCLEAR ENTHALPY HOT CHANNEL FACTORS- $F_Q(Z)$ and $F^N_{\Delta H}$

The limits on heat flux and nuclear enthalpy hot channel factors ensure that 1) the design limits on peak local power density and minimum DNBR are not exceeded and 2) in the event of a LOCA the peak fuel clad temperature will not exceed the ECCS acceptance criteria limit of 2200°F.

Each of these hot channel factors are measurable but will normally only be determined periodically as specified in Specifications 4.2.2 and 4.2.3. This periodic surveillance is sufficient to insure that the hot channel factor limits are maintained provided:

a. Control rod in a single group move together with no individual rod insertion differing by more than ± 12 steps from the group demand position.

b. Control rod groups are sequenced with overlapping groups as described in Specification 3.1.3.5.

c. The control rod insertion limits of Specifications 3.1.3.4 and 3.1.3.5 are maintained.

d. The axial power distribution, expressed in terms of AXIAL FLUX DIFFERENCE is maintained within the limits.

The relaxation in $F^N_{\Delta H}$ as a function of THERMAL POWER allows changes in the radial power shape for all permissible rod insertion limits. $F^N_{\Delta H}$ will be maintained within its limits provided conditions a thru d above, are maintained.

When an F_Q measurement is taken, both experimental error and manufacturing tolerance must be allowed for. 5% is the appropriate allowance for a full core map taken with the incore detector flux mapping system and 3% is the appropriate allowance for manufacturing tolerance.

