50-255

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

April 5, 1996

Mr. Richard W. Smedley
Manager Licensing
Palisades Plant
27780 Blue Star Memorial Highway
Covert, MI 49043

SUBJECT: PALISADES PLANT - ISSUANCE OF AMENDMENT RE: MISCELLANEOUS CHANGES TO THE FACILITY OPERATING LICENSE AND TECHNICAL SPECIFICATIONS (TAC NO. M93951)

Dear Mr. Smedley:

The Commission has issued the enclosed Amendment No. 171 to Facility Operating License No. DPR-20 for the Palisades Plant. The amendment consists of changes to the Technical Specifications (TS) in response to your application dated October 17, 1995.

The amendment modifies the Palisades Facility Operating License to reference 10 CFR Part 40, allow the use of source materials as reactor fuel, delete references to specific amendments and specific revisions in the listed titles of the Physical Security Plan, Suitability Training and Qualification Plan and the Safeguards Contingency Plan, and make minor editorial changes to the license. In addition, the TS are modified as follows: (1) TS 3.1.2 is modified to change the pressurizer cooldown limit from 100°F to 200°F/hour; (2) the shield cooling system requirements are relocated to the Final Safety Analysis Report; (3) several minor editorial changes and corrections are made, including corrections requested in the licensee's letter of March 24, 1995; and (4) several TS bases pages have been revised.

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

The portion of your request to delete paragraph 2.F from the Facility Operating License regarding reporting requirements has been reviewed by the NRC staff and we have concluded that your request cannot be approved. The basis for this finding is documented in the enclosed Safety Evaluation.

April 5, 1996

A copy of the Notice of Denial of Amendment to be published in the Federal Register is enclosed for your information.

Sincerely,

Janet Kennedy

Janet Kennedy, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-255

Enclosures:
1. Amendment No. 171 to DPR-20
2. Safety Evaluation
3. Notice of Denial

cc w/encl: See next page

By May 10, 1996 , the licensee may demand a hearing with respect to the denial described above. Any person whose interest may be affected by this proceeding may file a written petition for leave to intervene.

A request for hearing or petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date.

A copy of any petitions should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to Judd L. Bacon, Esquire, Consumers Power Company, 212 West Michigan Avenue, Jackson, Michigan 49201, attorney for the licensee.

For further details with respect to this action, see (1) the application for amendment dated October 17, 1995, and (2) the Commission's letter to the licensee dated April 5, 1996 .

These documents are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room located at the Van Wylen Library, Hope College, Holland, Michigan 49423.

Dated at Rockville, Maryland, this 5th day of April 1996.

For The Nuclear Regulatory Commission

Original Signed By:
Mark F. Reinhart, Acting Project Director
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

To receive a copy of this document, indicate in the box: "C" = Copy without attachment/enclosure **"E"** = Copy with attachment/enclosure **"N"** = No copy

OFFICE	LA:PD31	C	PM:PD31	C	OGC		(A)D:PD31	
NAME	CJamerson CJ		JKennedy JK		AHodgon*		MReinhart MR	
DATE	3/14/96		3/11/96		03/05/96		4/4/96	

* see previous concurrence

OFFICIAL RECORD COPY

A copy of the Notice of Denial of Amendment to be published in the Federal Register is enclosed for your information.

Sincerely,

Original Signed By:

Janet Kennedy, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-255

Enclosures: 1. Amendment No. to DPR-20
2. Safety Evaluation
3. Notice of Denial

cc w/encl: See next page

DOCUMENT NAME: G:\WPDOCS\PALISADE\PAL93951.AMD
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* see previous concurrence

OFFICE	LA:PD31	E	PM:PD31	E	*BC:PSGB		*BC:SRXB		*OGC		(A)D:PD31	
NAME	CJamerson		JKennedy JK		LCunningham		RJones		AHodgon		MReinhart	
DATE	3/14/96		3/11/96		02/23/96		02/25/96		03/05/96		3/14/96	

OFFICIAL RECORD COPY

DATED: April 5, 1996

AMENDMENT NO. 171 TO FACILITY OPERATING LICENSE NO. DPR-20-PALISADES

Docket File
PUBLIC
PDIII-1 Reading
J. Roe
J. Hannon
C. Jamerson
M. Gamberoni
OGC-WF
G. Hill (2)
C. Grimes, O-11F23
ACRS
W. Kropp, RIII
SEDB
E. Koup

cc: Plant Service list

100029

DF01

cc:

Mr. Thomas J. Palmisano
Plant General Manager
Palisades Plant
27780 Blue Star Memorial Highway
Covert, Michigan 49043

Mr. Robert A. Fenech
Vice President, Nuclear Operations
Palisades Plant
27780 Blue Star Memorial Highway
Covert, Michigan 49043

M. I. Miller, Esquire
Sidley & Austin
54th Floor
One First National Plaza
Chicago, Illinois 60603

Mr. Thomas A. McNish
Vice President & Secretary
Consumers Power Company
212 West Michigan Avenue
Jackson, Michigan 49201

Judd L. Bacon, Esquire
Consumers Power Company
212 West Michigan Avenue
Jackson, Michigan 49201

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, Illinois 60532-4351

Jerry Sarno
Township Supervisor
Covert Township
36197 M-140 Highway
Covert, Michigan 49043

Office of the Governor
Room 1 - Capitol Building
Lansing, Michigan 48913

U.S. Nuclear Regulatory Commission
Resident Inspector's Office
Palisades Plant
27782 Blue Star Memorial Highway
Covert, Michigan 49043

Nuclear Facilities and Environmental
Monitoring Section Office
Division of Radiological Health
Department of Public Health
3423 N. Logan Street
P. O. Box 30195
Lansing, Michigan 48909

Gerald Charnoff, Esquire
Shaw, Pittman, Potts and Trowbridge
2300 N Street, N. W.
Washington DC 20037

Michigan Department of Attorney
General
Special Litigation Division
630 Law Building
P.O. Box 30212
Lansing, Michigan 48909

CONSUMERS POWER COMPANY

DOCKET NO. 50-255

PALISADES PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 171
License No. DPR-20

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

   A. The application for amendment by Consumers Power Company (the licensee) dated October 17, 1995, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public; and (ii) that such activities will be conducted in compliance with the Commission's regulations;

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

   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 the license amendment and Facility Operating License No. DPR-20 is hereby amended to read as follows:

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

      A. The application for a license filed by Consumers Power Company (CPCo) complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and 10 CFR Chapter I, and all required notifications to other agencies or bodies have been duly made;

B. Construction of the Palisades Plant (the facility) has been completed in conformity with Provisional Construction Permit No. CPPR-25 and the application, as amended, the provisions of the Act, and the regulations of the Commission, and has been operating under a provisional operating license since March 24, 1971;

C. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission (except as exempted from compliance in Section 2.D. below);

D. There is reasonable assurance that the activities authorized by this Facility Operating License can be conducted without endangering the health and safety of the public;

E. CPCo is technically qualified to engage in the activities authorized by this license, as amended, in accordance with 10 CFR Chapter I;

F. CPCo has satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements";

G. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the public;

H. The issuance of this license is in accordance with 10 CFR Part 51 and all applicable requirements have been satisfied; and

I. The receipt, possession, and use of source, byproduct, and special nuclear material as authorized by this license will be in accordance with 10 CFR Parts 30, 40, and 70.

2. Provisional Operating License No. DPR-20, dated March 24, 1971 as amended, is superseded in its entirety by Facility Operating License No. DPR-20 hereby issued to Consumers Power Company (CPCo) to read as follows:

   A. This license applies to the Palisades Plant, a pressurized light water moderated and cooled reactor and electrical generating equipment (the facility). The facility is located in Van Buren County, Michigan, and is described in CPCo's Updated Final Safety Analysis Report, as supplemented and amended, and in CPCo's Environmental Report, as supplemented and amended.

   B. Subject to the conditions and requirements incorporated herein, the Commission hereby licenses:

(1) CPCo, pursuant to Section 104b of the Atomic Energy Act of 1954, as amended, and 10 CFR Part 50, "Domestic Licensing of Production and Utilization Facilities," to possess, use, and operate the facility in accordance with the limitations set forth in this license;

(2) CPCo, pursuant to the Act and 10 CFR Parts 40 and 70, to receive, possess, and use source and special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Updated Final Safety Analysis Report, as supplemented and amended;

(3) CPCo, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive possess, and use byproduct, source, and special nuclear material as sealed sources for reactor startup, reactor instrumentation, radiation monitoring equipment calibration, and fission detectors in amounts as required;

(4) CPCo, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material for sample analysis or instrument calibration, or associated with radioactive apparatus or components; and

(5) CPCo, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operations of the facility.

C. This license shall be deemed to contain and is subject to all applicable provisions of the Act; to the rules, regulations, and orders of the Commission now or hereafter in effect; and is subject to the additional conditions specified or incorporated below:

(1) CPCo is authorized to operate the facility at steady-state reactor core power levels not in excess of 2530 Megawatts thermal (100 percent rated power) in accordance with the conditions specified herein.

(2) The Technical Specifications contained in Appendix A, as revised through Amendment No. 171 , and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. CPCo shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) CPCo shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility and as approved in the

SERs dated 09/01/78, 03/19/80, 02/10/81, 05/26/83, 07/12/85, 01/29/86, 12/03/87, and 05/19/89 and subject to the following provisions:

a. CPCo may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

b. CPCo may alter specific features of the approved fire protection program provided:

   - Such changes do not result in failure to complete the fire protection program as approved by the Commission. CPCo shall maintain in auditable form, a current record of all such changes, including an analysis of the effects of the change on the fire protection program and shall make such records available to the Commission Inspectors upon request. All changes to the approved program shall be reported annually, along with the FSAR revision; and

   - Temporary changes to specific fire protection features which may be necessary to accomplish maintenance or modifications are acceptable provided interim compensatory measures are implemented.

D. The facility has been granted certain exemptions from the requirements of Section III.G of Appendix R to 10 CFR Part 50, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979." This section relates to fire protection features for ensuring the systems and associated circuits used to achieve and maintain safe shutdown are free of fire damage. These exemptions were granted and sent to CPCo in letters dated February 8, 1983, July 12, 1985, and July 23, 1985.

In addition, the facility has been granted certain exemptions from Appendix J to 10 CFR Part 50, "Primary Reactor Containment Leakage Testing for Water Cooled Power Reactors." This section contains leakage test requirements, schedules and acceptance criteria for tests of the leak-tight integrity of the primary reactor containment and systems and components which penetrate the containment. These exemptions were granted and sent to CPCo in a letter dated December 6, 1989.

These exemptions granted pursuant to 10 CFR 50.12, are authorized by law, will not present an undue risk to the public health and safety, and are consistent with the common defense and security. With these exemptions, the facility will operate, to the extent authorized herein, in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission.

E. CPCo shall fully implement and maintain in effect all provisions of the Commission-approved "Palisades Plant Physical Security Plan," "Palisades Plant Suitability Training and Qualification Plan," and "Palisades Plant Safeguards Contingency Plan," and all approved amendments. CPCo may make changes to these plans without prior Commission approval, if the changes do not decrease the safeguards effectiveness of the plans, in accordance with 10 CFR 50.54(p)(2).

F. Except as otherwise provided in the Technical Specifications or Environmental Protection Plan, the licensee shall report any violations of the requirements contained in Section 2.C of this license in the following manner: initial notification shall be made within 24 hours to the NRC Operations Center via the Emergency Notification System with written follow-up within 30 days in accordance with the procedures described in 10 CFR 50.73(b), (c), and (e).

G. CPCo shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

H. This license is effective as of the date of issuance and shall expire at midnight on March 14, 2007.

3. This license amendment is effective as of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Janet Kennedy

Janet Kennedy, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Attachments: (1) Pages 1-4 of License No. DPR-20*
(2) Changes to the Technical Specifications

Date of Issuance: April 5, 1996

* Pages 1-4 are attached, for convenience, for the composite license to reflect these changes.

ATTACHMENT TO LICENSE AMENDMENT NO. 171

FACILITY OPERATING LICENSE NO. DPR-20

DOCKET NO. 50-255

LICENSE

REMOVE	INSERT
Pages 1	Pages 1
2	2
3	3
4	4
5	-

TECHNICAL SPECIFICATIONS

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the attached pages. The revised pages are identified by amendment number and contain vertical lines indicating the areas of change.