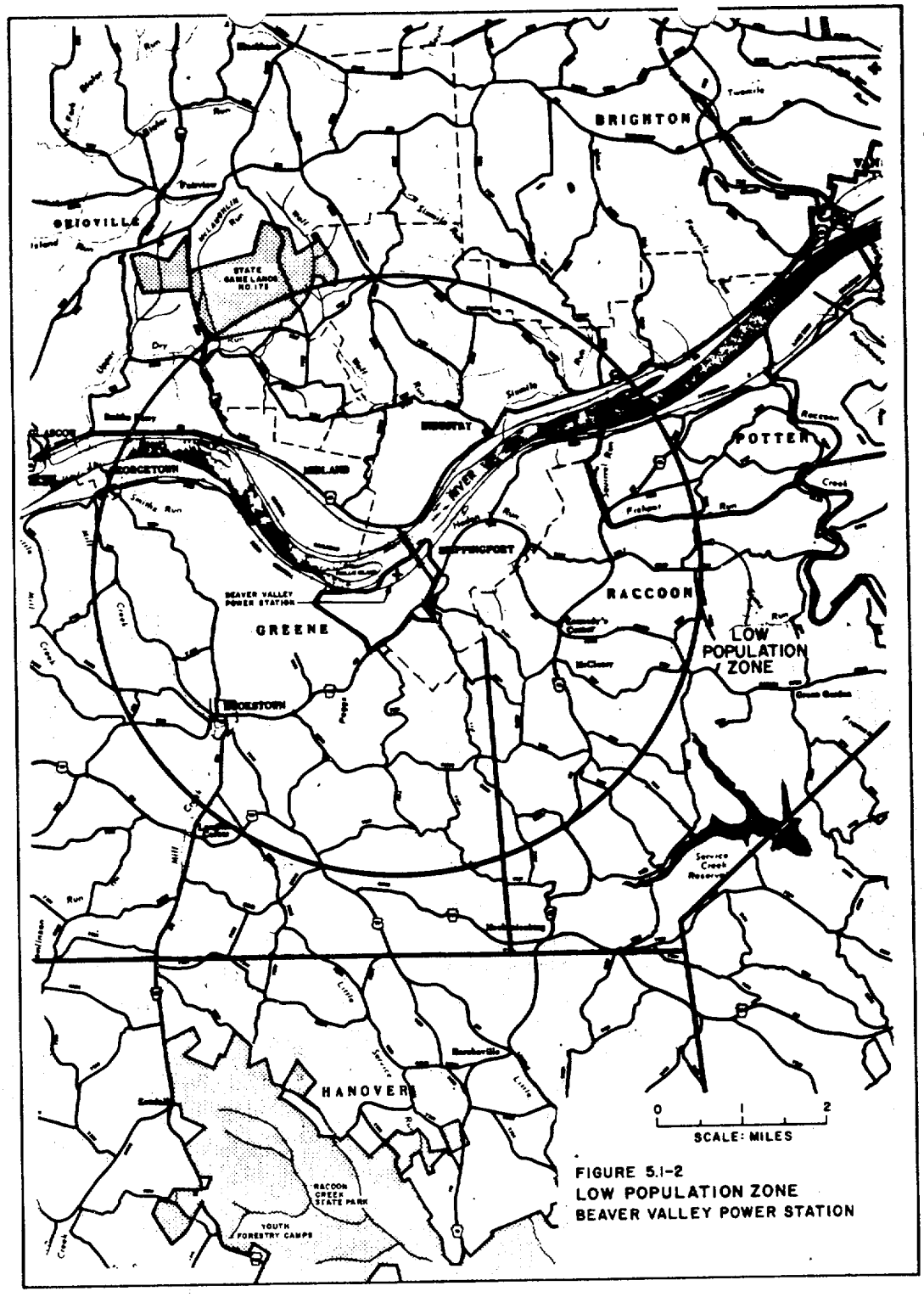

FIGURE 5.1-2
LOW POPULATION ZONE
BEAVER VALLEY POWER STATION

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 45 psig and a temperature of 280°F.

PENETRATIONS

5.2.3 Penetrations through the reactor containment building are designed and shall be maintained in accordance with the original design provisions contained in Section 5.2.4 of the FSAR with allowance for normal degradation pursuant to the applicable Surveillance Requirements.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 157 fuel assemblies with each fuel assembly containing 264 fuel rods clad with Zircaloy -4. Each fuel rod shall have a nominal active fuel length of 144 inches and contain a maximum total weight of 1766 grams uranium. The initial core loading shall have a maximum enrichment of 3.2 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.3 weight percent U-235.

CONTROL ROD ASSEMBLIES

5.3.2 The reactor core shall contain 48 full length and no part length control rod assemblies. The full length control rod assemblies shall contain a nominal 142 inches of absorber material. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmium. All control rods shall be clad with stainless steel tubing.

6.3 FACILITY STAFF QUALIFICATIONS

6.3.1 Each member of the facility staff shall meet or exceed the minimum qualifications of ANSI N18.1-1971 for comparable positions, except for the Station Reactor Control Chemist who shall meet or exceed the qualifications of Regulatory Guide 1.8, September 1975.

6.4 TRAINING

6.4.1 A retraining and replacement training program for the facility staff shall be maintained under the direction of the Training Supervisor and shall meet or exceed the requirements and recommendations of Section 5.5 of ANSI N18.1-1971 and Appendix "A" of 10 CFR Part 55.

6.4.2 A Training program for the Emergency Squad shall be maintained under the direction of the Training Supervisor and shall meet or exceed the requirements of Section 27 of the NEPA Code-1976.

6.5 REVIEW AND AUDIT

6.5.1 ONSITE SAFETY COMMITTEE (OSC)

FUNCTION

6.5.1.1 The OSC shall function to advise the Plant Superintendent on all matters related to nuclear safety.

COMPOSITION

6.5.1.2 The OSC shall be composed of the:

Chairman:	Plant Superintendent
Member:	Operations Supervisor
Member:	Technical Supervisor
Member:	Maintenance Supervisor
Member:	Chief Engineer
Member:	Results Coordinator
Member:	Reactor Control Chemist
Member:	Senior Engineer

ALTERNATES

6.5.1.3 All alternate members shall be appointed in writing by the OSC Chairman to serve on a temporary basis; however, no more than two alternates shall participate as voting members in OSC activities at any one time.

MEETING FREQUENCY

6.5.1.4 The OSC shall meet at least once per calendar month and as convened by the OSC Chairman or his designated alternate.

QUORUM

6.5.1.5 A quorum of the OSC shall consist of the Chairman or his designated alternate and four members including alternates.

RESPONSIBILITIES

6.5.1.6 The OSC shall be responsible for:

a. Review of 1) all procedures required by Specification 6.8 and changes thereto, 2) any other proposed procedures or changes thereto as determined by the Plant Superintendent to affect nuclear safety.

b. Review of all proposed tests and experiments that affect nuclear safety.

c. Review of all proposed changes to Appendix "A" Technical Specifications.

d. Review of all proposed changes or modifications to plant systems or equipment that affect nuclear safety.

e. Investigation of all violations of the Technical Specifications including the preparation and forwarding of reports covering evaluation and recommendations to prevent recurrence to the General Superintendent of Power Stations and to the Chairman of the Offsite Review Committee.

f. Review of those REPORTABLE OCCURRENCES requiring 24 hour notification to the Commission.

g. Review of facility operations to detect potential safety hazards.

h. Performance of special reviews, investigations or analyses and reports thereon as requested by the Chairman of the Offsite Review Committee.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 17 TO FACILITY OPERATING LICENSE NO. DPR-66

DUQUESNE LIGHT COMPANY
OHIO EDISON COMPANY
PENNSYLVANIA POWER COMPANY

BEAVER VALLEY POWER STATION UNIT NO. 1

DOCKET NO. 50-334

Introduction

By letter dated October 10, 1977 and supplemented by letter dated December 22, 1977, Duquesne Light Company (DLC) submitted application for amendment to Facility Operating License No. DPR-66 for Beaver Valley Power Station (BVPS), Unit No. 1. These applications requested a change to Technical Specification 3.4.9.2 to correct an error in the specified pressurizer heat-up rate. The DLC also requested correcting of typographical errors in Table 3.7-4 (page 3/4 7-30) Safety Related Hydraulic Snubbers.

By letter dated March 20, 1978, telecopied to the NRC, DLC requested a temporary one-time extension in outage time for the 1A Low Head Safety Injection Pump. This extension was granted by telephone and confirmed by letter on March 20, 1978.

By letter dated May 23, 1978, DLC requested correction of Technical Specification 3.5.1, limits on accumulator water level and nitrogen pressure to account for instrument inaccuracies in the ECCS analysis. By letter dated May 2, 1977, DLC responded to our letter dated March 16, 1977 regarding the requirements for a Radiation Protection Manager.

Amendment 9 to License No. DPR-66 was issued on July 12, 1977. The amendment removed a requirement to use the Axial Power Distribution Monitoring System (APDMS) for in-core monitoring of axial power. The surveillance requirement (Specification 4.2.2.2, item g) should have been removed at that time. No further evaluation of this administrative oversight is required.

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By letter dated October 27, 1978, supplemented by letter dated December 14, 1978, DLC proposed to eliminate the part length control rod requirement and requested deletion of reference to the Augmented Startup Program requirements. DLC plans to remove the part length control rods.

Our evaluation of the above proposals is presented below.

Pressurizer Heatup Rate

Discussion

To account for an error, DLC requested on October 10, 1977, that the maximum pressurizer heatup rate in the Technical Specifications be changed from 200°F/hr to 100°F/hr. This change had been recommended to DLC by Westinghouse Electric Corporation (Westinghouse), the nuclear steam system supplier for BVPS, Unit No. 1.

In August 1977, Mitsubishi Heavy Industries, Ltd., of Japan, noted inconsistent pressurizer heatup rates in its Technical Specifications; one required a heatup rate of 200°F/hr, while another required 100°F/hr. This discrepancy was reported to Westinghouse, who then reviewed its analysis of the pressurizer heatup rate and determined that the correct heatup rate is 100°F/hr, and confirmed that the correct cooldown rate is 200°F/hr. Since Technical Specification 3.4.9.2a for BVPS stated that pressurizer heatup and cooldown rates were 200°F/hr, Westinghouse notified the Nuclear Regulatory Commission and DLC of this problem.