REMOVE	INSERT
3-2	3-2
3-3	3-3
3-4	3-4
3-7	3-7
3-30	3-30
3-31	3-31
3-42	3-42
3-56	3-56
3-62	3-62
B 3.16-2	B 3.16-2
B 3.17-3	B 3.17-3
B 3.17-25	B 3.17-25
B 3.17-26	B 3.17-26
B 3.17-27	B 3.17-27
B 3.17-28	B 3.17-28
B 3.17-29	B 3.17-29
B 3.17-30	B 3.17-30
B 3.17-31	B 3.17-31
B 3.17-32	B 3.17-32
B 3.17-33	B 3.17-33
B 3.17-34	B 3.17-34
4-1	4-1
4-3	4-3
4-6	4-6
4-69	4-69
4-72	4-72
4-73	4-73
4-76	4-76
4-77	4-77
4-78	4-78
4-79	4-79
4-80	4-80
4-81	4-81
4-82	4-82
5-3	5-3

CONSUMERS POWER COMPANY

DOCKET NO. 50-255

PALISADES PLANT

FACILITY OPERATING LICENSE

License No. DPR-20

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   A. The application for a license filed by Consumers Power Company (CPCo) complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and 10 CFR Chapter I, and all required notifications to other agencies or bodies have been duly made;

   B. Construction of the Palisades Plant (the facility) has been completed in conformity with Provisional Construction Permit No. CPPR-25 and the application, as amended, the provisions of the Act, and the regulations of the Commission, and has been operating under a provisional operating license since March 24, 1971;

   C. The facility will operate in conformity with the application, as amended, the provisions of the Act, and the rules and regulations of the Commission (except as exempted from compliance in Section 2.D. below);

   D. There is reasonable assurance that the activities authorized by this Facility Operating License can be conducted without endangering the health and safety of the public;

   E. CPCo is technically qualified to engage in the activities authorized by this license, as amended, in accordance with 10 CFR Chapter I;

   F. CPCo has satisfied the applicable provisions of 10 CFR Part 140, "Financial Protection Requirements and Indemnity Agreements";

   G. The issuance of this license will not be inimical to the common defense and security or to the health and safety of the public;

   H. The issuance of this license is in accordance with 10 CFR Part 51 and all applicable requirements have been satisfied; and

   I. The receipt, possession, and use of source, byproduct, and special nuclear material as authorized by this license will be in accordance with 10 CFR Parts 30, 40, and 70.

2. Provisional Operating License No. DPR-20, dated March 24, 1971 as amended, is superseded in its entirety by Facility Operating License No. DPR-20 hereby issued to Consumers Power Company (CPCo) to read as follows:

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(3) CPCo, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive possess, and use byproduct, source, and special nuclear material as sealed sources for reactor startup, reactor instrumentation, radiation monitoring equipment calibration, and fission detectors in amounts as required;

(4) CPCo, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to receive, possess, and use in amounts as required any byproduct, source, or special nuclear material for sample analysis or instrument calibration, or associated with radioactive apparatus or components; and

(5) CPCo, pursuant to the Act and 10 CFR Parts 30, 40, and 70, to possess, but not separate, such byproduct and special nuclear materials as may be produced by the operations of the facility.

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(2) The Technical Specifications contained in Appendix A, as revised through Amendment No. 171 , and the Environmental Protection Plan contained in Appendix B are hereby incorporated in the license. CPCo shall operate the facility in accordance with the Technical Specifications and the Environmental Protection Plan.

(3) CPCo shall implement and maintain in effect all provisions of the approved fire protection program as described in the Final Safety Analysis Report for the facility and as approved in the SERs dated 09/01/78, 03/19/80, 02/10/81, 05/26/83, 07/12/85, 01/29/86, 12/03/87, and 05/19/89 and subject to the following provisions:

a. CPCo may make changes to the approved fire protection program without prior approval of the Commission only if those changes would not adversely affect the ability to achieve and maintain safe shutdown in the event of a fire.

b. CPCo may alter specific features of the approved fire protection program provided:

- Such changes do not result in failure to complete the fire protection program as approved by the Commission. CPCo shall maintain in auditable form, a current record of all such changes, including an analysis of the effects of the change on the fire protection program and shall make such records available to the Commission Inspectors upon request. All changes to the approved program shall be reported annually, along with the FSAR revision; and

- Temporary changes to specific fire protection features which may be necessary to accomplish maintenance or modifications are acceptable provided interim compensatory measures are implemented.

D. The facility has been granted certain exemptions from the requirements of Section III.G of Appendix R to 10 CFR Part 50, "Fire Protection Program for Nuclear Power Facilities Operating Prior to January 1, 1979." This section relates to fire protection features for ensuring the systems and associated circuits used to achieve and maintain safe shutdown are free of fire damage. These exemptions were granted and sent to CPCo in letters dated February 8, 1983, July 12, 1985, and July 23, 1985.

In addition, the facility has been granted certain exemptions from Appendix J to 10 CFR Part 50, "Primary Reactor Containment Leakage Testing for Water Cooled Power Reactors." This section contains leakage test requirements, schedules and acceptance criteria for tests of the leak-tight integrity of the primary reactor containment and systems and components which penetrate the containment. These exemptions were granted and sent to CPCo in a letter dated December 6, 1989.

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F. Except as otherwise provided in the Technical Specifications or Environmental Protection Plan, the licensee shall report any violations of the requirements contained in Section 2.C of this license in the following manner: initial notification shall be made within 24 hours to the NRC Operations Center via the Emergency Notification System with written follow-up within 30 days in accordance with the procedures described in 50.73(b), (c), and (e).

G. CPCo shall have and maintain financial protection of such type and in such amounts as the Commission shall require in accordance with Section 170 of the Atomic Energy Act of 1954, as amended, to cover public liability claims.

H. This license is effective as of the date of issuance and shall expire at midnight on March 14, 2007.

FOR THE NUCLEAR REGULATORY COMMISSION

Original Signed By: Thomas E. Murley

Thomas E. Murley, Director
Office of Nuclear Reactor Regulation

Attachments:
1. Appendix A - Technical Specifications
2. Appendix B - Environmental Protection Plan

Date of Issuance: February 21, 1991

Basis

When primary coolant boron concentration is being changed, the process must be uniform throughout the primary coolant system volume to prevent stratification of primary coolant at lower boron concentration which could result in a reactivity insertion. Sufficient mixing of the primary coolant is assured if one shutdown cooling or one primary coolant pump is in operation.[1] The shutdown cooling pump will circulate the primary system volume in less than 60 minutes when operated at rated capacity. By imposing a minimum shutdown cooling pump flow rate of 2810 gpm, sufficient time is provided for the operator to terminate the boron dilution under asymmetric flow conditions.[5] The pressurizer volume is relatively inactive, therefore it will tend to have a boron concentration higher than the rest of the primary coolant system during a dilution operation. Administrative procedures will provide for use of pressurizer sprays to maintain a nominal spread between the boron concentration in the pressurizer and the primary system during the addition of boron.[2]

The 57% pressurizer level, in section 3.1.1h(4), is not an analytical result, but simply a decision point between having and not having a bubble. It was chosen to agree with the maximum programmed level during power operation.

The limitation, in section 3.1.1i, on operating P-50A and P-50B together with T_c below 300°F allows the Pressure Temperature limits of Figures 3-1 and 3-2 to be higher than they would be without this limit.

The FSAR safety analysis was performed assuming four primary coolant pumps were operating for accidents that occur during reactor operation. Therefore, reactor startup above hot shutdown is not permitted unless all four primary coolant pumps are operating. Operation with three primary coolant pumps is permitted for a limited time to allow the restart of a stopped pump or for reactor internals vibration monitoring and testing.

Requiring the plant to be in hot shutdown with the reactor tripped from the C-06 panel, opening the 42-01 and 42-02 circuit breakers, assures an inadvertent rod bank withdrawal will not be initiated by the control room operator. Both steam generators are required to be operable whenever the temperature of the primary coolant is greater than the design temperature of the shutdown cooling system to assure a redundant heat removal system for the reactor.

The transient analyses were performed assuming a vessel flow at hot zero power (532°F) of 140.7 x 10^6 lb/hr minus 6% to account for flow measurement uncertainty and core flow bypass. A DNB analysis was performed in a parametric fashion to determine the core inlet temperature as a function of pressure and flow for which the minimum DNBR is equal to the DNB correlation safety limit. This analysis includes the following uncertainties and allowances: 2% of rated power for power measurement; ±0.06 for ASI measurement; ±22 psi for pressurizer pressure; ±7°F for inlet temperature; and 3% measurement and 3% bypass for core flow[4]. In addition, transient biases were included in the determination of the allowable reactor inlet temperature.

Basis (continued)

The limits of validity of the T_C equation are:

1800 ≤ pressure ≤ 2200 psia
100.0 x 10^6 ≤ Vessel Flow ≤ 150 x 10^6 lb/h
ASI as shown in the COLR.

With measured primary coolant system flow rates > 150 M lbm/hr, limiting the maximum allowed inlet temperature to the T_C LCO at 150 M lbm/hr increases the margin to DNB for higher PCS flow rates[4].

The Axial Shape Index alarm channel is being used to monitor the ASI to ensure that the assumed axial power profiles used in the development of the inlet temperature LCO bound measured axial power profiles. The signal representing core power (Q) is the auctioneered higher of the neutron flux power and the Delta-T power. The measured ASI calculated from the excore detector signals and adjusted for shape annealing (Y_I) and the core power constitute an ordered pair (Q,Y_I). An alarm signal is activated before the ordered pair exceed the boundaries specified in the COLR.

The requirement that the steam generator temperature be ≤ the PCS temperature when forced circulation is initiated in the PCS ensures that an energy addition caused by heat transferred from the secondary system to the PCS will not occur. This requirement applies only to the initiation of forced circulation (the start of the first primary coolant pump) when the PCS cold leg temperature is < 430°F. However, analysis (Reference 6) shows that under limited conditions when the Shutdown Cooling System is isolated from the PCS, forced circulation may be initiated when the steam generator temperature is higher than the PCS cold leg temperature.

References

(1) Updated FSAR, Section 14.3.2.
(2) Updated FSAR, Section 4.3.7.
(3) Deleted
(4) EMF-92-178, Revision 3, Section 15.0.7.1
(5) ANF-90-078
(6) Consumers Power Company Engineering Analysis EA-A-NL-89-14-1

3.1 PRIMARY COOL SYSTEM

Specification

3.1.2 PCS pressure, PCS temperature, and PCS heatup and cooldown rates shall be maintained within the following limits:

a. The primary coolant system (PCS) pressure shall be maintained within the limits of Figures 3-1 and 3-2.

b. The pressurizer heatup rate shall be maintained ≤ 100°F/hour**; the pressurizer cooldown rate shall be maintained ≤ 200°F/hour.

c. The primary coolant system (PCS) heatup and cooldown rates shall be maintained within the following limits:

Reactor Vessel Inlet Temperature (T)	Average Hourly Heatup Rate Limit	Average Hourly Cooldown Rate Limit
T ≤ 170°F	20°F/hour	40°F/hour
250 ≥ T > 170°F	40°F/hour	40°F/hour
350 > T > 250°F	60°F/hour **	60°F/hour
T ≥ 350°F	100°F/hour	100°F/hour

** When shutdown cooling isolation valves MO-3015 and MO-3016 are open, PCS heatup rate shall be maintained ≤ 40°F/hour and the pressurizer heatup rate shall be maintained ≤ 60°F/hour.

Applicability

Specification 3.1.2 applies at all times.

Action

a. If the limits of Specification 3.1.2 are exceeded:

1. Return to within limits within 30 minutes, and

2. Determine that the PCS condition is acceptable for continued operation within 72 hours.

b. If any action required by 3.1.2a is not met and the associated completion time has expired:

1. The reactor shall be placed in HOT SHUTDOWN within 12 hours, and

2. The reactor shall be placed in a COLD SHUTDOWN with PCS pressure less than 270 psia, within 48 hours.

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3.1 PRIMARY COOLANT SYSTEM

Basis - Pressure Temperature Limits:

The Primary Coolant System Pressure-Temperature limits are calculated for a reactor vessel wall fluence of 2.192 x 10^{19}nvt. Before the radiation exposure of the reactor vessel exceeds that fluence, Figures 3-1 and 3-2 shall be updated in accordance with the following criteria and procedure:

1. US Nuclear Regulatory Commission Regulatory Guide 1.99 Revision 2 has been used to predict the increase in transition temperature based on integrated fast neutron flux and surveillance test data. If measurements on the irradiated specimens show increase above this curve, a new curve shall be constructed such that it is above and to the left of all applicable data points.

2. Before the end of the integrated power period for which Figures 3-1 and 3-2 apply, the limit lines on the figures shall be updated for a new integrated power period. The total integrated reactor thermal power from start-up to the end of the new power period shall be converted to an equivalent integrated fast neutron exposure (E ≥ 1 MeV). Such a conversion shall be made consistent with the dosimetry evaluation of capsule W-290(12).

3. The limit lines in Figures 3-1 and 3-2 are based on the requirements of Reference 9, Paragraphs IV.A.2 and IV.A.3.

All components in the primary coolant system are designed to withstand the effects of cyclic loads due to primary system temperature and pressure changes.(1) These cyclic loads are introduced by normal unit load transients, reactor trips and start-up and shutdown operation. During unit start-up and shutdown, the rates of temperature and pressure changes are limited. A maximum plant heatup and cooldown limit of 100°F per hour for the reactor vessel and a maximum heatup limit of 100°F per hour and cooldown limit of 200°F per hour for the pressurizer are consistent with the design number of cycles and satisfies stress limits for cyclic operation.(2)

The reactor vessel plate and material opposite the core has been purchased to a specified Charpy V-Notch test result of 30 ft-lb or greater at an NDTT of + 10°F or less. The vessel circumferential weld has the highest RTNDT of plate, weld and HAZ materials at the fluence to which the Figures 3-1 and 3-2 apply.(10) The unirradiated RTNDT has been determined to be -56°F.(11) An RTNDT of -56°F is used as an unirradiated value to which irradiation effects are added. In addition, the plate has been 100% volumetrically inspected by ultrasonic test using both longitudinal and shear wave methods. The remaining material in the reactor vessel, and other primary coolant system components, meets the appropriate design code requirements and specific component function and has a maximum NDTT of +40°F.(5)

As a result of fast neutron irradiation in the beltline region of the core, there will be an increase in the RT_{NDT} with operation. The integrated fast neutron (E >1 MeV) fluxes of the reactor vessel are contained in Reference 13.

3.3.3 Prior to returning to the Power Operation Condition after every time the plant has been placed in the Refueling Shutdown Condition, or the Cold Shutdown Condition for more than 72 hours and testing of Specification 4.3.h has not been accomplished in the previous 9 months, or prior to returning the check valves in Table 4.3.1 to service after maintenance, repair or replacement, the following conditions shall be met:

a. All pressure isolation valves listed in Table 4.3.1 shall be functional as a pressure isolation device, except as specified in b. Valve leakage shall not exceed the amounts indicated.

b. In the event that integrity of any pressure isolation valve specified in Table 4.3.1 cannot be demonstrated, at least two valves in each high pressure line having a non-functional valve must be in and remain in, the mode corresponding to the isolated condition.[1]

c. If Specification a. and b. cannot be met, an orderly shutdown shall be initiated and the reactor shall be in hot shutdown condition within 12 hours, and cold shutdown within the next 24 hours.

[1] Motor-operated valves shall be placed in the closed position and power supplies deenergized.

3.3.4 Two HPSI pumps shall be operable when the PCS temperature is >325°F.

a) One HPSI pump may be inoperable provided the requirements of Section 3.3.2.c are met.

3.3.5 Two HPSI pumps shall be rendered incapable of injection into the PCS when PCS temperature is <300°F, if the reactor vessel head is installed.

Note: Specification 3.3.5 does not prohibit use of the HPSI pumps for emergency addition of makeup to the PCS.

Basis

The normal procedure for starting the reactor is, first, to heat the primary coolant to near operating temperature by running the primary coolant pumps. The reactor is then made critical by withdrawing control rods and diluting boron in the primary coolant.[1] With this mode of start-up, the energy stored in the primary coolant during the approach to criticality is substantially equal to that during power operation and, therefore, all engineered safety features and auxiliary cooling systems are required to be fully operable. During low-temperature physics tests, there is a negligible amount of stored energy in the primary coolant; therefore, an accident comparable in severity to the design basis accident is not possible and the engineered safeguards' systems are not required.

The SIRW tank contains a minimum of 250,000 gallons of water containing a minimum of 1720 ppm boron and a maximum of 2500 ppm. This is sufficient boron concentration to provide a 5% shutdown margin with all control rods withdrawn and a new core at a temperature of 60°F.

Heating steam is provided to maintain the tank above 40°F to prevent freezing. The 1.43% boron (2500 ppm) solution will not precipitate out above 32°F. The source of steam during normal plant operation is extraction steam line in the turbine cycle.

The limits for the safety injection tank pressure and volume assure the required amount of water injection during an accident and are based on values used for the accident analyses. The minimum 174-inch level corresponds to a volume of 1040 ft^3 and the maximum 200-inch level corresponds to a volume of 1176 ft^3.

Prior to the time the reactor is brought critical, the valving of the safety injection system must be checked for correct alignment and appropriate valves locked. Since the system is used for shutdown cooling, the valving will be changed and must be properly aligned prior to start-up of the reactor.

The operable status of the various systems and components is to be demonstrated by periodic tests. A large fraction of these tests will be performed while the reactor is operating in the power range. If a component is found to be inoperable, it will be possible in most cases to effect repairs and restore the system to full operability within a relatively short time. For a single component to be inoperable does not negate the ability of the system to perform its function, but it reduces the redundancy provided in the reactor design and thereby limits the

3.7 ELECTRICAL SYSTEMS (Cont'd)

a. Station power transformer 1-2 (2400 V) may be inoperable for up to 24 hours provided the operability of both diesel generators is demonstrated immediately.

b. Start-up transformer 1-2 (2400 V) may be inoperable for up to 24 hours provided the operability of both diesel generators is demonstrated immediately. Continued operation beyond 24 hours is permissible provided that a report is sent to the NRC immediately with an outline of the plans for prompt restoration of the start-up transformer and the additional precautions to be taken while the transformer is out of service, and continue operating until notified differently by the NRC.

c. 2400 V engineered safeguards bus 1C or 1D may be inoperable for up to 8 hours provided the operability of the diesel generator associated with the operable bus is demonstrated immediately and there are no inoperable engineered safety feature components associated with the operable bus.

d. 480 V distribution bus 11 or 12 may be inoperable for up to 8 hours provided there are no inoperable safety feature components associated with the operable bus.

e. MCC No. 1 and 7 or 2 and 8 may be inoperable for up to 8 hours provided there are no inoperable safety feature components associated with the operable pair of MCC.

f. 125 V d-c bus No. 1 or 2 may be inoperable for up to 8 hours provided there are no inoperable safety feature components associated with the operable bus and adequate portable emergency lighting is available during the inoperability of the No. 2 bus.

g. One of the four preferred a-c buses may be inoperable for 8 hours provided the reactor protection and engineered safety feature systems supplied by the remaining three buses are all operable.

h. One of the station batteries may be inoperable for 24 hours, providing both battery chargers on the affected bus are in operation.

i. One of the diesel generators may be inoperable for up to 7 days (total for both) during any month, provided the other diesel is started to verify operability, shutdown and the controls are left in the automatic mode, and there are no inoperable engineered safety feature components associated with the operable diesel generator.

3.11 POWER DISTRIBUTION INSTRUMENTATION

3.11.1 INCORE DETECTORS

LIMITING CONDITION FOR OPERATION

The incore detection system shall be operable:

a. With at least 160 of the 215 possible incore detectors and 2 incores per axial level per core quadrant.

b. With the incore alarming function of the datalogger operable and alarm set points entered into the datalogger.

APPLICABILITY

(1) Item a. above is applicable when the incore detection system is used for:

Measuring quadrant power tilt,
Measuring radial peaking factors,
Measuring linear heat rate (LHR), or
Determining target Axial Offset (AO) and excore monitoring allowable power level.

(2) Items a. and b. above are applicable when the incore detection system is used for monitoring LHR with automatic alarms. (Incore Alarm System.)

ACTION 1:

With less than the required number of incore detectors, do not use the system for the measuring and calibration functions under (1) above.

ACTION 2:

With the alarming function of the datalogger inoperable, do not use the system for automatic monitoring of LHR (Inoperable Incore Alarm System).

Operation may continue using the excore monitoring system as specified in 3.11.2 or by meeting the requirements of 3.23.1.

Basis

The operability of the incore detectors with the specified minimum complement of equipment ensures that the measurements obtained from use of this system accurately represent the spatial neutron flux distribution of the reactor core. The operability of the incore alarm system depends on the availability of the datalogger as well-as the operability of a minimum number of incore detectors. Incore alarm set points must be updated periodically based on measured power distributions. The incore detector Channel Check is normally performed by an on-line computer program that correlates readings with one another and with computed power shapes in order to identify inoperable detectors.

3.15 Deleted

Basis: Table 3.16 (continued)

4. Steam Generator Low Pressure - A separate Steam Generator Low Pressure (SGLP) signal is provided from each generator. The individual channel signals from each generator are combined in 2 out of 4 logic to initiate a SGLP signal for that generator. Each SGLP signal actuates closure of both Main Steam Isolation Valves (MSIVs) and closure of the feed water regulating valve and its bypass for the associated generator.

The setpoint was chosen to be low enough to avoid actuation during plant operation, but be close enough to full power operating pressure to be actuated quickly in the event of a MSLB. The setting includes a -22 psi uncertainty allowance and was the setting used in the FSAR Section 14 analysis.[2]

5. Steam Generator Low Level - The Auxiliary Feedwater Actuation Signal (AFAS) is initiated by 2 out of 4 low level signals occurring for either steam generator. The setpoint is the same as that for Reactor Trip. The setpoint was chosen to assure that Auxiliary Feedwater Flow would be initiated while the steam generator could still act as a heat sink and steam source, and to assure that a reactor trip would not occur on low level without the actuation of Auxiliary Feedwater.

6. SIRW Tank Low Level - Four SIRWT level sensors are arranged to provide two independent Recirculation Actuation Signals. Each low level sensor is powered from a separate Preferred AC bus; thus two are ultimately powered from each station battery. Each Recirculation Actuation Signal (RAS) circuit is wired with the contacts from the pair of level sensors powered from the same battery in parallel. These two parallel circuits are wired in series, producing a "1 out of 2 taken twice" logic. RAS for each train is actuated by either switch from the left battery sensing low level concurrently with either switch from the right battery. This circuit is illustrated in reference 3.

The RAS signal is actuated by separate sensors from those which provide tank level indication. The allowable range of 21" to 27" above the tank floor corresponds to 1.1% to 3.3% indicated level. Typically the actual setting is near the midpoint of the allowable range.

Each RAS actuates the valves in the injection and spray pump suction lines for the associated train switching the water supply from the SIRW tank to the containment sump for a recirculation mode of operation. The time required to reach the RAS setpoint depends on the initiating event. Following a DBA, RAS would occur after a period of approximately 20 minutes assuming all engineered safeguards pumps are operating at runout flows. The setpoint was chosen to provide adequate water in the containment sump for HPSI pump net positive suction head following an accident, but prevent the pumps from running dry during the 60 second switchover.

Basis: Instrumentation Systems 3.17 (continued)

Table B 3.17-1

Instruments Affecting Multiple Specifications

Required Instrument channels	Affected Specifications
Source Range NI-01/03 & 02/04 Count Rate Signal	3.8.e
Source Range NI-01/03 & 02/04 Count Rate Signal	3.17.6 #1
Source Range NI-01/03 Count Rate Indication @ C-150	3.17.5 #1
Wide Range NI-01/03 & 02/04 Flux level 10-4 interlock	3.17.1 #3, 4, 6, 10, & 11
Wide Range NI-01/03 & 02/04 Start-up Rate	3.17.1 #3
Wide Range NI-01/03 & 02/04 Flux Level Indication	3.17.4 #3
Wide Range NI-01/03 & 02/04 Flux Level Indication	3.17.6 #1
Power Range NI-05 - 08, Power level signal	3.17.6 #12, 15, 18
Power Range NI-05 - 08, Power level signal	3.23.1 #A.2
Power Range NI-05 - 08, Q-power	3.17.1 #2 & 4
Power Range NI-05 - 08, ASI	3.17.1 #2 & 4
Power Range NI-05 - 08, ASI	3.17.6 #16
Power Range NI-05 - 08, ASI	3.1.1 #g
Power Range NI-05 & 06; 15% interlock	3.17.1 #3 & 7
Power Range NI-05 & 06; 15% interlock	3.17.6 #18
PCS TC, Temperature signal	3.17.1 #4
PCS TC, Temperature indication	3.17.5 #6 &7
PCS TC, Q-power	3.17.1 #2 & 4
PCS TC, Q-power	3.17.6 # 14
PCS TH, Temperature indication	3.17.5 #5 & 6
PCS TH, Q-power	3.17.1 #2 & 4
PCS TH, Q-power	3.17.6 # 14
Pressurizer Pressure PI-0102 A-D, Pressure signal	3.17.1 #4 & 5
Pressurizer Pressure PI-0102 A-D, Pressure signal	3.17.2 #1.d
Pressurizer Pressure PI-0110, Pressure indication	3.17.5 #2
Steam Generator Level LI-0751 & 0752 A-D, Level Signal	3.17.1 #8 & 9
Steam Generator Level LI-0751 & 0752 A-D, Level Signal	3.17.2 #3.c & d
Steam Generator Level LI-0751A & 0752A, Level indication	3.17.5 # 10 & 11
Steam Generator Pressure PI-0751 & 0752 A-D, Pressure Signal	3.17.1 #10 & 11
Steam Generator Pressure PI-0751 & 0752 A-D, Pressure Signal	3.17.3 #3.c & d
Steam Generator Pressure PI-0751 & 0752 A-D, Pressure Indication	3.17.4 #13 & 14
Steam Generator Pressure PI-0757 & 0758 A&B, WR Pressure Indication	3.17.5 #8 & 9
Containment Pressure PS-1801, 2, 3, & 4, switch output	3.17.1 # 12
Containment Pressure PS-1801, 2, 3, & 4, switch output	3.17.3 #1.a & b

Basis: 3.17.6 Other Safety Features

The Safety Functions required by Specification 3.17.6 provide alarm and indication functions to assist the operator in monitoring plant conditions. None of the required functions provide automatic actions assumed to be available in the safety analysis, therefore, operation may continue even though the function is degraded or lost provided that the specified action is met.