Evaluation

In designing the pressurizer, Westinghouse performed a thermal stress analysis which analyzed the fatigue resulting from a heatup rate of 100°F/hr and a cooldown rate of 200°F/hr. This analysis meets the standards of the ASME Code, Section III, which requires that the analysis be based on a usage factor. The usage factor represent the fraction of the fatigue life (the total amount of stress that a particular component is designed to handle). A usage factor of one implies that the stress exerted on the component is equal to the amount of stress that the component is designed to handle. For any piece of equipment, certain components receive more stress than others. For the pressurizer, this component is the surge nozzle, which has a usage factor of 0.9 for the design numbers listed above. This usage factor is such that if the heatup and cooldown rates used in the analysis were exceeded more than a few times, the actual surge factor for the surge nozzle would exceed 1.0. Such is not allowable under the ASME Code.

Because Technical Specification 3.4.9.2a authorized higher rates of pressurizer heatup than the correct limit, the question arose as to whether the correct limit of 100°F per hour has been exceeded in the past. Discussions with Westinghouse indicate that this is unlikely. This is because system capabilities and Technical Specification limits on the rate of reactor coolant system heatup and pressurization effectively preclude pressurizer heatup rates in excess of 50°F to 75°F per hour. Nevertheless, by letter of October 20, 1977, the NRC staff requested DLC to examine BVPS operating records and advise of any instances where the pressurizer heatup rate exceeded 100°F per hour. The pertinent data for each such occurrence including the maximum rate and the time interval during which the rate exceeded 100°F per hour was requested to determine if any corrective measures should be taken. By letter of December 22, 1977, DLC advised that a review of pre-operational tests of pressurizer heatup rate indicated that heatup rates in excess of 100°F per hour cannot be achieved at BVPS.

Accordingly, we conclude that the only action required for BVPS is modification of the Technical Specifications to reduce the limiting pressurizer heatup rate of 200°F per hour to 100°F per hour.

On September 25, 1978 Westinghouse was requested to perform an audit review of the stress analyses for components of the reactor coolant pressure boundary to assure that no similar inadvertent error appears in any other portion of the applicable Technical Specifications. By letter dated October 27, 1978 Westinghouse responded by stating that in the past year it had carefully reviewed the stress analysis inputs to the Technical Specifications for five separate plants and in addition, is completing a very careful, systematic review on the generic June 15, 1978 version of the Westinghouse Standard Technical Specifications and if any further inconsistencies surface during this review process, suitable action would be taken (in the forum of the Westinghouse STS). We find this response acceptable.

Based on our review of DLC's proposal as discussed herein, we conclude that reducing the maximum allowed heatup rate from 200°F/hr to 100°F/hr is necessary to maintain thermal stresses in the pressurizer to allowable levels. We also conclude that the maximum cooldown rate limit presently allowed by the Technical Specifications remains acceptable.

Low Head Safety Injection (LHSI) Pump 1A

Discussion

By letter dated March 20, 1978, telecopied to the NRC, DLC requested a one-time temporary change to allow an additional 24 hours to realign

the LHSI pump without shutting down the plant. The request was granted by telephone and by letter on March 20, 1978.

Evaluation

By letter dated February 15, 1978, we had required DLC to perform a 30-day endurance test to demonstrate the adequacy of the LHSI pumps at BVPS, Unit No. 1. The test was started on March 17, 1978, after disassembly of the 1A pump to install the required specific test instrumentation. However, pump shaft vibration measurements after pump reinstallation revealed excessive vibration (16 mils compared to 11 mils during prior tests) as reported by DLC on March 20, 1978.

Prior to starting the 30-day endurance test, pump 1A was run for 96 hours in the recirculation mode with steady vibration readings of 11 mils peak-to-peak. Since the completion of that phase of testing, the pump had not been run. Only preparations had been made to install additional instrumentation for the endurance test. Deep-draft pump experience has shown that whenever these pumps are disassembled great care must be exercised in reassembly so that shaft runout is maintained at a minimum value. The status of the 30-day endurance test (as of March 20, 1978) was that excessive shaft runout had caused excessive vibration which had been monitored with the proximity probe instrumentation. It was believed that misalignment of pump to motor supports and discharge piping attachment irregularities had caused these unexpectedly high vibration readings. It was further expected that when these assembly irregularities were corrected pump operation should continue with the same characteristics as were observed at the end of the 96 hour test (about 11 mils vibration). On March 21, 1978, DLC confirmed orally that when the pump test was started after realignment vibration was back to 10-11 mils.