The provisions of Specifications 3.0.4 and 4.0.4 are not applicable to several required instrument functions, as noted in Table 3.17.6. These instrument functions have sufficient redundancy to provide their required functions with one or more installed channels operable. The exception to 3.0.4 and 4.0.4 allows changing of plant operating conditions, but the required actions require eventual return to service.

Basis: Applicability 3.17.6

Specification 3.17.6 involves miscellaneous instruments with widely differing function. The applicability for each required instrument is provided in the Applicable Conditions column of Table 3.17.6.

Basis: Action statements 3.17.6

The listed Action is required to be completed within the specified time if the conditions of the specification are not met. If, prior to expiration of the specified completion time, the required conditions are restored, completion of the Action is not required. Each specified completion time starts at the time it is discovered that the Action statement is applicable.

Since Table 3.17.6 consists of instruments of widely different function, each table entry has its own Action statements whose numbering corresponds to that of the table entries. These Actions are discussed following the basis for the associated channel.

Basis: Table 3.17.6

1. Neutron Flux Monitoring - Two channels of neutron flux monitoring (NI-01/03 and NI-02/04) are required to be OPERABLE when there is fuel in the reactor and the flux is $\leq 10^{-4}$% RATED POWER. When flux is greater than 10^{-4}%, the requirements of Specification 3.17.1 assure adequate flux monitoring capability. Neutron flux channels are used to monitor core reactivity changes.

If only one section of a neutron flux monitoring channel, source or wide range, is functioning, the neutron flux monitoring channel may be considered OPERABLE if it is capable of detecting the existing reactor neutron flux. For example, with the source range count rate indicator functioning properly, indicating 150 cpm, in reasonable agreement with the other source range, a neutron flux monitor channel may be considered OPERABLE even though its wide range indicator is not functioning.

Basis: Table 3.17.6 (continued)

The count rate section of the wide range neutron flux monitoring channels is capable of detecting flux levels below the indicating scale. Flux levels decrease with time while the reactor is shutdown. After extended shutdowns the flux level may decrease below the indicating range. When flux is below the indication range, channel OPERABILITY may be verified by using scaler-counters or other additional instrumentation.

Action 3.17.6.1 - One or two Neutron Flux Monitoring channels inoperable - When there are fewer than two OPERABLE Neutron Flux Monitoring channels, complete monitoring of core reactivity is not possible. All positive reactivity changes must be terminated immediately, the reactor must be shutdown, if it was critical, and SHUTDOWN MARGIN verified within 4 hours and each 12 hours thereafter until the required monitoring is restored. The completion time of "immediately" does not mean "instantaneously", rather it implies "start as quickly as plant conditions permit and continue until completed."

2. Rod Position Indication - Two channels of rod position indication are required to be OPERABLE for each full-length and each part-length rod. Rod position indication is required to allow verification that the rods are positioned and aligned as required. Rod position channels are required to be OPERABLE whenever more than one CRDM is capable of rod withdrawal. It is not required when only a single rod may be withdrawn for two reasons: first, it is necessary to withdraw a rod in order to perform the calibration necessary to declare the position indication channels OPERABLE, and second, the safety analyses assume that the most reactive rod is stuck in the fully withdrawn position. Both rod position channels are calibrated to read the height of the bottom of the control rod blade, in inches, above the fully inserted position.

Primary rod position indication is provided by a gear train driven from the CRDM drive package, below the clutch. The gear train actuates cam operated limit switches and a synchro. The limit switches also actuate some of the rod position indication lights. The synchro, together with the PIP section of the plant computer, operate the digital position indication. Both the limit switches and the synchro signal are also used for alarm and control functions. The "upper electric limit" and "lower electric limit" limit switches stop the CRDM motor when the rod nears full mechanical travel of the CRDM. This provides redundant assurance that the CRDM buffer piston is not driven into the lower hard stop and that the pinion gear is not driven beyond the end of the teeth on the rack.

The primary rod position system is considered OPERABLE, for purposes of this specification, if the digital position readout or the cam operated position indication lights give positive indication of rod position.

Secondary Position Indication (SPI) is operated by a magnet integral with the connector nut and a magnetically operated reed switch stack attached to the CRDM housing. The reed switches are located at uniform intervals along the travel of the connector nut.

Basis: Table 3.17.6 (continued)

The reed switches are wired so that the voltage read across the reed switch stack is proportional to rod position. There is a dead band, near the bottom of the travel, where the CRDM housing seismic support prevents operation of the switches. SPI also provides alarms, position indication lights, and control functions based on rod position.

The SPI for each control rod is considered OPERABLE, for purposes of LCO, if there are no occurrences, other than the seismic support dead band, where two adjacent switches fail to respond to rod motion.

Action 3.17.6.2 - One rod position indication channel inoperable - If one channel of rod position indication is inoperable, control and alarm functions may also be inoperable. The position of each rod in the associated group must be verified to be within the limits of specification 3.10 within 15 minutes after moving any rod.

3. Safety Injection Refueling Water Tank Temperature - SIRWT temperature instrumentation is required to verify that the SIRWT temperature is within limits. Two channels of temperature indication are provided.

SIRWT temperature instrumentation is not required below 300°F Tave because the SIRWT and systems supported by the SIRWT are not required to be OPERABLE below 300°F Tave.

SIRWT temperature indication has been excepted from the provisions of Specifications 3.0.4 and 4.0.4 because alternate means of obtaining the required information are readily available.

Action 3.17.6.3 - One or two SIRWT temperature channels inoperable - With installed SIRWT temperature indication inoperable, operation may continue as long as temperature can be verified to be above the limit. The tank is not insulated and is accessible so alternate means of determining temperature are relatively simple. When ambient temperatures are well above the SIRWT limit, outside air temperature may be assumed to represent SIRWT temperature.

4. Main Feedwater Flow Indication - The Main Feedwater Flow measurements are necessary to perform the required daily calorimetric calculation. One feedwater flow instrument is provided for each feed line. These flow indicators are the same instruments which provide flow indication to the Feedwater Control System.

The instrumentation is not required below 15% RATED POWER where calorimetric calculations are not required.

Action 3.17.6.4 - Main Feedwater Flow indication inoperable - If feedwater flow indication is inoperable, this specification allows operation to continue if alternate indication can be provided to allow completion of the required daily calorimetric calculation. The inoperable channel must be restored to OPERABLE status prior to the next reactor startup, as required by Specification 3.0.4.

Basis: Table 3.17.6 (continued)

5. Main Feedwater Temperature Indication - The Main Feedwater Temperature measurements are necessary to perform the required daily calorimetric calculation. One feedwater temperature instrument is provided for each feed line. The instrumentation is not required below 15% RATED POWER where calorimetric calculations are not required.

Action 3.17.6.5 - Main Feedwater Temperature indication inoperable - If feedwater temperature indication is inoperable, this specification allows operation to continue if alternate indication can be provided to allow completion of the required daily calorimetric calculation. The inoperable channel must be restored to OPERABLE status prior to the next reactor startup, as required by Specification 3.0.4.

6. Auxiliary Feedwater Flow - The AFW system is arranged as two independent trains of pumps, piping, flow control valves and electrical controls. Each train is capable of feeding each steam generator through separate feed lines and flow control valves. Each AFW feed line is provided with two separate flow indication channels. One channel provides an input to the associated AFW flow control valve as well as control room flow indication; the other provides flow indication in the control room. A flow switch from each of the flow indicator channels provides a flow signal to the AFW pump sequencing circuitry. In addition, two of the flow transmitters associated with the turbine driven AFW pump, those which do not provide flow control, can be manually switched into a completely separate circuit which provides AFW flow information at Alternate Shutdown Panel C-150.

The AFW flow channels are not required to be OPERABLE when the PCS is below 300°F because the AFW system in not required to be OPERABLE below 300°F.

Action 3.17.6.6.1 - One AFW flow indicator inoperable - If one flow channel becomes inoperable, the OPERABILITY of the associated flow control valve must be determined. Those flow indication channel failures which could prevent flow through that feed line cause the valve to be inoperable. Flow indication channel failures which affect only indication, or which cause the valve to fail open do not necessarily cause the valve to be inoperable.

Action 3.17.6.6.2 - Two AFW flow indicators inoperable - If two flow indication channels for one AFW feed line become inoperable there is no way to verify flow through that line; the associated AFW flow control valve must be declared inoperable. The completion time of "immediately" does not mean "instantaneously", rather it implies "start as quickly as plant conditions permit and continue until completed."

7. PCS Leakage Detection Instrumentation - Four diverse systems for PCS leak detection are required to be OPERABLE, any one Containment Humidity Monitor, any one Containment Atmosphere Gaseous Activity Monitor, any one Containment Air Cooler Condensate Level Switch, and any one Containment Sump Level indicator. The air cooler level switch must be associated with an operating air cooler.

Basis: Table 3.17.6 (continued)

Footnotes (b) and (c) are intended to clarify that the requirement is for one instrument channel of each type to be operable, and that continued operation is not permitted unless at least one instrument channel, out of all those specified, is operable. If one OPERABLE Instrument of each type is not available, the appropriate Action statement must be followed; if no PCS leakage Detection instrument channels are operable, Action 3.17.6.21 is applicable.

The PCS leakage detection instrumentation systems are not required to be OPERABLE when the PCS temperature is below 300°F because the consequence of leakage at reduced temperature and pressure is small, and because the PCS is accessible for local inspection.

Action 3.17.6.7.1 - One required leak detection system inoperable - Operation may continue with one of the required four types of leak detection systems inoperable, but one instrument of each type must be restored to OPERABLE status prior to the next startup from COLD SHUTDOWN. Several of the instruments cannot be conveniently repaired with the plant at elevated temperature due to their location or their impact on containment integrity. Three separate leak detection systems, together with daily PCS inventory checks, are considered adequate for continued operation.

Action 3.17.6.7.2 - Two or three required leak detection systems inoperable - Daily PCS inventory calculations provide adequate leakage detection for limited periods. Thirty days is considered adequate time in which to accomplish repairs necessary to return at least three of the required instruments to operable status.

8. Primary Safety Valve Position Indication - Each Primary Safety valve is provided with two means of detecting an open or leaking valve; one acoustical monitor and one tail pipe temperature indicator.

Primary Safety Valve position indication instrumentation is not required to be OPERABLE when the PCS temperature is below 300°F because the consequence of leakage at reduced temperature and pressure is small, and because the PCS is accessible for local inspection.

Primary Safety Valve position indication has been excepted from the requirements of Specifications 3.0.4 and 4.0.4 to permit a startup from HOT SHUTDOWN with an inoperable channel. Without such an exception, no startup could be made without cooling down to repair the inoperable channel.

Action 3.17.6.8 - One Primary Safety Valve position indication channel inoperable - The Primary Safety valves are located on top of the pressurizer. During operation at elevated temperatures, the position indication is not accessible for repair. One OPERABLE channel provides sufficient capability to detect leakage for limited periods of time. The inoperable channel must be restored to OPERABLE status prior to the next start up from COLD SHUTDOWN.

Basis: Table 3.17.6 (continued)

9. Power Operated Relief Valve Position Indication - Each PORV is provided with three means of position detection; a stem position indicator, an acoustical monitor mounted on the valve, and a temperature indicator mounted on the common tailpipe. The acoustic monitors and temperature indicator provide indication of leakage through the PORV and its associated block valve.

PORV position indication is required to be OPERABLE except when the PCS is in COLD SHUTDOWN or when the PORV is isolated by a closed PORV block valve which has OPERABLE position indication. When the plant is in COLD SHUTDOWN, PORV leakage is of little consequence. When the PORV is isolated, the block valve position indication provides the needed information.

PORV position indication has been excepted from the requirements of Specifications 3.0.4 and 4.0.4 to permit a startup from HOT SHUTDOWN with an inoperable channel. Without such an exception, no startup could be made without cooling down to repair the inoperable channel.

Action 3.17.6.9 - One or two PORV position indication channels inoperable - The PORVs are located on top of the pressurizer. During operation at elevated temperatures, the position indication is not accessible for repair. One OPERABLE channel provides sufficient capability to detect leakage for limited periods of time. The inoperable channels must be restored to OPERABLE status prior to the next start up from COLD SHUTDOWN.

10. PORV Block Valve Position Indication - Each PORV block valve is provided with position indication lights operated by limit switches on the valve motor operator and by a temperature indicator mounted on the common PORV tailpipe.

PORV block valve position indication is required to be OPERABLE except when the PCS is depressurized and vented through a monitored path. The PORV block valves are required to be open at low temperatures so that the PORVs can provide Low Temperature Over Pressure protection.

PORV block Valve position indication has been excepted from the requirements of Specifications 3.0.4 and 4.0.4 to permit a startup from HOT SHUTDOWN with an inoperable channel. Without such an exception, no startup could be made without cooling down to repair the inoperable channel.

Action 3.17.6.10 - PORV Block Valve position indication inoperable - The PORV block valves are located on top of the pressurizer. During operation at elevated temperatures, the position indication is not accessible for repair. The PORV block valves are motor operated valves and are not likely to inadvertently change position. They are in series with the PORVs. During operation at elevated temperatures, when LTOP is not required, operation may continue with only one channel of position indication. When LTOP protection is required, but valve position lights are inoperable, the PCS is accessible and operation may continue if position of the block valves is verified each 12 hours. Inoperable channels must be restored to OPERABLE status prior to the next start up from COLD SHUTDOWN.

Basis: Table 3.17.6 (continued)

11. SWS Break Detector - Flow indicators measuring Service Water System flow into and out of the containment are used to actuate an alarm if in flow significantly exceeds outflow. Such a mismatch could be indicative of a cooler leak or a pipe break. The Break Detector is intended to allow identification of a leaking cooler by isolation of service water to each cooler in succession until the alarm clears. The Break Detector is strictly a maintenance aid and is intended to provide no safety function. (Ref. 12)

The SWS Break Detector is required to be OPERABLE at HOT STANDBY and above, when the containment is inaccessible for direct observation of leakage, yet the SWS is required. Specifications 3.0.4 and 4.0.4 are not applicable to the SWS break detector because it does not provide an accident related safety function.

Action 3.17.6.11 - SWS Break Detector inoperable - If the break detector is inoperable, it must be restored to OPERABLE status prior to the next startup.

12. Flux - ΔT Power Comparator - The Flux - ΔT Power comparator compares the two Q Power inputs (Excore Power Range flux and ΔT power) for that RPS channel, provides a meter indicating the difference between these inputs, and initiates an alarm if the difference exceeds a set value. Existence of a significant difference between the monitored signals indicates that either a flux tilt is developing, or that a calibration of the Excore Power Range or ΔT power circuits is required.

The Flux - ΔT Power Comparator is not required to be OPERABLE when the reactor is below 2% power because the differential temperature measurement is not meaningful at very low power levels.

Action 3.17.6.12.1 - One Flux - ΔT Power Comparator channel inoperable - With one channel inoperable, the three remaining power comparator channels are sufficient to assure that no unobserved flux tilt is developing. The inoperable channel must be restored to OPERABLE status prior to the next reactor startup.

Action 3.17.6.12.2 - Two Flux - ΔT Power Comparator channels inoperable - With two power comparator channels inoperable, power must be limited to 70% of RATED POWER to assure that no unobserved flux tilt causes local power limits to be exceeded.

13. Rod Group Sequence Control/Alarm - The Rod position indication provides two regulating rod group sequence related functions. The PIP section of the plant computer, using the signals from primary rod position indication synchros on the selected target rods, provides the actual control of group sequencing relays in addition to monitoring the target rods for the correct group position relative to the other groups. If the group position is not correct relative to the position of the other regulating groups, an Out-of-Sequence alarm is annunciated on the main control panels.

Basis: Table 3.17.6 (continued)

The SPI system - composed of the SPI input module and the Host computer - also uses the signals from the primary rod position indication synchros to monitor the target rods for the correct group position relative to the other groups. If the group position is not correct relative to the position of the other regulating groups, an Out-of-Sequence alarm is annunciated on the computer system. If a primary rod position indication synchro input card were to lose power, the corresponding reed switch position from the SPI input module would be used in the Out-of-Sequence monitoring on the SPI system. The Out-of-Sequence alarm provides assurance that the operator is aware of abnormal regulating rod positioning.

When only one control rod is capable of being withdrawn, group sequencing and Out-of-Sequence alarm provide no useful function and are not required.

Action 3.17.6.13 - Group Rod Group Sequence Control/Alarm channel inoperable - When either sequence function is inoperable, one of the methods of assuring correct control rod alignment is not available. Adequate assurance of correct rod positioning is retained by manual verification of regulating rod position after each occurrence of rod motion.

14. Concentrated Boric Acid Tank Low Level Alarm - A common "Conc Boric Acid Tank Lo Level" alarm notifies the operator that one boric acid tank is below the required total inventory. There is one level switch mounted on each tank, either of which actuates the common alarm in the control room. These two switches and the common alarm comprise the required channels.

The Concentrated Boric Acid Tank low level alarm is not required to be OPERABLE when the reactor is at HOT SHUTDOWN or below, because the inventory of boric acid is not required.

Action 3.17.6.14 - One or Two Conc Boric Acid Tank low level alarm channels inoperable - When either a boric acid tank low level alarm switch or the common alarm is inoperable, the level in the tank or tanks without an operable level alarm should be verified to be within limits each shift.

15. Excore Detector Deviation Alarm - An alarm is derived by the Excore Detector Deviation Alarm channel on excessive flux tilt. The Excore Detector Deviation Alarm compares the combined average power reading of all four Excore Power Range channels to the average from each channel, and alarms if the setpoint is exceeded. One channel being significantly different from the average could indicate a developing Quadrant Power Tilt (Tq).

The Excore Detector Deviation Alarm is required to be OPERABLE above 25% RATED POWER, when the Tq specification is applicable.

Action 3.17.6.15 - Excore Deviation Alarm inoperable - When the Excore Deviation Alarm is inoperable, continuous monitoring of Tq is unavailable. The function of Tq monitoring must be maintained by manually calculating Tq each 12 hours.

Basis: Table 3.17.6 (continued)

16. AXIAL SHAPE INDEX Alarm - The ASI Alarm Channel monitors the ASI using the Excore upper and lower detector signals as inputs and provides an alarm when ASI administrative limits are exceeded.

This alarm is only functional above a nominal 15% indicated power when the High Startup Rate trip is bypassed. It uses the High Startup Rate Pre-Trip Unit to provide the alarm function, and shares the same alarm window. It is not required to be OPERABLE below 25% RATED POWER.

Action 3.17.6.16 - One or two ASI alarm channels inoperable - The ASI alarm is one function of the Thermal Margin Monitor. Four channels are provided, but two are sufficient for ASI monitoring. If one or two channels are inoperable, they must be restored prior to the next startup from COLD SHUTDOWN.

17. Shutdown Cooling (SDC) Suction Valve Interlocks - Interlocks are provided for each SDC suction valve. These interlocks are pressure switches which prevent opening of the associated valve when PCS pressure is above the design pressure of the SDC system. One pressure switch is provided for each valve.

The interlocks are required to be OPERABLE when PCS pressure exceeds 200 psia to assure that the SDC System is not over pressurized by inadvertent opening of the suction valves at high PCS pressure.

Action 3.17.6.17 - One or two SDC suction interlocks inoperable - When an interlock is inoperable assurance that the valve will not be opened with high PCS pressure is reduced. The circuit breaker for the motor operator on the associated SDC suction valve must be racked out, except during actual operation of the valve. The extra action of having to rack in and close the breaker prior to valve operation provides protection against inadvertent valve operation.

18. Power Dependant Insertion Limit (PDIL) Alarm - PDIL Alarms are provided by both the PIP and the SPI sections of the plant computer. The SPI system is composed of the SPI input module and the Host computer. Each system monitors the position of each regulating group target rod from the primary rod position synchros and compares it to a setpoint which is a function of power level. If a primary rod position indication synchro input card were to lose power, the corresponding secondary reed switch position from the SPI input module would be used in PDIL monitoring on the SPI System. The group deviation alarms assure that the operator is aware of any group misalignment. Maintaining the rods above the PDIL, when the reactor is critical, assures that adequate SHUTDOWN MARGIN is available.

The PDIL alarm is not required at HOT SHUTDOWN and below, since no more than one control rod would be withdrawn and the SHUTDOWN Margin calculation accounts for that.

Basis: Table 3.17.6 (continued)

Action 3.17.6.18 - One PDIL alarm inoperable - With one PDIL alarm inoperable assurance of proper SHUTDOWN MARGIN is reduced. Additional assurance of proper SHUTDOWN MARGIN is provided by verification of proper group position within 15 minutes following any regulating rod motion.

19. Fuel Pool Area Radiation Monitor - The spent fuel pool is provided with two radiation monitors. These instruments provide warning of a release in the case of a fuel handling accident and provide the fuel pool criticality monitoring required by 10 CFR 70.24.

Action 3.17.6.19 - One or two Fuel Pool Area Monitors inoperable - With one or two Fuel Pool Area Radiation Monitors inoperable, fuel movement in the spent fuel pool area must be stopped. The monitor must be restored to OPERABLE status or equivalent monitoring capability provided within 72 hours. The Fuel Pool is designed to be adequately subcritical even at zero ppm boron concentration. The specified 72 hours is adequate to repair the installed instrumentation or to provide other monitoring equipment without incurring undue risk of a criticality.

20. Containment Refueling Radiation Monitors - Two radiation monitors are located in the refueling area of the containment which actuate the Containment High Radiation Logic when switched to the refueling mode. In this mode, a high level alarm on either monitor will actuate containment isolation through the associated CHR logic channel. The arrangement of these controls is illustrated in reference 7.

Action 3.17.6.20 - One or two Containment Refueling Monitors inoperable - With one or two Containment Refueling Radiation Monitors inoperable, stop REFUELING OPERATIONS in the containment. This eliminates the possibility of damaging an irradiated fuel bundle.

Action 3.17.6.21 - Required action AND associated completion time not met - If any action specified by Action statements 3.17.6.1 through 3.17.6.18 (items 19 and 20 are not associated with reactor operation) is not met AND its completion time has expired, the plant must be placed in a condition where the inoperable equipment is not required. Twelve hours are allowed to bring the plant to HOT SHUTDOWN, and 48 hours to reach conditions where the affected equipment is not required, to avoid unusual plant transients. Both the 12 and the 48 hour time periods start when it is discovered that Action 3.17.21.6 is applicable.

4.0 SURVEILLANCE REQUIREMENTS

4.0.1 Surveillance requirements shall be applicable during the reactor operating conditions associated with individual Limiting Conditions for Operation unless otherwise stated in an individual surveillance requirement.

4.0.2 Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval.

4.0.3 Failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by Specification 4.0.2, shall constitute noncompliance with the operability requirements for a Limiting Condition for Operation. The time limits of the action requirements are applicable at the time it is identified that a Surveillance Requirement has not been performed. The action requirements may be delayed for up to 24 hours to permit the completion of the surveillance when the allowable outage time limits of the action requirements are less than 24 hours. Surveillance Requirements do not have to be performed on inoperable equipment.

4.0.4 Entry into a reactor operating condition or other specified condition shall not be made unless the Surveillance Requirements associated with a Limiting Condition of Operation has been performed within the stated surveillance interval or as otherwise specified. This provision shall not prevent passage through or to plant conditions as required to comply with action requirements.

4.0.5 Surveillance Requirements for inservice inspection and testing of ASME Code Class 1, 2, and 3 components shall be applicable as follows:

a. Inservice inspection of ASME Code Class 1, 2, and 3 components and inservice testing of ASME Code Class 1, 2, and 3 pumps and valves shall be performed in accordance with Section XI of the ASME Boiler and Pressure Vessel Code and applicable Addenda as required by 10 CFR 50, Section 50.55a(g), except where specific written relief has been granted by the Commission pursuant to 10 CFR 50, Section 50.55a(g)(6)(i).

Specifications 4.0.1 through 4.0.5 establish the general requirements applicable to Surveillance Requirements. These requirements are based on the Surveillance requirements stated in the code of Federal Regulations, 10 CFR 50.36(c)(3):

> "Surveillance requirements are requirements relating to test, calibration, or inspection to ensure that the necessary quality of systems and components is maintained, that facility operation will be within safety limits, and that the limiting conditions of operation will be met."

Specification 4.0.1 establishes the requirement that surveillances must be performed during reactor operating conditions or other conditions for which the requirements of the Limiting Conditions for Operation apply, unless otherwise stated in an individual Surveillance Requirement. The purpose of this specification is to ensure that surveillances are performed to verify the operational status of systems and components and that parameters are within specified limits to ensure safe operation of the facility when the plant is in a reactor operating condition or other specified condition for which the associated Limiting Conditions for Operation are applicable. Surveillance Requirements do not have to be performed when the facility is in an operational condition for which the requirements of the associated Limiting Condition for Operation do not apply, unless otherwise specified. The Surveillance Requirements associated with a Special Test Exception are only applicable when the Special Test Exception is used as an allowable exception the requirements of a specification.