We reviewed the information provided by DLC in its March 20, 1978 letter. The granting of an additional 24 hour period for the 1A LHSI pump realignment appeared reasonable and in the best interest of the public. DLC proposed additional surveillance testing of the diesel generator which provides emergency power for the 1B LHSI pump (the redundant pump for 1A). Such added testing gave further assurance that the LHSI system would function in the unlikely event that the system had been required.

On the basis of the information noted above and because of the short time interval (24 hours) involved to complete the realignment of the 1A LHSI pump, we concluded that the 24 hour extension was acceptable.

Accumulator Water Level and Nitrogen Pressures

Discussion

By letter dated May 23, 1978, DLC proposed to change the Technical Specification 3.5.1 on accumulator water levels and nitrogen pressures to account for instrument inaccuracies. Westinghouse had advised DLC that the technical specification values were used in the ECCS analysis and did not account for instrument inaccuracies. On the advice of Westinghouse, the BVPS Unit No. 1 has been operating with alarms more restrictive than the Technical Specification values. DLC's proposed change is to correct this oversight. In addition, changes in the Action statements and Surveillance Requirements have been made to be consistent with the latest standard Technical Specification wording. DLC has agreed to these modifications.

Evaluation

Specification 3.5.1 currently requires an accumulator water inventory of between 7510 and 7970 gallons and a nitrogen blanket pressure between 585 and 681 pounds per square inch guage (psig). To account for instrument inaccuracies, DLC proposes to reduce these allowable bands to between 7664 and 7816 gallons and between 605 and 661 psig respectively.

DLC's proposed corrections are consistent with the intent of Regulatory Guide 1.105 on Instrument Setpoints and in particular, instrument inaccuracies. Also, changes to the Specification 3.5.1 Action Statement and the 4.5.1.1 and 4.5.1.2 Surveillance Requirements have been proposed by the staff and agreed to by DLC. These changes are to achieve conformance with the latest Westinghouse Standard Technical Specifications (W-STS) being used for similar plants.

Changes in frequency with which surveillance must be performed and the type of surveillance actions required have been adjusted to provide time for more orderly and thorough planning and accomplishment of the required tasks and reduce the radiation exposure of plant personnel without a significant impact on safety.

Based on our review of DLC's proposal, we conclude that the technical specifications should be changed to account for instrument inaccuracies on accumulator water level and nitrogen pressures and that the Action statements and Surveillance Requirements changes to be consistent with the latest acceptable Standard Technical Specifications are appropriate and acceptable.

Radiation Protection Manager

Discussion

By letter dated May 2, 1977, DLC responded to our March 16, 1977 letter regarding qualifications for the Radiation Protection Manager. Regulatory Guide 1.8 dated September 1975 lists acceptable minimum qualifications for the Radiation Protection Manager. DLC has equated the Radiation Protection Manager as defined by Regulatory Guide 1.8 to the Station Reactor Control Chemist at BVPS Unit No. 1. DLC considers that the present incumbent has the adequate qualifications.

Evaluation

The minimum qualifications listed as acceptable in Regulatory Guide 1.8 requires a college degree in science or engineering with five (5) years experience in applied radiation protection, at least three (3) of which must be in nuclear power stations. The incumbent Station Reactor Control Chemist is a graduate chemist with eleven (11) years of related experience, all at nuclear power stations. Although no change in current assignments or operation is proposed by DLC, the proposed change would require subsequent assignments or replacements to meet Regulatory Guide 1.8.

We find this to be an acceptable resolution of this issue.

Axial Power Distribution Monitoring System
Safety Related Hydraulic Snubbers

These proposed changes are administrative in nature and require no safety review.

Part Length Rod Removal

Discussion and Evaluation

Part length control rods were initially installed to supress xenon induced power oscillations in the axial direction, should such oscillations occur.