Specification 4.0.2 establishes the limit for which the specified time interval for Surveillance Requirements may be extended. It permits an allowable extension of the normal surveillance interval to facilitate surveillance scheduling and consideration of plant operating conditions that may not be suitable for conducting the surveillance; e.g., transient conditions or other ongoing surveillance or maintenance activities. It also provides flexibility to accommodate the length of a fuel cycle for surveillances that are performed at each refueling outage and are specified with an 18-month surveillance interval. It is not intended that this provision be used repeatedly as a convenience to extend the surveillance intervals beyond that specified for surveillances that are not performed during refueling outages. The limitation of Specification 4.0.2 is based on engineering judgment and the recognition that the most probable result of any particular surveillance being performed is the verification of conformance with the Surveillance Requirements. This provision is sufficient to ensure that the reliability ensured through surveillance activities is not significantly degraded beyond that obtained from the specified surveillance interval.

Specification 4.0.3 establishes the failure to perform a Surveillance Requirement within the allowed surveillance interval, defined by the provisions of Specification 4.0.2, as a condition that constitutes a failure to meet the operability requirements for a Limiting Condition for Operation. Under the provisions of this specification, systems and components are assumed to be operable when Surveillance Requirements have

4.1 OVERPRESSURE PROTECTION SYSTEM TESTS

Surveillance Requirements

In addition to the requirements of Specification 4.0.5, each PORV flow path shall be demonstrated OPERABLE by:

1. Testing the PORVs in accordance with the inservice inspection requirements for ASME Boiler and Pressure Vessel Code, Section XI, Section IWV, Category B valves.

2. Performance of a CHANNEL CALIBRATION on the PORV actuation channel at least once per 18 months.

3. When the PORV flow path is required to be OPERABLE by Specification 3.1.8.1:

   (a) Performing a complete cycle of the PORV with the plant above COLD SHUTDOWN at least once per 18 months.

   (b) Performing a complete cycle of the block valve prior to heatup from COLD SHUTDOWN, if not cycled within 92 days.

4. When the PORV flow path is required to be OPERABLE by Specification 3.1.8.2:

   (a) Performance of a CHANNEL FUNCTIONAL TEST on the PORV actuation channel, but excluding valve operation, at least once per 31 days.

   (b) Verifying the associated block valve is open at least once per 72 hours.

5. Both High Pressure Safety Injection pumps shall be verified incapable of injection into the PCS at least once per 12 hours, unless the reactor head is removed, when either PCS cold leg temperature is <300°F, or when both shutdown cooling suction valves, MO-3015 and MO-3016, are open.

Basis

With the reactor vessel head installed when the PCS cold leg temperature is less than 300°F, or if the shutdown cooling system isolation valves MO-3015 and MO-3016 are open, the start of one HPSI pump could cause the Appendix G or the shutdown cooling system pressure limits to be exceeded; therefore, both pumps are rendered inoperable.

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4.14 <u>AUGMENTED INSERVICE INSPECTION PROGRAM FOR STEAM GENERATORS</u> (Cont'd)

7. <u>Unserviceable</u> described the condition of a tube if it leaks or contains a defect large enough to affect its structural integrity in the event of an Operating Basis Earthquake, a loss-of-coolant accident, or a steam line or feedwater line break as specified in 4.14.4.c, above.

8. <u>Tube Inspection</u> means an inspection of the steam generator tube from the point of entry (hot leg side) completely around the U-bend to the top support of the cold leg.

9. <u>Preservice Inspection</u> means an inspection of the full length of each tube in steam generator performed by eddy current techniques prior to service to establish a baseline condition of the tubing. This inspection shall be performed after the shop hydrostatic test and prior to initial POWER OPERATION using the equipment and techniques expected to be used during subsequent inservice inspections.

b. The steam generator shall be determined OPERABLE after completing the corresponding actions (plug all tubes exceeding the plugging limit and all tubes containing through-wall cracks) required by Table 4.14-2.

4.14.6 <u>Reports</u>

a. Within 15 days following the completion of each inservice inspection of steam generator tubes, the number of tubes plugged in each steam generator shall be reported to the Commission in a Special Report pursuant to 10 CFR 50.4.

b. The complete results of the steam generator tube inservice inspection shall be submitted to the Commission in a Special Report pursuant to 10 CFR 50.4 within 12 months following completion of the inspection. This Special Report shall include:

   1. Number and extent of tubes inspected.

   2. Location and percent of wall-thickness penetration for each indication of an imperfection.

   3. Identification of tubes plugged

c. Results of steam generator tube inspections that fall into Category C-3 shall require 24 hour verbal notification to the NRC prior to resumption of plant operation. A written followup within the next 30 days shall provide a description of investigations and corrective measures taken to prevent recurrence.

4.16.1 b. Visual Inspection Acceptance Criteria (continued)

A review and evaluation shall be performed and documented to justify continued operation with an unacceptable snubber. If continued operation cannot be justified, the snubber shall be declared inoperable and the action requirements shall be met.

c. Functional Tests

At least once per 18 months during shutdown, a representative sample (10% of the total safety-related snubbers in use at the plant) shall be functionally tested either in place or in a bench test. The test shall verify the snubber has freedom of movement and is not frozen up. For each snubber which did not meet the functional test acceptance criteria of Specification 4.16.1.d or 4.16.1.e, an additional 10% of the total shall be functionally tested.[1]

The representative sample selected for functional testing shall include the various configurations, operating environments and the range of size and capacity of snubbers. Snubbers identified as especially difficult to remove or in high radiation zones during shutdown shall also be included in the representative sample.[2]

In addition to the regular sample, snubbers which failed the previous functional test shall be retested during the next test period. If a spare snubber has been installed in place of a failed snubber, then both the failed snubber (if it is repaired and installed in another position) and the spare snubber shall be retested. Test results of these snubbers may not be included for the resampling.

If any snubber selected for functional testing either fails to lockup or fails to move, i.e., frozen in place, the cause will be evaluated and, if caused by manufacturer or design deficiency, all snubbers of the same design subject to the same defect shall be functionally tested. This testing requirement shall be independent of the requirements stated above for snubbers not meeting the functional test acceptance criteria.

[1] Snubbers of rated capacity greater than 50,000 pounds are not to be included when defining the total number of safety-related snubbers in use at the plant.

[2] Permanent or other exemptions from functional testing for individual snubbers may be granted by the Commission only if a justifiable basis for exemption is presented and/or snubber life destructive testing was performed to qualify snubber operability for all design conditions at either the completion of their fabrication or at a subsequent date.

4.16.1 c. Functional Tests (continued)

Snubbers of rated capacity greater than 50,000 pounds will be functionally tested in lots comprising 25% of their total during each refueling outage. In the event of one snubber failure out of the four tested, no additional snubbers will be tested provided the problem is non-generic. For each additional snubber failure, however, two additional snubbers will be tested until no further snubber failures are identified or all snubbers have been tested. Generic failures will be handled as the specific circumstances require.

For the snubber(s) found inoperable, an engineering evaluation shall be performed on the components which are suppressed by the snubber(s). The purpose of this engineering evaluation shall be to determine if the components suppressed by the snubber(s) were adversely affected by the inoperability of the snubber(s) in order to ensure that the suppressed component remains capable of meeting the designed service.

d. Hydraulic Snubbers Functional Test Acceptance Criteria

The hydraulic snubber functional test shall verify that:

1. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

2. Snubber bleed, or release rate, where required, is within the specified range in compression or tension. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

e. Mechanical Snubbers Functional Test Acceptance Criteria

The mechanical snubber functional test shall verify that:

1. The force that initiates free movement of the snubber rod in either tension or compression is less than the specified maximum drag force (break away friction).

2. Activation (restraining action) is achieved within the specified range of velocity or acceleration in both tension and compression.

3. Snubber release rate, where required, is within the specified range in compression or tension. For snubbers specifically required not to displace under continuous load, the ability of the snubber to withstand load without displacement shall be verified.

4.17 INSTRUMENTATION SYSTEMS TESTS

Table 4.17.1

Instrumentation Surveillance Requirements for Reactor Protective System

	Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1.	Manual Trip	NA	(a)	NA
2.	Variable High Power	12 hours	31 days	(b, c, & d)
3.	High Start Up Rate	12 hours	(a)	18 months[e]
4.	Thermal Margin/ Low Pressure	12 hours	31 days	18 months
5.	High Pressurizer Pressure	12 hours	31 days	18 months
6.	Low PCS Flow	12 hours	31 days	18 months
7.	Loss of Load	NA	(a)	18 months
8.	Low "A" SG Level	12 hours	31 days	18 months
9.	Low "B" SG Level	12 hours	31 days	18 months
10.	Low "A" SG Pressure	12 hours	31 days	18 months
11.	Low "B" SG Pressure	12 hours	31 days	18 months
12.	High Containment Pressure	NA	31 days	18 months
13.	RPS Matrix Logic	NA	31 days	NA
14.	RPS Initiation Logic	NA	31 days	NA

15. Thermal Margin Monitor; Verify constants each 92 days.

(a) Once within 7 days prior to each reactor startup.

(b) Calibrate with Heat Balance each 24 hours, when > 15% RATED POWER.

(c) Calibrate Excores channels with test signal each 31 days.

(d) CHANNEL CALIBRATION each 18 months.

(e) Include verification of automatic Zero Power Mode Bypass removal.

4.17 INSTRUMENTATION SYSTEMS TESTS

Table 4.17.2

Instrumentation Surveillance Requirements for Engineered Safety Features

	Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1.	Safety Injection Signal (SIS)			
a.	Manual Initiation	NA	18 months	NA
b.	SIS Logic (Initiation, Actuation, and low pressure block auto reset)	NA	(a)	NA
c.	CHP Signal SIS initiation (5P Relay Output)	NA	18 months	NA
d.	Pressurizer Pressure Instrument Channels	12 hours	31 days	18 months
2.	Recirculation Actuation Signal (RAS)			
a.	Manual Initiation	NA	18 months	NA
b.	RAS Logic	NA	18 months	NA
c.	SIRWT Level Switches	NA	18 months	18 months
3.	Auxiliary Feedwater Actuation Signal (AFAS)			
a.	Manual Initiation	NA	18 months	NA
b.	AFAS Logic	NA	92 days	NA
c.	"A" SG Level	12 hours	31 days	18 months
d.	"B" SG Level	12 hours	31 days	18 months
4.	Emergency Power Sequencers			
a.	DBA Sequencer	NA	92 days	18 months
b.	Normal Shutdown Sequencer	NA	18 Months	18 months

(a) Test normal and emergency power functions using test circuits each 92 days. Verify all automatic actuations and automatic resetting of low pressure block each 18 months.

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Table 4.17.3

Instrumentation Surveillance Requirements for Isolation Functions

Functional Unit	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1. Containment High Pressure (CHP)			
a. CHP logic Trains	NA	18 months	NA
b. Containment Pressure Switches - Left Train	NA	31 days	18 months
c. Containment Pressure Switches - Right Train	NA	31 days	18 months
2. Containment High Radiation (CHR)			
a. Manual Initiation	NA	18 months	NA
b. CHR Logic Trains	NA	18 months	NA
c. Containment Area Radiation Monitors	12 hours	31 days	18 months
3. Steam Generator Low Pressure (SGLP)			
a. Manual Actuation	NA	18 months	NA
b. SGLP Logic Trains	NA	18 months	NA
c. "A" Steam Generator Pressure	12 hours	31 days	18 months
d. "B" Steam Generator Pressure	12 hours	31 days	18 months
4. Engineered Safeguards Pump Room High Radiation			
a. East Room Monitor	12 hours	31 days	18 months
b. West Room Monitor	12 hours	31 days	18 months

Table 4.17.4

Instrumentation Surveillance Requirements for Accident Monitoring

	Instrument	CHANNEL CHECK	CHANNEL CALIBRATION
1.	Wide Range T_H	31 days	18 months
2.	Wide Range T_C	31 days	18 months
3.	Wide Range Flux	31 days	18 months
4.	Containment Floor Water Level	31 days	18 months
5.	Subcooled Margin Monitor	31 days	18 months
6.	Wide Range Pressurizer Level	31 days	18 months
7.	Containment H_2 Concentration	31 days	18 months
8.	Condensate Storage Tank Level	31 days	18 months
9.	Wide Range Pressurizer Pressure	31 days	18 months
10.	Wide Range Containment Pressure	31 days	18 months
11.	Wide Range "A" SG Level	31 days	18 months
12.	Wide Range "B" SG Level	31 days	18 months
13.	Narrow Range "A" SG Pressure	31 days	18 months
14.	Narrow Range "B" SG Pressure	31 days	18 months
15.	Position Indication for each Containment Isolation Valve	31 days	18 months
16.	Core Exit Thermocouples (CET) Quadrant 1	31 days	18 months[a]
17.	Core Exit Thermocouples (CET) Quadrant 2	31 days	18 months[a]
18.	Core Exit Thermocouples (CET) Quadrant 3	31 days	18 months[a]
19.	Core Exit Thermocouples (CET) Quadrant 4	31 days	18 months[a]
20.	Reactor Vessel Water Level (RVWL)	31 days	18 months
21.	High Range Containment Radiation	31 days	18 months

(a) Calibrate by substituting a known voltage for thermocouple.

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4.17 INSTRUMENTATION SYSTEMS TESTS

Table 4.17.5

Instrumentation Surveillance Requirements for Alternate Shutdown System

	Instrument or Control	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1.	Start-up Range Flux	(a)	(a)	18 months
2.	Pressurizer Pressure	92 days	NA	18 months
3.	Pressurizer Level	92 days	NA	18 months
4.	#1 Hot Leg Temperature	92 days	NA	18 months
5.	#2 Hot Leg Temperature	92 days	NA	18 months
6.	#1 Cold Leg Temperature	92 days	NA	18 months
7.	#2 Cold Leg Temperature	92 days	NA	18 months
8.	"A" SG Pressure	92 days	NA	18 months
9.	"B" SG Pressure	92 days	NA	18 months
10.	"A" SG Level	92 days	NA	18 months
11.	"B" SG Level	92 days	NA	18 months
12.	SIRW Tank Level	92 days	NA	18 months
13.	P-8B Flow to "A" SG	18 months	18 months	18 months
14.	P-8B Flow to "B" SG	18 months	18 months	18 months
15.	P-8B Low Suction Alarm	NA	18 months	18 months
16.	P-8B Steam Valve Control	NA	18 months	NA
17.	AFW Flow Control "A" SG	NA	18 months	NA
18.	AFW Flow Control "B" SG	NA	18 months	NA
19.	Transfer Switches, C-150	NA	18 months	NA
20.	Transfer Switch, C-150A	NA	18 months	NA

(a) Once within 7 days prior to each reactor startup.

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Table 4.17.6

Instrumentation Surveillance Requirements for Other Safety Functions

	Instrument	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
1.	Neutron Flux Monitoring	12 hours	(a)	18 months
2.	Rod Position Indication	12 hours	(b)	18 months
3.	SIRW Tank Temperature	12 hours	NA	18 months
4.	Main Feedwater Flow	12 hours	Not Required	18 months
5.	Main Feedwater Temp.	12 hours	Not Required	18 months
6.	AFW Flow Indication	12 hours	18 months	18 months
7.	PCS Leakage Detection:			
	a. Sump Level	12 hours	18 months	18 months
	b. Atmos. Gas Monitor	12 hours	18 months	18 months
	c. Humidity Monitor	12 hours	18 months	18 months
	d. Air Cooler Condensate Flow Switch	NA	18 months	Not Required
8a.	Primary Safety Valve acoustical monitor	NA	18 months	18 months
8b/ 9a.	Safety Valve / PORV[c] tailpipe temperature	12 hours	18 months	18 months
9b.	PORV Acoustical Monitor	NA	18 months	18 months
9c.	PORV Stem Position	12 hours	18 months	18 months
10.	PORV Block Valve Position Indication	12 hours	NA	18 months

(a) Once within 7 days prior to each reactor startup.

(b) Verification of Regulating Rod Withdrawal and Shutdown Rod Insertion interlocks OPERABILITY only, once within 92 days prior to each reactor startup AND once prior to startup after each refueling.

(c) The tailpipe temperature indicator is common to the safety valves and PORVs

(continued)

Table 4.17.6 (continued)

Instrumentation Surveillance Requirements for Other Safety Functions

	Instrument	CHANNEL CHECK	CHANNEL FUNCTIONAL TEST	CHANNEL CALIBRATION
11.	SWS Break Detector	NA	18 months	18 months
12.	Flux - ΔT Comparator	12 hours	31 days	18 months
13.	Rod Group Sequence Control/Alarm	NA	18 months	18 months
14.	BAT Low Level Alarm	NA	18 months	Not Required
15.	Excore Deviation Alarm	NA	18 months	18 months
16.	ASI Alarm	NA	18 months	18 months
17.	SDC Suction Interlocks	NA	18 months	18 months
18.	PDIL Alarm	NA	31 days[d]	18 months
19.	Fuel Pool Rad Monitor	24 hours	31 days	18 months
20.	Containment Refueling Radiation Monitor	24 hours	31 days	18 months

(d) Setpoint verification only.

Amendment No. ~~162~~, ~~164~~, 171

5.3 NUCLEAR STEAM SUPPLY SYSTEM (NSSS) (Continued)

5.3.2 Reactor Core and Control

a. The reactor core shall approximate a right circular cylinder with an equivalent diameter of about 136 inches and an active height of about 132 inches.

b. The reactor core shall consist of approximately 43,000 Zircaloy-4 clad fuel rods containing depleted, natural, or slightly enriched uranium in the form of sintered UO_2 pellets. The fuel rods shall be grouped into 204 assemblies. A core plug or plugs may be used to replace one or more fuel assemblies subject to the analysis of the resulting power distribution.

c. The fully loaded core shall contain approximately 211,000 pounds UO_2 and approximately 56,000 pounds of Zircaloy-4. Poison may be placed in the fuel bundles for long-term reactivity control.

d. The core excess reactivity shall be controlled by a combination of boric acid chemical shim, cruciform control rods, and mechanically fixed absorber rods where required. Forty-five control rods shall be distributed throughout the core as shown in Figure 3-2 of the FSAR. Four of these control rods may consist of part-length absorbers.

5.3.3 Emergency Core Cooling System

An emergency core cooling system shall be installed consisting of various subsystems each with internal redundancy. These subsystems shall include four safety injection tanks, two high-pressure and two low-pressure safety injection pumps, a safety injection and refueling water storage tank, and interconnecting piping as shown in Section 6 of the FSAR.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 171 TO FACILITY OPERATING LICENSE NO. DPR-20

CONSUMERS POWER COMPANY

PALISADES PLANT

DOCKET NO. 50-255

1.0 INTRODUCTION

By letter dated October 17, 1995, the Consumers Power Company (the licensee) requested an amendment to Facility Operating License No. DPR-20 for the Palisades Plant to reference 10 CFR Part 40, allow the use of source materials as reactor fuel, delete references to specific amendments and specific revisions in the listed titles of the Physical Security Plan Suitability Training and Qualification Plan and the Safeguards Contingency Plan, delete paragraph 2.F on reporting requirements, and make minor editorial changes. In addition, the licensee has proposed changes to the Technical Specifications (TS) as follows: (1) TS 3.1.2 would be modified to change the pressurizer cooldown limit from 100°F to 200°F/hour; (2) the shield cooling system requirements would be relocated to the Palisades Final Safety Analysis Report (FSAR); (3) several minor editorial changes to various sections of the TS are proposed; and (4) revisions to several TS bases pages are proposed.

2.0 EVALUATION

2.1 Proposed Change to Facility Operating License Paragraph 2.B.(2)

The licensee has proposed a change to paragraph 2.B.(2) of the Facility Operating License to allow the use of source materials as reactor fuel. Specifically, paragraph 2.B.(2) would be changed to read:

> CPCo, pursuant to the Act and 10 CFR Parts 40 and 70, to receive, possess, and use source and special nuclear material as reactor fuel, in accordance with the limitations for storage and amounts required for reactor operation, as described in the Updated Final Safety Analysis Report, as supplemented and amended;

Changing paragraph 2.B.(2) to add reference to 10 CFR Part 40 and to allow the use of source materials as reactor fuel would allow the use of depleted or natural uranium in addition to using slightly enriched uranium which is currently allowed. The licensee has stated that the use of depleted or natural uranium in future core designs would result in enhanced fuel economy and reduced neutron leakage.

One potential use of depleted or natural uranium would be selective fuel rod loading in the axial direction. Small segments at the top and bottom of selected fuel rods would be loaded with depleted or natural uranium. The remainder of the fuel rod would be loaded with slightly enriched uranium. The reduced enrichment zones at the top and bottom of the core are referred to as "axial blankets." The axial blankets can help reduce fuel cost by optimizing the core power distribution.

Another potential use of depleted or natural uranium would be selective fuel rod loading in the radial direction. Fuel rods near the exterior of the core would be loaded with depleted or natural uranium to reduce neutron leakage to the reactor vessel. The depleted uranium content acts like a shield compared to a fuel rod with an enriched loading.

The NRC staff has reviewed the licensee's proposed change to paragraph 2.B.(2) and finds it acceptable. The reference to 10 CFR Part 40 is necessary to allow the licensee to use source materials. Source material by definition does not include special nuclear material, and therefore needs to be identified separately in this paragraph from special nuclear material. The licensee's potential use of depleted or natural uranium is acceptable because it does not increase the probability or consequences of a previously analyzed accident, create the potential for a new kind of accident or reduce the margin of safety.

2.2 <u>Proposed Change to Facility Operating License 2.E</u>

The licensee has proposed to remove references to specific revisions in the plan titles included in paragraph 2.E to remove the implication that the Facility Operating License must be amended when any subject plan revision is approved. Paragraph 2.E still requires the licensee to implement and maintain in effect all provisions of the Commission-approved Palisades Plant Physical Security Plan, Suitability Training and Qualification Plan, and the Plant Safeguards Contingency Plan. Paragraph 2.E is also reworded to explicitly require compliance with all approved amendments. A sentence is added to clarify that changes which do not decrease the safeguards effectiveness of the plans may be made in accordance with 10 CFR 50.54(p)(2).

These changes are administrative in nature and serve to clarify the intent of paragraph 2.E. These changes are therefore acceptable.

2.3 <u>Minor Editorial Changes to the Facility Operating License</u>

The licensee has proposed the following editorial changes to the Facility Operating License:

2.3.1 Consistent abbreviations have been used throughout the license; "the Commission" for the NRC and "CPCo" for the licensee. The license currently contains both "the Commission" and "the NRC" and both "CPCo" and "the licensee." This change is proposed for consistency and brevity within the license.

2.3.2 Punctuation of series has been made consistent with the recommendations of NUREG-1379, "NRC Editorial Style Guide." Some series within the license included a comma before the final element; others did not. This change is proposed for consistency.

2.3.3 Where two different styles of writing are used to imply the same meaning the more concise wording had been used. For example, several paragraphs use the words "in accordance with the Commission's regulations in 10 CFR 70," while others use the more concise "pursuant to 10 CFR 70." The more concise wording has been proposed in paragraphs where this occurs.

2.3.4 Subparagraph 1.D(ii) has been deleted. It is redundant to Paragraph 1.C.

2.3.5 The location of the facility has been deleted from 2.B.(1). The location is specified in Paragraph 2.A.

2.3.6 Paragraphs 2.C.(1), 2.C.(2), and 2.E are changed to remove the titles, "Maximum Power Level," "Technical Specifications," and "Physical Protection." Removing the titles from Paragraph 2.C.(1), 2.C.(2), and 2.E will result in a more consistent format for the license.

2.3.7 The first subparagraph listed after 2.C.(3)b has been deleted. It is redundant to 10 CFR 50.59.

The above changes are editorial only and are therefore acceptable.

2.4 <u>Deletion of Paragraph 2.F. from the Facility Operating License</u>

Paragraph 2.F. states, in part, "Except as otherwise provided in the Technical Specifications or Environmental Protection Plan, the licensee shall report any violations of the requirements contained in Section 2.C of this license..." The requirements included in Section 2.C include: all regulations in 10 CFR Chapter I, all applicable provisions of the Act, all rules, regulations and orders of the Commission, all requirements of the Technical Specifications, all requirements of the Environmental Protection Plan, and all provisions of the Fire Protection Plan. In addition, paragraph 2.C states in part that "The licensee is authorized to operate the facility at steady-state reactor core power levels not in excess of 2530 Megawatts thermal (100 percent rated power) in accordance with the conditions specified herein." The current paragraph 2.F. requires 24-hour reporting and 30-day written follow-up of any violation of those requirements included in license paragraph 2.C. The licensee is proposing to delete paragraph 2.F because it is contradictory to the reporting requirements spelled out in 10 CFR 50.72 and 50.73. The licensee has stated that compliance with 10 CFR 50.72 and 50.73 is required by paragraph 2.C. which makes paragraph 2.F. redundant and unnecessary.

The staff has reviewed the proposed change to delete paragraph 2.F from the Facility Operating License and finds the change unacceptable. Although some of the reporting requirements contained in paragraph 2.F may be redundant to those required by 10 CFR 50.72 and 50.73, there is one requirement in paragraph 2.F that is not found in 10 CFR 50.72 and 50.73. Specifically, if the licensee were to operate the Palisades plant in excess of 2530 Megawatts thermal, the current paragraph 2.F would require that this violation be reported to the NRC Operations Center within 24 hours. If paragraph 2.F were deleted, the licensee would not be required to report the overpower event because neither 10 CFR 50.72 nor 50.73 contain reporting requirements specific to a licensee exceeding 100 percent rated power. Also, the Palisades TS do not require a report to the NRC if the power level were to exceed 100 percent. Because of this, the staff has determined that deletion of paragraph 2.F cannot be approved.