The Technical Specifications, as now written, require that the part length rods (PLRs) be withdrawn and excluded from the core at all times during reactor operations. The PLRs are not needed, used or assumed

to be available to achieve required reactor shutdown conditions. Methods have been developed to allow effective control of xenon oscillations using full length control rods. The proposed removal, therefore, will not cause any change in required reactivity characteristics, or safety margins at full power, low power or shutdown.

In order to preserve the current dynamic operating characteristics of the reactor (i.e., pressure drops, coolant flow rates, etc.) which could be affected if just removal of the PLRs were to be performed, DLC proposes to install thimble plug assemblies in the spaces previously occupied by PLRs. The thimble plug assembly consists of a flat base plate with short rods suspended from the bottom surface and a spring pack assembly. The twenty-four short rods, called thimble plugs, project into the upper ends of the guide thimbles to reduce the bypass flow area. Fuel assemblies without control rods, burnable poison rods, or source rods use identical devices. Similar short rods are also used on the source assemblies and fuel assembly guide thimbles. At installation in the core, the thimble plug assemblies interface with both the upper core plate and with the fuel assembly top nozzles by resting on the adapter plate. The spring pack is compressed by the upper core plate when the upper internals assembly is lowered into place. Each thimble plug is permanently attached to the base plate by a nut which is locked to the threaded end of the plug by a small lock bar welded to the nut.

The PLR lead screws will be left in place and an anti-rotation device installed on each PLR lead screw. The anti-rotational device is constructed of 304 stainless steel.

All components in the thimble plug assembly, except for the springs, are constructed from type 304 stainless steel. The springs are wound from an aged hardened nickel base alloy for corrosion resistance and high strength.

These thimble plugs will effectively limit bypass flow through the rod cluster control guide thimbles in the fuel assemblies from which the PLRs have been removed, just as they currently limit bypass flow in those assemblies which do not contain control rods, source rods, or burnable poison rods.

Based on the considerations that the PLRs are not needed for reactor operation and that insertion of the thimble plug assemblies will preserve the current dynamic operating characteristics of the reactor, we conclude that this change is acceptable.

Augmented Startup Program

Discussion and Evaluation

The test program has been completed at the BVPS, Unit No. 1 and the data transmitted to Westinghouse. Since no further tests are required to be performed by DLC under the Augmented Startup Program, deletion of reference to this program is acceptable.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR 50.5(d)(4) that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: March 27, 1979

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-334

DUQUESNE LIGHT COMPANY
OHIO EDISON COMPANY
PENNSYLVANIA POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 17 to Facility Operating License No. DPR-66, issued to Duquesne Light Company, Ohio Edison Company, and Pennsylvania Power Company (the licensees), which revised Technical Specifications for operation of the Beaver Valley Power Station, Unit No. 1 (the facility) located in Beaver County, Pennsylvania. This amendment is effective as of the date of its issuance, except for Technical Specification change relating to the one-time extension in outage time for the 1A Low Head Safety Injection Pump which was effective on March 20, 1978.

The amendment corrects an error in the pressurizer heatup rate modifies limits on accumulator water level and gas pressure, adds requirements for a radiation protection manager, revises the list of hydraulic snubbers, deletes part length rods requirements, deletes reference to the completed augmented startup program, and deletes a residual reference to a system, APDMS, dropped in Amendment No. 9. The amendment also provides the safety evaluation for an emergency authorization granted on March 20, 1978, which allowed a one-time extension in outage time for the 1A Low Head Safety Injection Pump.

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The applications for the amendment comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

For further details with respect to this action, see (1) the applications for amendment dated October 10, 1977, December 22, 1977, March 20, 1978, May 23, 1978, May 2, 1977, October 27, 1978 and December 14, 1978, (2) Amendment No. 17 to License No. DPR-66, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Beaver Area Memorial Library, 100 College Avenue, Beaver, Pennsylvania. A copy of items (2) and (3) may be obtained upon request addressed to the

U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 27th day of March, 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

A. Schwencer

A. Schwencer, Chief
Operating Reactors Branch #1
Division of Operating Reactors