2.5 Technical Specifications and Basis Changes

2.5.1 Proposed Change to TS 3.1.2b

The pressurizer cooldown rate was changed by Amendment 163 to the Palisades TS. Amendment 163 reduced the pressurizer heatup and cooldown rate limit from 200°F/hour to 100°F/hour to address an inconsistency between the heatup rate assumed in the pressurizer stress analysis and the pressurizer heatup rate limit in the TS. The licensee has stated that at the time the changes made by Amendment 163 were proposed, it was not realized that a 100°F/hour cooldown rate might become limiting under any anticipated operating conditions, so it was proposed to simply change the combined heatup and cooldown limit from 200°F/hour to 100°F/hour. During implementation of Amendment 163, the licensee noted that the new cooldown rate limit would unnecessarily restrict the rate of primary coolant system depressurization following a steam generator tube rupture.

Prior to Amendment 163, the pressurizer cooldown rate was 200°F/hour. The proposed change to TS 3.1.2b would separate the limits for the heatup and cooldown rates, returning the specified cooldown rate to the original value which is consistent with the plant design. The current heatup rate would be retained. Changing TS 3.1.2b to make the pressurizer cooldown rate ≤200°F/hour is therefore acceptable.

2.5.2 Proposed Changes to TS 3.1.2c

TS 3.1.2c. is being modified to add the word "shall" and to add "Average Hourly" to the column headings for the heatup and cooldown rate limits. Amendment 163 modified the wording of TS 3.1.2c. and inadvertently left out the word "shall." This change will restore the intended wording. The addition of "Average Hourly" to the temperature limit column headings is intended to clarify the intent of the requirement. Amendment 163 changed the wording of TS 3.1.2c. but did not keep the words that referred to the average heatup or cooldown rate in any one hour. This change will clarify the intent of TS 3.1.2c. which is to maintain the heatup and cooldown rates within an average hourly limit.

2.5.3 Proposed Deletion of TS 3.15

Section 182a of the Atomic Energy Act (the "Act") requires applicants for nuclear power plant operating licenses to include TS as part of the license. The Commission's regulatory requirements related to the content of TS are set forth in 10 CFR 50.36. That regulation requires that the TS include items in five specific categories, including (1) safety limits, limiting safety system settings and limiting control settings; (2) limiting conditions for operation; (3) surveillance requirements; (4) design features; and (5) administrative controls. However, the regulation does not specify the particular requirements to be included in a plant's TS.

The Commission has provided guidance for the contents of TS in its "Final Policy Statement on Technical Specifications Improvements for Nuclear Power Reactors" ("Final Policy Statement"), 58 FR 39132 (July 22, 1993), in which the Commission indicated that compliance with the Final Policy Statement satisfies Section 182a of the Act. In particular, the Commission indicated that certain items could be relocated from the TS to licensee-controlled documents, consistent with the standard enunciated in *Portland General Electric Co.* (Trojan Nuclear Plant), ALAB-531, 9 NRC 263, 273 (1979). In that case, the Atomic Safety and Licensing Appeal Board indicated that "technical specifications are to be reserved for those matters as to which the imposition of rigid conditions or limitations upon reactor operation is deemed necessary to obviate the possibility of an abnormal situation or event giving rise to an immediate threat to the public health and safety."

Consistent with this approach, the Final Policy Statement identified four criteria to be used in determining whether a particular matter is required to be included in the TS. These criteria were subsequently incorporated into the regulations by an amendment to 10 CFR 50.36, 60 FR 36953 (July 19, 1995). The criteria incorporated into the rule are as follows:

(1) Installed instrumentation that is used to detect, and indicate in the control room, a significant abnormal degradation of the reactor coolant pressure boundary;

(2) a process variable, design feature, or operating restriction that is an initial condition of a design-basis accident or transient analysis that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;

(3) a structure, system, or component that is part of the primary success path and which functions or actuates to mitigate a design-basis accident or transient that either assumes the failure of or presents a challenge to the integrity of a fission product barrier;

(4) a structure, system, or component which operating experience or probabilistic safety assessment has shown to be significant to public health and safety.

As a result, existing TS requirements which fall within or satisfy any of the criteria must be retained in the TS, while those TS requirements which do not fall within or satisfy these criteria may be relocated to other, licensee-controlled documents.

The licensee is proposing to delete TS 3.15, Reactor Primary Shield Cooling System, because it is not safety-grade, does not contribute to plant response to any accident or transient, is not used as a success path in any of the emergency operating procedures, and its ability to function, or failure to function, has no effect on any result in the Palisades probabilistic risk assessment. The licensee is proposing to relocate relevant information to the Palisades FSAR. The licensee has stated that the shield cooling system functional requirements are discussed in the revisions to the FSAR.

Accordingly, the staff has concluded that the requirements for the reactor primary shield cooling system do not meet the 10 CFR 50.36 criteria. Therefore, the deletion of TS 3.15 and the incorporation of the shield cooling system functional requirements into the FSAR is acceptable.

2.5.4 <u>Proposed Change to TS 4.0.2</u>

In accordance with NRC Generic Letter (GL) 89-14, "Line-Item Improvements in Technical Specifications - Removal of the 3.25 Limit on Extending Surveillance Intervals," the licensee is proposing to change TS 4.0.2 to delete the "3.25 times" limit for the performance of surveillance requirements. TS 4.0.2 will now read, "Each Surveillance Requirement shall be performed within the specified surveillance interval with a maximum allowable extension not to exceed 25 percent of the specified surveillance interval." The basis for TS 4.0.2 has also been changed to reflect the change in wording. Both of these changes are proposed in accordance with GL 89-14, and are therefore acceptable.

2.5.5 <u>Administrative TS Changes</u>

The following changes have been proposed by the licensee and are administrative in nature:

1. TS 3.3.3c. is moved from page 3-31 to 3-30. This change is necessary because when newly numbered TS 3.3.4 and 3.3.5 were issued by Amendment 163, these new items were inadvertently located between TS 3.3.3b. and 3.3.3c. which was located on the following page. This change corrects the placement of TS 3.3.3c. and addresses the licensee's request for corrections to this TS dated March 24, 1995.

2. Amendment 88 added a note to allow a one-time only diesel generator allowed outage time extension to 10 days during May 1985. Amendment 164 added several notes allowing a one-time deferral of several surveillance requirements during Cycle 11. The licensee is proposing to delete these notes because they are no longer valid and only serve to clutter the TS.

3. The references to TS 6.9.3.3b. on page 4-69 have been changed to reference 10 CFR 50.4. TS 6.9.3.3 had been renumbered to 6.9.4 by Amendment 154 and TS 6.9.4b. references 10 CFR 50.4. This change only serves to clarify the reference and make it consistent with TS 6.9.4b.

4. TS 5.3.2b. is being modified to include the use of depleted and natural uranium in addition to the currently specified "slightly enriched uranium." This change is being made to reflect the change to the Palisades Facility Operating License discussed above.

5. The FSAR figure referenced in Design Features Section 5.3.2d. is

incorrect. When the Updated FSAR was produced, the subject figure was revised to remove fuel designations applicable only to the initial core and its number was changed from 3-5 to 3-2. The TS reference to that figure was not updated. This change corrects that omission.

The staff has reviewed the above changes and found them administrative in nature, serving only to clarify or make the TS consistent. Therefore, these changes are acceptable.

2.5.6 Changes to the TS Bases

The following Basis changes were proposed and are acceptable. The revised Bases pages have been included with this amendment.

1. The basis for Specification 3.1.1g. has been changed to reflect the 22 psi uncertainty used in the verification of the T_c equation for the Cycle 12 Disposition of Events Report, and to correct a reference to the Core Operating Limits Report by Amendment 169. The reference to T_{inlet} has been corrected from a previous Basis change to read T_c.

2. The basis for TS 3.11.1 has been changed to reflect the capability of the new plant computer to perform a channel check of the incore instruments on-line rather than off-line as was formerly done. The new plant computer was recently installed during the 1995 refueling outage.

3. The basis for TS 3.16 regarding the safety injection and refueling water (SIRW) tank low level actuation of the recirculation actuation signal (RAS) has been changed to clarify the conditions under which the RAS could occur in as little as 20 minutes. In addition, a typographical error was corrected in item 4 to delete the word "of."

4. The basis for TS 3.17 has been changed to correct and enhance Table B 3.17-1. Table B 3.17-1 provides information on instruments which affect multiple TS.

5. The basis for TS 3.17.6, item 1, Neutron Flux Monitoring while shutdown, has been changed to clarify the effect on channel operability of the failure of either a wide range or a source range part of a channel.

6. The basis for TS 3.17.6, item 2, rod position indication, was changed to provide additional information about the functions of the rod position indication equipment. A typographical error in item 3, Safety Injection Refueling Water Tank Temperature, was corrected from "form" to "from."

7. The bases for TS 3.17.6, item 13, Rod Group Sequencing Control and Out of Sequencing Alarm, and item 18, Power Dependent Insertion Limit Alarm, have been changed to reflect changes in the plant computer system. Palisades formerly used two digital computer systems. During the 1995 refueling outage, these computers were replaced with a new computer system. A typographical error in item 11, Service Water System Break Detector, was corrected from "secession" to "succession."

3.0 STATE CONSULTATION

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

4.0 ENVIRONMENTAL CONSIDERATION

Portions of the amendment that modify the Facility Operating License to allow the use of source materials as reactor fuel and that revise the Palisades TS, change requirements with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 and change surveillance requirements. The staff has determined that these portions of the amendment involve no significant increase in the amounts, and no significant changes in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that the amendment involves no significant hazards consideration and there has been no public comment on such finding (60 FR 58399). Accordingly, the amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with issuance of the above-mentioned portions of the amendment.

Pursuant to 10 CFR 51.21, 51.32, and 51.35 an Environmental Assessment and Finding of No Significant Impact has been prepared for the administrative changes to the Palisades Facility Operating License and published in the Federal Register on March 15, 1996 (61 FR 10811). 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.

5.0 CONCLUSION

The staff has concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendment will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: J. Kennedy

Date: April 5, 1996

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

CONSUMERS POWER COMPANY

DOCKET NO. 50-255

NOTICE OF DENIAL OF AMENDMENT TO FACILITY OPERATING LICENSE AND OPPORTUNITY FOR HEARING

The U.S. Nuclear Regulatory Commission (the Commission) has denied a portion of a request by Consumers Power Company (the licensee) for an amendment to Facility Operating License No. DPR-20 issued to the licensee for operation of the Palisades Plant located in Van Buren County, Michigan. Notice of Consideration of Issuance of this amendment was published in the FEDERAL REGISTER on November 27, 1995 (60 FR 58399).

The purpose of the licensee's amendment request was to revise the Facility Operation License (FOL) to reference 10 CFR Part 40, allow the use of source materials as reactor fuel, delete references to specific amendments and specific revisions in the listed titles of the Physical Security Plan, Suitability Training and Qualification Plan, and the Safeguards Contingency Plan, delete paragraph 2.F on reporting requirements, and make minor editorial changes to the license. The Technical Specifications (TS) would also be revised to: (1) modify TS 3.1.2 to change the pressurizer cooldown limit from 100°F to 200°F/hour; (2) relocate the shield cooling system requirements to the Final Safety Analysis Report; (3) make minor editorial changes and corrections; and (4) revise several TS bases pages.

The NRC staff has concluded that the licensee's request to delete paragraph 2.F of the FOL cannot be granted. The licensee was notified of the Commission's denial of the proposed change by a letter dated April 5, 1996 .

By May 10, 1996 , the licensee may demand a hearing with respect to the denial described above. Any person whose interest may be affected by this proceeding may file a written petition for leave to intervene.

A request for hearing or petition for leave to intervene must be filed with the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555, Attention: Docketing and Services Branch, or may be delivered to the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, by the above date.

A copy of any petitions should also be sent to the Office of the General Counsel, U.S. Nuclear Regulatory Commission, Washington, DC 20555, and to Judd L. Bacon, Esquire, Consumers Power Company, 212 West Michigan Avenue, Jackson, Michigan 49201, attorney for the licensee.

For further details with respect to this action, see (1) the application for amendment dated October 17, 1995, and (2) the Commission's letter to the licensee dated April 5, 1996 .

These documents are available for public inspection at the Commission's Public Document Room, the Gelman Building, 2120 L Street, NW., Washington, DC, and at the local public document room located at the Van Wylen Library, Hope College, Holland, Michigan 49423.

Dated at Rockville, Maryland, this 5th day of April 1996.

For The Nuclear Regulatory Commission

Mark F. Reinhart

Mark F. Reinhart, Acting Project Director
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation