Docket No. 50-305

July 5, 1985

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Mr. D. C. Hintz
Manager - Nuclear Power
Wisconsin Public Service Corporation
Post Office Box 19002
Green Bay, Wisconsin 54307-9002

Dear Mr. Hintz:

The Commission has issued the enclosed Amendment No. 63 to Facility Operating License No. DPR-43 for the Kewaunee Nuclear Power Plant. The amendment consists of changes to the Technical Specifications (TS) in response to your application transmitted by letters dated August 24, 1983, (supplemented June 29, 1984), March 30, 1984, and March 19, 1985. This amendment completes our TAC numbers 43008, 56410 and 56436.

The amendment provides a definition of the term "OPERABLE", adds decay heat removal capability requirements, deletes ambiguities in TS 3.3, revises the control room ventilation system and includes many miscellaneous changes.

This license amendment will become effective 60 days from date of issuance in order to provide time, as requested by Wisconsin Public Service Corporation by letter dated April 4, 1985, to put required supporting plant procedures and training in effect.

A copy of the related Safety Evaluation is enclosed. A Notice of Issuance will be included in the Commission's next regular bi-weekly Federal Register notice.

Sincerely,

/s/MFairtile

Morton B. Fairtile, Project Manager
Operating Reactors Branch #1
Division of Licensing

Enclosures:
1. Amendment No. 63 to DPR-43
2. Safety Evaluation

cc: w/enclosures
See next page

ORB#1:DL	ORB#1:DL	BC-ORB#1:DL	OELD	AD-OR:DL
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UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

WISCONSIN PUBLIC SERVICE CORPORATION
WISCONSIN POWER AND LIGHT COMPANY
MADISON GAS AND ELECTRIC COMPANY

DOCKET NO. 50-305

KEWAUNEE NUCLEAR PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 63
License No. DPR-43

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

   A. The application for amendment by Wisconsin Public Service Corporation, Wisconsin Power and Light Company, and Madison Gas and Electric Company (the licensee) dated August 24, 1983, (as supplemented June 29, 1984), March 30, 1984 and March 19, 1985, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act) and the Commission's rules and regulations set forth in 10 CFR Chapter I;

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

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

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

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 2.C.(2) of Facility Operating License No. DPR-43 is hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 63 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This amendment is effective on issuance, to be implemented within 60 days of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Steven A. Varga, Chief
Operating Reactors Branch #1
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: July 5, 1985

ATTACHMENT TO LICENSE AMENDMENT

AMENDMENT NO. 63 TO FACILITY OPERATING LICENSE NO. DPR-43

DOCKET NO. 50-305

Revise Appendix A as follows:

Remove Pages	Insert Pages
i	i
ii	ii
iii	iii
v	v
1.1-2	1.1-2
-----------	1.1-2a
3.1-1	3.1-1
-----------	3.1-1a
3.1-2	3.1-2
3.2-2	3.2-2
3.3-1 through 3.3-8	3.3-1 through 3.3-12
3.4-1	3.4-1
3.4-1a	3.4-1a
3.4-2	3.4-2
Table TS 3.5-3 (Page 1 of 2)	Table TS 3.5-3 (Page 1 of 2)
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3.8-2a	3.8-2a
-----------	3.12-1
-----------	3.12-2
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Table TS 4.1-3 (Page 1 of 2)	Table TS 4.1-3 (Page 1 of 2)
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4.4-5 through 4.4-7	4.4-5 through 4.4-7
4.4-11	4.4-11
----------	4.4-12
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TECHNICAL SPECIFICATIONS
APPENDIX A

Amendment No. 63

Amendment No.63

e. Operable-Operability

A system or component is operable or has operability when it is capable of performing its intended function within the required range. The system or component shall be considered to have this capability when: (1) it satisfies the Limiting Conditions for Operation defined in Specification 3, and (2) it has been tested periodically in accordance with Specification 4 and has met its performance requirements.

Implicit in this definition shall be the assumption that all necessary attendent instrumentation, controls, normal and emergency electrical power sources, cooling or seal water, lubrication or other auxiliary equipment that is required for the system or component to perform its intended function is also capable of performing their related support functions.

f. Operating

A system or component is considered to be operating when it is performing the intended function in the intended manner.

g. <u>Containment System Integrity</u>

Containment System integrity is defined to exist when:

1. The non-automatic Containment System isolation valves and blind flanges are closed as required.
2. The Reactor Containment Vessel and Shield Building equipment hatches are properly closed.
3. At least ONE door in both the personnel and the emergency airlocks is properly closed.
4. The required automatic Containment System isolation valves are operable or are deactivated in the closed position or at least one valve in each line having an inoperable valve is closed.
5. All requirements of Specification 4.4 with regard to Containment System leakage and test frequency are satisfied.
6. The Shield Building Ventilation System and the Auxiliary Building Special Ventilation System satisfy the requirements of Specification 3.6.b.

h. <u>Protective Instrumentation Logic</u>

1. Protection System Channel

   A protection system channel is an arrangement of components and modules as required to generate a single protective action signal when required by a plant condition. The channel loses its identity where single action signals are combined.

3.0 LIMITING CONDITIONS FOR OPERATION

3.1 REACTOR COOLANT SYSTEM

Applicability

Applies to the operating status of the Reactor Coolant System.

Objective

To specify those limiting conditions for operation of the Reactor Coolant System which must be met to ensure safe reactor operation.

Specifications

a. OPERATIONAL COMPONENTS

Specification:

1. Reactor Coolant Pumps

   A. At least one reactor coolant pump or one residual heat removal pump shall be in operation when a reduction is made in the boron concentration of the reactor coolant.

   B. When the reactor is in the operating mode of operation, except for low power tests, both reactor coolant pumps shall be in operation.

2. Decay Heat Removal Capability

A. At least TWO of the following FOUR heat sinks shall be operable whenever the average reactor coolant temperature is less than or equal to 350°F but greater than 200°F.

1. Steam Generator 1A
2. Steam Generator 1B
3. Residual Heat Removal Train A
4. Residual Heat Removal Train B

If less than the above number of required heat sinks are operable, corrective action shall be taken immediately to restore the minimum number to the operable status.

B. TWO residual heat removal trains shall be operable whenever the average reactor coolant temperature is less than or equal to 200°F and irradiated fuel is in the reactor, except when in the refueling mode one train may be inoperable for maintenance.

1. Each residual heat removal train shall be comprised of:

   a) ONE operable residual heat removal pump

   b) ONE operable residual heat removal heat exchanger

   c) An operable flow path consisting of all valves and piping associated with the above train of components and required to remove decay heat from the core during normal shut down situations. This flow path shall be capable of taking suction from the appropriate reactor coolant system hot leg and returning to the reactor coolant system.

2. If one residual heat removal train is inoperable, corrective action shall be taken immediately to return it to the operable status.

3. Pressurizer Safety Valves

A. At least one pressurizer safety valve shall be operable whenever the reactor head is on the reactor pressure vessel, except for a hydro test of the RCS the pressurizer safety valves may be blanked provided the power operated relief valves and the safety valve on the discharge of the charging pump are set for test pressure plus 35 psi to protect the system.

B. Both pressurizer safety valves shall be operable whenever the reactor is critical.

4. Pressure Isolation Valves

Applicability:

Operational defined as Operating, and Hot Standby.

Objective:

To increase the reliability of reactor coolant system pressure isolation valves thereby reducing the potential of an intersystem loss of coolant accident.

Specification:

A. All pressure isolation valves listed in Table TS 3.1-2 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 as specified in Table TS 3.1-2 cannot be demonstrated, reactor operation may continue, provided that at least two valves in each high pressure line having a non-functional valve are in and remain in, the mode corresponding to the isolated condition. (a)

C. If Specification A and B cannot be met, an orderly shutdown shall be initiated and the reactor shall be in the Hot Shutdown condition within the next 4 hours, the Intermediate Shutdown condition in the next 6 hours and the Cold Shutdown condition within the next 24 hours.

(a) Manual valves shall be locked in the closed position; motor operated valves shall be placed in the closed position with their power breakers locked out.

c. Any one of the following conditions of inoperability may exist during the time intervals specified. The reactor shall be placed in the hot shutdown condition if operability is not restored within the time specified, and it shall be placed in the cold shutdown condition if operability is not restored within an additional 48 hours.

1. ONE of the operable charging pumps may be removed from service provided two pumps are again operable within 24 hours.

2. ONE boric acid transfer pump may be out of service provided both pumps are operable within 24 hours.

3. ONE channel of heat tracing may be out of service provided it is restored to operable status within 48 hours.

Basis

The Chemical and Volume Control System provides control of the Reactor Coolant System boron inventory. This is normally accomplished by using any one of the three charging pumps in series with any one of the two boric acid transfer pumps. An alternate method of boration will be use of the charging pumps directly from the Refueling Water Storage Tank. A third method will be to use the safety injection pumps. There are two sources of borated water available for injection through 3 different paths.

(1) The boric acid transfer pumps can deliver the boric acid tank contents to the suction of the charging pumps.

(2) The charging pumps can take suction directly from the Refueling Water Storage Tank containing a concentration of 1950 ppm boron solution. Reference is made to Specification 3.3.a.

3.3 ENGINEERED SAFETY FEATURES AND AUXILIARY SYSTEMS

Applicability

Applies to the operating status of Engineered Safety Features and Auxiliary Systems.

Objective

To define those limiting conditions for operation that are necessary: (1) to remove decay heat from the core in emergency or normal shutdown situations, and (2) to remove heat from containment in normal operating and emergency situations.

Specification

a. Accumulators

1. **The reactor shall not be made critical unless the following conditions are satisfied, except for low-power physics tests and except as provided by Specification 3.3.a.2.**

   A. Each accumulator is pressurized to at least 700 psig and contains 1250 $ft^3 \pm 25\ ft^3$ of water with a boron concentration of at least 1900 ppm, and is not isolated.

   B. Accumulator isolation valves SI-20A and SI-20B shall be opened with their power breakers locked out at or before the reactor coolant system pressure exceeds 1000 psig.

2. During power operation or recovery from an inadvertent trip, ONE accumulator may be inoperable for a period of 1 hour. If operability is not

restored within the time specified, then within 1 hour action shall be initiated to:

-Achieve Hot Standby within the next 6 hours.

-Achieve Hot Shutdown within the following 6 hours.

-Achieve Cold Shutdown within an additional 36 hours.

b. Safety Injection/Residual Heat Removal Systems

1. The reactor shall not be made critical unless the following conditions are satisfied, except for low-power physics tests and except as provided by Specification 3.3.b.2.

   A. The Refueling Water Storage Tank contains not less than 272,500 gal. of water with a boron concentration of at least 1950 ppm.

   B. TWO SI/RHR trains are operable with each train comprised of:

      1. ONE operable safety injection pump.

      2. ONE operable residual heat removal pump.

      3. ONE operable residual heat removal heat exchanger.

      4. An operable flow path consisting of all valves, piping and interlocks associated with the above train of components and required to function during accident conditions. This flow path shall be capable of taking suction from the selected boric acid tank and the refueling water storage tank upon a safety injection signal and after manual transfer taking suction from the containment sump.

   C. Isolation valves SI-9A, SI-11A and SI-11B in the discharge of the high head SIS and block valve SI-3 are in the open position with their power breaker locked out.

D. During the Quarterly Valve Operation Surveillance Testing of the Safety Injection System it is permissible to close the hand operated valve isolating the Boric Acid Storage Tanks from the Safety Injection Pumps Suction. During this short test period an operator shall stand by the valve to open it if Safety Injection is required. He will have headset communication with the Control Room.

2. During power operation or recovery from an inadvertent trip, ONE SI/RHR train may be inoperable for a period of 72 hours.

A. If the inoperability is due to a component in the safety injection system and operability is not restored within 72 hours, then within 1 hour action shall be initiated to:

-Achieve Hot Standby within the next 6 hours.
-Achieve Hot Shutdown within the following 6 hours.
-Achieve Cold Shutdown within an additional 36 hours.

B. If the inoperability is due to a component in the residual heat removal system and operability is not restored within 72 hours, then within 1 hour action shall be initiated to:

-Achieve Hot Standby within the next 6 hours.
-Achieve Hot Shutdown within the following 6 hours.

-Achieve and maintain the Reactor Coolant System T_{avg} less than 350°F by use of alternate heat removal methods within an additional 36 hours.

c. Containment Cooling Systems

1. The reactor shall not be made critical unless the following conditions are satisfied, except for low-power physics tests and except as provided by Specification 3.3.c.2.

   A. A minimum of three hundred (300) gallons of not less than 30% by weight of NaOH solution is available as a containment spray additive.

   B. Two containment spray trains are operable with each train comprised of:

      1. ONE containment spray pump.

      2. An operable flow path consisting of all valves and piping associated with the above train of components and required to function during accident conditions. This flow path shall be capable of taking suction from the refueling water storage tank and the spray additive tank upon a Hi-Hi Containment Pressure signal and after manual transfer being supplied from the containment sump.

   C. TWO trains of containment fan-coil units are operable with two fan-coil units in each train.

2. During power operation or recovery from inadvertent trip, any one of the following conditions of inoperability may exist during the time inter-

vals specified. If operability is not restored within the time specified, then within 1 hour action shall be initiated to:

-Achieve Hot Standby within the next 6 hours.
-Achieve Hot Shutdown within the following 6 hours.
-Achieve Cold Shutdown within an additional 36 hours.

A. The quantity of NaOH solution available as a containment spray additive may be less than that specified in TS 3.3.c.1.A for a period of 48 hours.

B. Any ONE of the following FOUR trains of equipment may be out of service for a period of 7 days provided the remaining THREE trains are operable.

1. Containment Spray - Train A
2. Containment Spray - Train B
3. Containment Fan Coil Units - 1A and 1B
4. Containment Fan Coil Units - 1C and 1D

C. Any TWO of the following FOUR trains of equipment may be out of service for a period of 72 hours provided the remaining TWO trains are operable.

1. Containment Spray - Train A
2. Containment Spray - Train B
3. Containment Fan Coil Units - 1A and 1B
4. Containment Fan Coil Units - 1C and 1D

d. Component Cooling System

1. The reactor shall not be made or maintained critical unless the following conditions are satisfied, except for low power physics tests and except as provided by Specification 3.3.d.2.

   A. TWO component cooling water trains are operable with each train consisting of:

      1. ONE component cooling water pump

      2. ONE component cooling water heat exchanger

      3. An operable flow path consisting of all valves and piping associated with the above train of components and required to function during accident conditions.

2. During power operation on recovery from an inadvertent trip, ONE component cooling water train may be inoperable for a period of 72 hours. If operability is not restored within 72 hours, then within 1 hour action shall be initiated to:

   -Achieve Hot Standby within the next 6 hours.

   -Achieve Hot Shutdown within the following 6 hours.

   -Achieve Cold Shutdown within an additional 36 hours.

e. Service Water System

1. The reactor shall not be made critical unless the following conditions are satisfied, except for low-power physics tests and except as provided by Specification 3.3.e.2.

   A. TWO service water trains are operable with each train consisting of:

1. TWO service water pumps
2. An operable flow path consisting of all valves and piping associated with the above train of components and required to function during accident conditions. This flow path shall be capable of taking a suction from the forebay and supplying water to the redundant safeguards headers.

B. The forebay water level trip system is operable.

2. During power operation or recovery from an inadvertent trip, ONE service water train may be inoperable for a period of 72 hours. If operability is not restored within 72 hours, then within 1 hour action shall be initiated to:

-Achieve Hot Standby within the next 6 hours.
-Achieve Hot Shutdown within the following 6 hours.
-Achieve Cold Shutdown within an additional 36 hours.

Basis

The normal procedure for starting the reactor is, first, to heat the reactor coolant to near operating temperature by running the reactor coolant pumps. The reactor is then made critical by withdrawing control rods and/or diluting boron in the coolant.(1) With this mode of start-up, the energy stored in the reactor coolant during the approach to criticality is substantially equal to that during power operation and therefore, to be conservative, most engineered safety features components and auxiliary cooling systems shall be fully operable.

The operable status of the various systems and components is to be demonstrated by periodic tests, defined by Specification 4.5. These periodic tests ensure, with a high reliability, that the various systems will function properly if required to do so. 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. Limiting conditions of operation permit temporary outages of redundant components and are specified for specific time intervals that are consistent with minor maintenance. These permissible conditions and time intervals are specified in such a manner as to apply identically during sustained power operation and during recovery from an inadvertent trip. The transient condition of restart in the latter case in no way alters the types of safety features equipment nor the extent of redundancy that must be available.

Inoperability of a single component does not negate the ability of the system to perform its function, but it reduces the redundancy provided in the plant design and thereby limits the ability to tolerate additional equipment failures. However, the equipment out-of-service times specified in the limiting conditions for operation are a temporary relaxation of the single failure criterion, which, consistent with overall system reliability considerations, provides a limited time to restore equipment to the operable condition. If the inoperable component is not repaired within the specified allowable time period or a second component in the same or related system is found to be inoperable and cannot be repaired within the specified time, the reactor will initially be put in hot standby and subsequently in the hot shutdown condition to reduce the stored energy in the reactor coolant system and to provide for the reduction of the decay heat from the fuel. These actions result in a reduction of the cooling requirements after a postulated loss-of-coolant accident. If the malfunction(s) are not corrected after the specified time in a hot shutdown condition, the reactor will be placed in the cold shutdown condition, utilizing normal shutdown and cooldown procedures. In the cold shutdown condition there is no possibility of an accident that would release fission products or damage the fuel elements.

When the inoperable component is part of the residual heat removal (RHR) system the average reactor coolant system temperature (T_{avg}) will be maintained below 350°F through an alternate heat removal method. The various alternate heat removal methods include the redundant RHR train and the steam generators.

Assuming the reactor has been operating at full rated power for at least 100 days, the magnitude of the decay heat decreases as follows after initiating hot shutdown.

Time After Shutdown	Decay Heat, % of Rated Power
1 min.	4.5
30 min.	2.0
1 hour	1.62
8 hours	0.96
48 hours	0.62

Thus the requirement for core cooling in case of a postulated loss-of-coolant accident while in the hot shutdown condition is significantly reduced below the requirements for a postulated loss-of-coolant accident during power operation. Putting the reactor in the hot shutdown condition significantly reduces the potential consequences of a loss-of-coolant accident, and also allows more free access to some of the engineered safety features in order to effect repairs. Failure to complete repairs after placing the reactor in the hot shutdown condition may be indicative of need for major maintenance, and in such cases the reactor should therefore be placed in the cold shutdown condition.

The accumulator and refueling water storage tank conditions specified are consistent with those assumed in the LOCA analysis.(2)

The containment cooling function is provided by two independent systems: fan-coil units and containment sprays. During normal operation, usually three of the four fan-coil units are required to remove heat lost from equipment and piping within the containment.(3) In the event of the Design Basis Accident, any one of the following combinations will provide sufficient cooling to reduce containment pressure: four fan-coil units, two containment spray pumps, or two fan-coil units plus one containment spray pump.(4)

In addition to heat removal, the containment spray system is also effective in scrubbing fission products from the containment atmosphere. However, no credit is taken for this scrubbing action in the analysis of the Design Basis Accident.

Caustic (NaOH) is added to the spray solution for ph adjustment to preclude chloride stress corrosion cracking of stainless steel components in the postaccident environment. Test data has shown that no significant stress corrosion cracking will occur provided the ph is adjusted within two (2) days following the Design Basis Accident.(4)(7) A minimum of 300 gallons of not less than 30% by weight of NaOH solution is sufficient to adjust the ph of the spray solution adequately. The additive will still be considered available whether it is contained in the spray additive tank or the containment spray system piping due to an inadvertent opening of the spray additive valves (CI-1001A and CI-1001B).

One component cooling water pump together with one component cooling heat exchanger can accommodate the heat removal load either following a loss-of-coolant accident, or during normal plant shutdown. If, during the post-accident phase, the component cooling water supply were lost, core and containment cooling could be maintained until repairs were effected.(5)

A total of four service water pumps are installed, and a minimum of two are required to operate during the postulated loss-of-coolant accident.(6) The service water valves in the redundant safeguards headers have to be operable in order for the components that they supply to be considered operable.

The various trains of equipment referred to in the specifications are separated by their power supplies (i.e.: SI Pump 1A, RHR Pump 1A, Valves SI-2A and SI-4A, etc.). Shared piping and valves are considered to be common to both trains of the systems (i.e.: SI-3, etc.).

The closure of the hand operated valve for a brief period of time during the surveillance testing of the automatic valves in the safety injection system will prevent dilution of the concentrated boric acid or loss of concentrated boric acid to the refueling water storage tank.

References

(1) FSAR Section 3.2

(2) FSAR Section 14.3

(3) FSAR Section 6.3

(4) FSAR Section 6.4

(5) FSAR Section 9.3

(6) FSAR Section 9.6

(7) Westinghouse Chemistry Manual SIP 5-1, Rev. 2, dated 3-77, Section 4.

3.4 STEAM AND POWER CONVERSION SYSTEM

Applicability

Applies to the operating status of the Steam and Power Conversion System.

Objective

To assure minimum conditions of steam-relieving capacity and auxiliary feedwater supply necessary to assure the capability of removing decay heat from the reactor, and to limit the concentrations of water activity that might be released by steam relief to the atmosphere.

Specification

a. The reactor shall not be heated above 350°F unless the following conditions are satisfied.

   1. TWO steam generators are operable.

      A. System piping and valves directly associated with providing auxiliary feedwater flow to the steam generators are operable.

      B. Five main steam safety valves per operable steam generator are operable, except during required surveillance tests or during inservice testing of these valves and steam generators in accordance with 10 CFR 50.55a, provided that at least two main steam safety valves associated with the steam generator under test are operable.

   2. Three auxiliary feedwater pumps are operable.

   3. A minimum of 30,000 gallons of water is available in the condensate storage tanks and the Service Water system is capable of delivering an unlimited supply from Lake Michigan.

   4. The iodine-131 activity on the secondary side of the steam generators does not exceed 1.0 μCi/cc.

b. If, when the reactor is above 350°F, any of the conditions of Specification 3.4.a cannot be met within 48 hours, and except for the conditions of 3.4.c, the reactor shall be shut down and cooled below 350°F using normal operating procedures.

c. When the reactor is above 350°F, one auxiliary feedwater pump may be inoperable provided the pump is restored to operable status within 72 hours, or the reactor shall be shutdown and cooled below 350°F using normal operating procedures.

Basis

Two steam generators are required to be operable when the average reactor coolant temperature is above 350°F to ensure that sufficient heat removal capability exists for power operation and decay heat removal. Although one steam generator would provide sufficient decay heat removal capability, two steam generators are required in order to provide the necessary redundancy to meet the single failure criterion. An operable steam generator is defined by TS 3.4.a.1.

The ten main steam safety valves (five per steam generator) have a total combined rated capability of 7,765,000 lbs/hr. The maximum full-power steam flow is 7,449,000 lbs/hr; therefore, the main steam safety valves will be able to relieve the total maximum steam flow if necessary. The requirement that five main steam safety valves per operable steam generator are available will assure sufficient steam relief capability.

Testing of the main steam system while the plant is in hot shutdown conditions is permitted provided that at least two main steam safety valves associated with the steam generator under test are available to provide sufficient relief capacity to protect the system during the test.

In the unlikely event of complete loss of electrical power to the plant, continued capability of decay heat removal would be assured by the availability of either the steam-driven auxiliary feedwater pump or one of the two motor-driven auxiliary feedwater pumps, and by steam discharge to the atmosphere through the main steam safety valves. Each motor-driven pump is normally aligned with one steam generator; the discharge of the turbine-driven pump, which starts automatically, is manually valved as necessary to backup either or both motor-driven pumps, or to replace the standby function of either motor-driven pump when it is out of service. Any single auxiliary feedwater pump can supply sufficient feedwater for removal of decay heat from the reactor.

The specified minimum water supply in the condensate storage tanks is sufficient for ninety minutes of hot shutdown plus a suitable margin to prevent loss of net positive suction head prior to switching suction to the Service Water System. Unlimited replenishment of the condensate storage supply is available from Lake Michigan through the Service Water System.

TABLE TS 3.5-3 (Page 1 of 2)

EMERGENCY COOLING

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MINIMUM OPERABLE CHANNELS	4 MINIMUM DEGREE OF REDUNDANCY	5 PERMISSIBLE BYPASS CONDITIONS	6 OPERATOR ACTION IF CONDITIONS OF COLUMN 3 OR 4 CANNOT BE MET
1	SAFETY INJECTION						
	a. Manual	2	1	1	-		Hot Shutdown***
	b. High Containment Pressure	3	2	2	-		Hot Shutdown***
	c. Low Steam Pressure/Line	3	2	2	-		Hot Shutdown***
	d. Pressurizer Low Pressure	3	2	2	-	Primary pressure < 2000 psig	
2	SELECTED BORIC ACID STORAGE TANK LEVEL	4	-	4	-		One channel may be inoperable for a period of 72 hrs. otherwise take action in accordance with TS 3.3.b.2.A.
3	CONTAINMENT SPRAY						
	a. Manual	2	2	2	**		Hot Shutdown***
	b. Hi-Hi Containment Pressure (Containment Spray)	3 sets of 2	1 of 2 in each set	1 per set	1/set		Hot Shutdown***

(Deleted)

TABLE TS 3.5-3 (Page 2 of 2)
EMERGENCY COOLING

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MINIMUM OPERABLE CHANNELS	4 MINIMUM DEGREE OF REDUNDANCY	5 PERMISSIBLE BYPASS CONDITIONS	6 OPERATOR ACTION IF CONDITIONS OF COLUMN 3 OR 4 CANNOT BE MET
4	MOTOR DRIVEN AUXILIARY FEEDWATER PUMPS						
	a. Either Steam Generator Lo-Lo Level	3/loop	2/loop	2/loop	-		Maintain hot shutdown
	b. Loss of Main Feed water ****	1	1	1			Maintain hot shutdown
	c. Safety Injection	(Refer to Item 1 of this Table)					
	d. 4 KV Buses 1-5 and 1-6 under voltage	2/Bus*	1/Bus	1/Bus*****			Maintain hot shutdown or operate diesel generators.
5	TURBINE DRIVEN AUXILIARY FEEDWATER PUMPS						
	a. Both Steam Generator Lo-Lo Level	3/loop	2/loop	2/loop	-		Maintain hot shutdown
	b. 4 KV Buses 1-1 and 1-2 under voltage	(Refer to Item 13 of Table TS 3.5-2)					

* Each channel consists of one instantenous and one-time relay connected in series.

** Must actuate 2 switches

*** If minimum conditions are not met within 24 hours, steps shall be taken to place the plant in cold shutdown condition.

**** Tripping of both Main feedwater Pump Breakers starts both motor driven auxiliary feedwater pumps.

***** When one component of a channel is taken out of service, that component shall be in the tripped condition.

TABLE TS 3.5-4 (Page 2 of 2)

INSTRUMENT OPERATING CONDITIONS FOR ISOLATION FUNCTIONS

NO.	FUNCTIONAL UNIT	1 NO. OF CHANNELS	2 NO. OF CHANNELS TO TRIP	3 MINIMUM OPERABLE CHANNELS	4 MINIMUM DEGREE OF REDUNDANCY	5 PERMISSIBLE BYPASS CONDITIONS	6 OPERATOR ACTION IF CONDITIONS OF COLUMN 3 OR 4 CANNOT BE MET
3	CONTAINMENT VENTILATION ISOLATION						
	a. High Containment Radiation	2	1	1	-	-	These channels are not required to activate containment ventilation isolation when the containment purge and ventilation system isolation valves are maintained closed.*
	b. Safety Injection	(Refer to Item 1 of Table TS 3.5-3)					
	c. Containment Spray	(Refer to Item 3 of Table TS 3.5-3)					

* The detectors are required for reactor coolant system leak detection as referenced in Technical Specifications 3.1.d.5

*** If minimum conditions are not met within 24 hours, steps shall be taken to place the plant in a cold shutdown condition.

b. During power operation or recovery from inadvertent trip, any of the following conditions of inoperability may exist during the time intervals specified. If operability is not restored within the time specified, then within 1 hour action shall be initiated to achieve hot standby within the next 6 hours.

1. Either Auxiliary Transformer may be out of service for a period not exceeding 7 days provided the other Auxiliary Transformer and both diesel generators are operable.
2. ONE diesel generator may be inoperable for a period not exceeding 7 days provided the other diesel generator is tested daily to ensure operability and the engineered safety features associated with this diesel generator are operable.
3. ONE battery may be inoperable for a period not exceeding 24 hours provided the other battery and two battery chargers remain operable with one charger carrying the d-c supply system.
4. The North Appleton Line may be out of service for a period not to exceed 7 days provided at least two other transmission lines serving the substation are in service.
5. Three off site power supply transmission lines may be out of service for a period of 7 days provided reactor power is reduced to 50% of rated power and the two diesel generators shall be tested daily for operability.
6. One 4160V or 480V Engineered Safety Features bus may be out of service for 24 hours provided the redundant bus and its loads remain operable.

c. When its normal or emergency power source is inoperable, a system, train or component may be considered operable for the purpose of satisfying the requirements of its applicable limiting condition for operation, provided:

1. Its corresponding normal or emergency power source is operable; and

2. Its redundant system, train or component is operable.

Basis

The intent of this specification is to provide assurance that at least one external source and one standby source of electrical power is always available to accomplish safe shutdown and containment isolation and to operate required engineered safety features equipment following an accident.

3.8 REFUELING

Applicability

Applies to operating limitations during refueling operations.

Objective

To ensure that no incident occurs during refueling operations that would affect public health and safety.

Specification

a. During refueling operations:

   1. The equipment hatch and at least one door in each personnel air lock shall be closed. In addition, each line that penetrates containment and which provides a direct air path from containment atmosphere to the outside atmosphere shall have a closed isolation valve or an operable automatic isolation valve.
   2. Radiation levels in fuel handling areas, the containment and the spent fuel storage pool shall be monitored continuously.
   3. The reactor will be subcritical for 100 hours prior to movement of its irradiated fuel assemblies. Core subcritical neutron flux shall be continuously monitored by at least TWO neutron monitors, each with continuous visual indication in the control room and ONE with audible indication in the containment whenever core geometry is being changed. When core geometry is not being changed at least ONE neutron flux monitor shall be in service.
   4. At least ONE residual heat removal pump shall be operable.
   5. When there is fuel in the reactor, a minimum boron concentration of 2100 ppm shall be maintained in the Reactor Coolant System during reactor vessel head removal or while loading and unloading fuel from the reactor and verified by sampling daily.

6. Direct communication between the control room and the operating floor of the containment shall be available whenever changes in core geometry are taking place.

7. Heavy loads, greater than the weight of a fuel assembly, will not be transported over or placed in either spent fuel pool when spent fuel is stored in that pool. Placement of additional fuel storage racks is permitted, however, these racks may not traverse directly above spent fuel stored in the pools.

8. The containment ventilation and purge system, including the capability to initiate automatic containment ventilation isolation, shall be tested and verified to be operable immediately prior to and daily during refueling operations.

9. A. The spent fuel pool sweep system, including the charcoal adsorbers, shall be operating during fuel handling and when any load is carried over the pool if irradiated fuel in the pool has decayed less than 30 days. If the spent fuel pool sweep system, including the charcoal adsorber, is not operating when required, fuel movement shall not be started (any fuel assembly movement in progress may be completed).

B. Performance Requirements

(1) The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filters and charcoal adsorber banks shall show $\geq$99% DOP removal and $\geq$ 99% halogenated hydrocarbon removal.

(2) The results of laboratory carbon sample analysis from spent fuel pool sweep system carbon shall show $\geq$90% radioactive methyl iodide removal at conditions of 66°C and 95% RH.

(3) Fans shall operate within $\pm$ 10% of design flow when tested.

10. The minimum water level above the vessel flange shall be maintained at 23 feet.

11. A dead-load test shall be successfully performed on both the fuel handling and manipulator cranes before fuel movement begins. The load assumed by the cranes for this test must be equal to or greater than the maximum load to be assumed by the cranes during the refueling operation. A thorough visual inspection of the cranes shall be made after the dead-load test and prior to fuel handling.

3.12 CONTROL ROOM POSTACCIDENT RECIRCULATION SYSTEM

APPLICABILITY

Applies to the operability of the Control Room Postaccident Recirculation System.

OBJECTIVE

To specify operability requirements for the Control Room Postaccident Recirculation System.

SPECIFICATIONS

a. The reactor shall not be made critical unless both trains of the Control Room Postaccident Recirculation System are operable.

b. Both trains of the Control Room Postaccident Recirculation System, including filters shall be operable or the reactor shall be shut down within 12 hours, except that when one of the two trains of the Control Room Postaccident Recirculation System is made or found to be inoperable for any reason, reactor operation is permissible only during the suceeding SEVEN days provided that the other train is demonstrated to be operable within 2 hours and daily thereafter.

c. During testing the system shall meet the following performance requirements:

   1. The results of the in-place cold DOP and halogenated hydrocarbon tests at design flows on HEPA filter and charcoal adsorber banks shall show $\geq$ 99% DOP removal and $\geq$ 99% halogenated hydrocarbon removal.

2. The results of the laboratory carbon sample analysis from the control room post accident recirculation system carbon shall show $\geq$ 90% radioactive methyl iodide removal at conditions of 66°C, and 95% RH.

3. Fans shall operate within $\pm$ 10% of design flow when tested.

BASIS

CONTROL ROOM POSTACCIDENT RECIRCULATION SYSTEM

The control room postaccident recirculation system is designed to filter the control room atmosphere during control room isolation conditions. The control room postaccident recirculation system is designed to automatically start upon SIS or high radiation signal at inlet of unit.

If the system is found to be inoperable, there is no immediate threat to the control room and reactor operation may continue for a limited period of time while repairs are being made. If the system cannot be repaired within seven days, the reactor is placed in hot standby until the repairs are made.

TABLE TS 4.1-2

MINIMUM FREQUENCIES FOR SAMPLING TESTS

Sampling Tests	Test	Frequency	Maximum Time Between Tests (Days)
1. Reactor Coolant Samples	Gross Beta-Gamma activity (excluding tritium)	5/week	3
	Tritium activity	Monthly	37
	*Chemistry (Cl, F, O_2)	3/week	4
2. Reactor Coolant Boron (1)	*Boron concentration	2/week	5
3. Refueling Water Storage(2) Tank Water Sample	Boron concentration	Monthly*****	37
4. Boric Acid Tanks	Boron concentration	Weekly	8
5. Accumulator	Boron concentration	Monthly	37
6. Spent Fuel Pool	Boron concentration	Monthly**	37
7. Secondary Coolant	Gross Beta-Gamma activity	Weekly	8
	Iodine concentration	Weekly when gross Beta-Gamma activity $\geq$1.0 uCi/cc	8
8. Waste Disposal System Liquid Effluent Monitor	Gross Beta-Gamma activity	Prior to each batch release	N.A.
9. Circulating Water Monitor	Radioactivity analysis	Continuous***	N.A.
10. Auxiliary Building Vent Monitor	Gross Beta-Gamma activity	Continuous****	N.A.
11. Containment Vessel Vent Air Particulate Monitor	I-131 and particulate activity	Continuous***	N.A.
12. Containment Vessel Vent Radiogas Monitor	I-131	Continuous***	N.A.

Notes

* See Spec 4.1.D

** Sample will be taken monthly when fuel is in the pool.

*** Continuous monitoring takes place when reactor is in operation.

**** Operable during refueling also.

***** And after adjusting tank contents.

TABLE TS 4.1-2 (Cont.)

MINIMUM FREQUENCIES FOR SAMPLING TESTS

(1) A reactor coolant boron concentration sample does not have to be taken when the core is completely unloaded.

(2) A refueling water storage tank (RWST) boron concentration sample does not have to be taken when the RWST is empty during refueling outages.

TABLE 4.1-3

MINIMUM FREQUENCIES FOR EQUIPMENT TESTS
(Page 1 of 2)

Equipment Tests***	Test	Frequency	Maximum Time Between Test (Days)
1. Control Rods	Rod drop times of all full length rods	Each refueling outage	N.A.
	Partial movement of all rods	Every 2 weeks	17
1a. Reactor Trip Breakers	Open trip	Monthly	37
1b. Reactor Coolant Pump Breakers-Open-Reactor Trip	Operability	Each refueling outage	N.A.
2. Deleted			
3. Deleted			
4. Containment Isolation Trip	Operability	Each refueling outage	N.A.
5. Refueling System Interlocks	Operability	Prior to each refueling outage	N.A.
6. Deleted			
7. Fire Protection Pump and Power Supply	*Operability	Monthly	37
8. RCS Leak Detection	Operability	Weekly	8
9. Diesel Fuel Supply	*Fuel inventory	Weekly	8
10. Turbine Stop and Governor Valves	Operability	Monthly(1)	37(1)
11. Fuel Assemblies	Visual Inspection	Each refueling outage	N.A.
12. Guard Pipes	Visual Inspection	Each refueling outage	N.A.

TABLE TS 4.1-3

MINIMUM FREQUENCIES FOR EQUIPMENT TESTS
(Page 2 of 2)

Equipment Test	Test	Frequency	Maximum Time Between Test (Days)
13. Pressurizer PORV's	Operability	Each Refueling Cycle	NA
14. Pressurizer PORV Block Valves	Operability	Quarterly*****	NA
15. Pressurizer Heaters	Operability******	Each Refueling Cycle	NA
16. Containment Purge and Vent Isolation Valves	Operability****	Each Refueling Cycle	NA

NOTES

* See Specification 4.1.d

*** Followinq maintenance on the above equipment that could affect the operation of the equipment tests should be performed to verify operability

**** This test shall demonstrate that the valve(s) close in less than or equal to 5 seconds.

***** Not required when valve is administratively closed.

****** Test will verify operability of heaters and availability of an emergency power supply.

(1) This test may be waived for end of cycle operations when boron concentrations are less than 150 ppm, due to operational limitations.

to this value.

c. Residual Heat Removal System

1. Those portions of the Residual Heat Removal System external to the isolation valves at the containment shall be hydrostatically tested at 350 psig at each major refueling outage, or they shall be tested during their use in normal operation at least once between successive major refueling outages.
2. The total leakage from either train shall not exceed two gallons per hour. Visible leakage that cannot be stopped at test conditions shall be suitably measured to demonstrate compliance with this Specification.
3. Any repairs necessary to meet the specified leak rate shall be accomplished within seven days of resumption of power operation.

d. Shield Building Ventilation System

1. At least once per operating cycle or once every 18 months whichever occurs first, the following conditions shall be demonstrated:
   A. Pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 10 inches of water and the pressure drop across any HEPA filter bank is less than 4 inches of water at the system design flow rate ($\pm$10%).
   B. Automatic initiation of each train of the system.
   C. Operability of heaters at rating and the absence of defects by visual inspection.
2. A. The in-place DOP test for HEPA filters shall be performed (1) at least once per 18 months and (2) after each complete or partial replacement of a HEPA filter bank or after any maintenance on the system that could affect the HEPA bank bypass leakage.
   B. The laboratory tests for activated carbon in the charcoal filters shall be performed (1) at least once per 18 months for filters in a standby status or after 720 hours of filter operation, and

(2) following painting, fire or chemical release in any ventilation zone communicating with the system.

C. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of a charcoal adsorber bank or after any maintenance on the system that could affect the charcoal adsorber bank bypass leakage.

D. Each train shall be operated with the heaters on at least 10 hours every month.

3. Perform an air distribution test on the HEPA filter bank after any maintenance or testing that could affect the air distribution within the systems. The test shall be performed at design flow rate ($\pm$10%). The results of the test shall show the air distribution is uniform within $\pm$20%.*

4. Each train shall be determined to be operable at the time of its periodic test if it produces measurable indicated vacuum in the annulus within two minutes after initiation of a simulated safety injection signal and obtains equilibrium discharge conditions that demonstrate the Shield Building leakage is within acceptable limits.

e. Auxiliary Building Special Ventilation System

1. Periodic tests of the Auxiliary Building Special Ventilation System, including the door interlocks, shall be performed in accordance with Specifications 4.4.d.1 through 4.4.d.3 except for Specification 4.4.d.2.D.

2. Each train of Auxiliary Building Special Ventilation System shall be operated with the heaters on at least 15 minutes every month.

* See Note p.p. TS 4.12-2

3. Each system shall be determined to be operable at the time of its periodic test if it starts with coincident isolation of the normal ventilation ducts and produces a measurable vacuum throughout the Special Ventilation Zone with respect to the outside atmosphere.

f. Containment Vacuum Breaker System

The power operated valve in each vent line shall be tested during each refueling outage to demonstrate that a simulated containment vacuum of 0.5 psi will open the valve and a simulated accident signal will close the valve. The check and butterfly valves will be leak tested in accordance with specification 4.4.b during each refueling.

Basis

The Containment System consists of a steel Reactor Containment Vessel within a concrete Shield Building and a Shield Building Ventilation System which, in the event of a loss-of-coolant accident, will produce a vacuum in the Shield Building annulus and will cause all leakage from the Reactor Containment Vessel to be mixed in the annulus volume and recirculated through a filter system before its deferred release to the environment through the exhaust fan that maintains vacuum in the annulus. Potential leakage from the RHRS or from the majority of lines that span the Shield Building annulus is collected in a special ventilation zone of the Auxiliary Building and filtered before its release.

The free-standing Reactor Containment Vessel is designed to accomodate the maximum internal pressure that would result from the Design Basis Accident.(1) For initial conditions typical of normal operation, 120°F and 15 psia, an instantaneous double-ended break with minimum safety features results in a peak pressure of 42.2 psig at 268°F.

the adsorbent thoroughly, and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. The use of multi-sample assemblies for test samples is an acceptable alternate to mixing one bed for a sample. If the iodine removal efficiency test results are unacceptable, all adsorbent in the system should be replaced. Any HEPA filters found defective should be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52 (Rev. 1) dated June 1976.

Operation of the systems every month will demonstrate operability of the filters and adsorber system. Operation of the Sheild Building Ventilation System will result in a discharge to the environment. This discharge is made after at least 2 samples of the building atmosphere have been analyzed to determine the concentration of activity in the atmosphere.

If painting, fire or chemical release occurs such that the charcoal adsorbers become contaminated from the fumes, chemicals, or foreign materials, the same tests and sample analysis should be performed as required for operational use.

Degradation of the HEPA filters due to painting, fire or chemical release in a communicating ventilation zone would be detected by an increased pressure drop across the filters. Should the filters become contaminated, engineering judgment would be used to determine if further leakage and/or efficiency testing was required.

Demonstration of the automatic initiation capability is necessary to assure system performance capability.

Periodic checking of the inlet heaters and associated controls for each train will provide assurance that the system has the capability of reducing inlet air humidity so that charcoal adsorber efficiency is enhanced.

In-place testing procedures will be established utilizing applicable sections of ANSI N510 - 1975 standard as a procedural guideline only.

References:

(1) FSAR Section 5
(2) FSAR Section 14.3.3
(3) Proposed 10 CFR Part 50, Appendix J (Revised)
(4) FSAR Section 5.5

4.12 SPENT FUEL POOL SWEEP SYSTEM

Applicability

Applies to testing and surveillance requirements for the spent fuel pool sweep system in Specifications 3.8.a.9.

Objective

To verify the performance capability of the spent fuel pool sweep system.

Specification

a. At least once per operating cycle or once every 18 months, whichever occurs first, the following conditions shall be demonstrated:
   1. Pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 10 inches of water and the pressure drop across any HEPA bank is less than 4 inches of water at the system design flow rate (±10%).
   2. Automatic initiation of each train of the system.

b. 1. The in-place DOP test for HEPA filters shall be performed (1) at least once per 18 months and (2) after each complete or partial replacement of a HEPA filter bank or after any maintenance on the system that could affect the HEPA bank bypass leakage.
   2. The laboratory tests for activated carbon in the charcoal filters shall be performed (1) at least once per 18 months for filters in a standby status or after 720 hours of filter operation, and (2) following painting, fire, or chemical release in any ventilation zone communicating with the system.

3. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of a charcoal adsorber bank or after any maintenance on the system that could affect the charcoal adsorber bank bypass leakage.

c. Perform an air distribution test on the HEPA filter bank after any maintenance or testing that could affect the air distribution within the system. The test shall be performed at design flow rate (±10%). The results of the test shall show the air distribution is uniform within ±20%.*

* This note applies here and also to 4.4.d.3 on p.p. TS 4.4-6.

In WPS letter of August 25, 1976 to Mr. Al Schwencer (NRC) from Mr. E. W. James, we relayed test results for flow distribution for tests performed in accordance with ANSI N510-1975. This standard refers to flow distribution tests performed upstream of filter assemblies. Since the test results upstream of filters were inconclusive due to high degree of turbulence, tests for flow distribution were performed downstream of filter assemblies with acceptable results (within 20%). The safety evaluation attached to Amendment 12 references our letter of August 25, 1976 and acknowledges acceptance of the test results.

Basis

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 10 inches of water and 4 inches across any HEPA filter bank at the system design flow rate ($\pm$10%) will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. A test frequency of once per operating cycle establishes system performance capability. This pressure drop is approximately 6 inches of water when filters are clean.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Replacement adsorbent should be qualified according to the guidelines of Regulatory Guide 1.52 dated June 1973. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly, and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. The use of multi-sample assemblies for test samples is an acceptable alternate to mixing one bed for a sample. If the iodine removal efficiency test results are unacceptable, all adsorbent in the system should be replaced. Any HEPA filters found defective should be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52 (Rev. 1) dated June 1976.

If painting, fire, or chemical release occurs such that the charcoal adsorbers become contaminated from the fumes, chemicals, or foreign materials, the same tests and sample analysis should be performed as required for operational use.

Degradation of the HEPA filters due to painting, fire or chemical release in a communicating ventilation zone would be detected by an increased pressure drop across the filters. Should the filters become contaminated, engineering judgment would be used to determine if further leakage and/or efficiency testing was required.

Demonstration of the automatic initiation capability is necessary to assure system performance capability.

4.17 CONTROL ROOM POSTACCIDENT RECIRCULATION SYSTEM

APPLICABILITY

Applies to testing and surveillance requirements for the Control Room Postaccident Recirculation System in Specification 3.12.

OBJECTIVE

To verify the performance capability of the control room postaccident recirculation system.

SPECIFICATION

a. At least once per operating cycle or once every 18 months, whichever occurs first, the following conditions shall be demonstrated:

   1. Pressure drop across the combined HEPA filters and charcoal adsorber banks is less than 6 inches of water and the pressure drop across any HEPA bank is less than 4 inches of water at the system design flow rate ($\pm$ 10%).

   2. Automatic initiation of the system on a high radiation signal at the inlet of the unit and a safety injection signal.

b. 1. The in-place DOP test for HEPA filters shall be performed (1) at least once per 18 months and (2) after each complete or partial replacement of a HEPA filter bank or after any maintenance on the system that could affect the HEPA bank bypass leakage.

   2. The laboratory test for activated carbon in the charcoal filters shall be performed (1) at least once per 18 months for filters in a standby status or after 720 hours of filter operation, and (2) following painting, fire, or chemical release in any ventilation zone communicating with the system.

3. Halogenated hydrocarbon testing shall be performed after each complete or partial replacement of a charcoal adsorber bank or after any maintenance on the system that could affect the charcoal adsorber bank bypass leakage.
4. Each train shall be operated at least 10 hours each month.

BASIS

CONTROL ROOM POSTACCIDENT RECIRCULATION SYSTEM

Pressure drop across the combined HEPA filters and charcoal adsorbers of less than 6 inches of water and 4 inches across any HEPA filter bank at the system design flow rate ($\pm$ 10%) will indicate that the filters and adsorbers are not clogged by excessive amounts of foreign matter. A filter test frequency of once per operating cycle establishes system performance capability.

The frequency of tests and sample analysis are necessary to show that the HEPA filters and charcoal adsorbers can perform as evaluated. Replacement adsorbent should be qualified according to the guidelines of Regulatory Guide 1.52, dated June 1973. The charcoal adsorber efficiency test procedures should allow for the removal of one adsorber tray, emptying of one bed from the tray, mixing the adsorbent thoroughly, and obtaining at least two samples. Each sample should be at least two inches in diameter and a length equal to the thickness of the bed. The use of multi-sample assemblies for test samples is an acceptable alternate to mixing one bed for a sample. If the iodine removal efficiency test results are unacceptable, all adsorbent in the system should be replaced.

Any HEPA filters found defective should be replaced with filters qualified pursuant to Regulatory Position C.3.d of Regulatory Guide 1.52 (Rev. 1), dated June 1976. If painting, fire, or chemical release occurs such that the charcoal adsorber could become contaminated from the fumes, chemicals, or foreign materials, the same tests and sample analysis should be performed as required for operational use.

Demonstration of the automatic initiation capability is necessary to assure system performance capability.

In-place testing procedures will be established utilizing applicable sections of ANSI N510-1975 standard as a procedural guideline only.

RESPONSIBILITIES

6.5.1.6 The PORC shall be responsible for:

a. Review of operating, maintenance and other procedures including emergency operating procedures which affect nuclear safety as determined by the plant manager. Changes to those procedures are made in accordance with the provisions of TS 6.8.1

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

c. Review of all proposed changes to the Technical Specifications.

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

e. Review of all proposed changes to the Security Plan and Emergency Plan and their respective implementing procedures.

f. Review all reports covering the investigation of all violations of the Technical Specifications and the recommendations to prevent recurrence.

g. Review plant operations to detect potential safety hazards.

h. Performance of special reviews and investigations and prepare reports thereon as requested by the Chairman of the Nuclear Safety Review and Audit Committe.

i. Review of all Reportable Events

AUTHORITY

6.5.1.7 The PORC shall:

a. Recommend to the Plant Manager approval or disapproval of items considered under 6.5.1.6a through e above.
b. Make determinations with regard to whether or not each item considered under 6.5.1.6 above constitutes an unreviewed safety question.
c. Provide immediate notification in the form of draft meeting minutes to the Manager-Nuclear Power and the Chairman-Nuclear Safety Review and Audit Committee of disagreement between the PORC and the Plant Manager. The Plant Manager shall have responsibility for resolution of such disagreements.

RECORDS

6.5.1.8 Minutes shall be kept of all meetings of the PORC and copies shall be sent to the Manager - Nuclear Power and the Chairman - Nuclear Safety Review and Audit Committee.

6.5.2 CORPORATE NUCLEAR ENGINEERING STAFF (CNES)

FUNCTION

6.5.2.1 The CNES shall function to provide engineering,

(c) Disposition including date and destination if shipped offsite.

c. Deleted

6.10 RECORD RETENTION

6.10.1 The following records shall be retained for at least five years:

a. Records and logs of plant operation, including power levels and periods of operation at each power level.
b. Records and logs of principal maintenance activities, inspections, repair and replacement of principal items of equipment pertaining to nuclear safety.
c. Reports of all Reportable Events.
d. Records of periodic checks, inspections, and calibrations required by these Technical Specifications.
e. Records of nuclear safety related tests or experiments.
f. Records of radioactive shipments.
g. Records of changes to operating procedures.
h. Records of sealed source leak tests and results.
i. Records of annual physical inventory of all source material of record.
j. Records of Quality Assurance activities required by the Operational Quality Assurance Program (OQAP) except where it is determined that the records should be maintained for a longer period of time.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 63 TO FACILITY OPERATING LICENSE NO. DPR-43

WISCONSIN PUBLIC SERVICE CORPORATION

WISCONSIN POWER AND LIGHT COMPANY

MADISON GAS AND ELECTRIC COMPANY

KEWAUNEE NUCLEAR POWER PLANT

DOCKET NO. 50-305

Introduction

By letters dated August 24, 1983, June 29 and March 30, 1984 and March 19, 1985, Wisconsin Public Service Corporation (WPSC or the licensee) requested amendments to Facility Operating License No. DPR-43 for the Kewaunee Nuclear Power Plant (KNPP).

The August 24, 1983 letter, Proposed Amendment 55 (PA 55), requested miscellaneous changes to the Technical Specifications (TS) involving decay heat removal capability, engineered safety features and administrative controls. On June 29, 1984, (PA 55A), the licensee amended the above submittal to include the definition of the term "Operable". In response to our Generic Letter 83-37 regarding NUREG-0737 TS requirements the licensee submitted PA 59 dated March 30, 1984. We have previously issued a License Amendment, dated January 9, 1985, that implemented all the NUREG-0737 TS except for the control room post-accident recirculation system. On March 19, 1985(PA 59A), WPSC in response to questions by the staff modified PA 59. This license amendment completes our review of the NUREG-0737 TS by issuance of the control room post-accident recirculation system TS.

Evaluation of PA 55, PA 59 and PA 59A

A. Addition of Decay Heat Removal Capability Requirements (pages TS 3.1-1, 3.1-1a, 3.4-1, 3.4-1a, and 3.4-2)

Following a number of events at operating Pressurized Water Reactor (PWR) facilities, where decay heat removal capability had been seriously degraded due to inadequate administrative controls during shutdown modes, IE Bulletin 80-12 was issued on May 9, 1980. The Bulletin requested immediate implementation of administrative controls to ensure availability of redundant methods of decay heat removal. The licensee responded on June 20, 1980 that adequate safeguards existed at KNPP to protect against residual heat removal degradation after certain procedural changes were made.

On June 11, 1980, the staff requested all PWR licensees to propose TS changes that would provide for redundancy in decay heat removal in all operating modes. The licensee responded to the staff's request by letter, dated July 13, 1981, stating that the KNPP TS adequately addressed the staff's concerns. The staff reviewed this response and on July 11, 1983, again requested that changes to the KNPP TS be submitted covering:

1. Surveillance requirements consistent with the means of providing decay heat removal capability in the equivalence of Modes 1 and 2.

2. Technical Specifications meeting the intent of the Standard Technical Specification which require either another residual heat removal (RHR) loop to be operable or a reactor coolant loop available for decay heat removal in the equivalence of Modes 4 and 5. Consistent surveillance requirements should be included.

3. Technical Specifications that require one RHR train operating and an additional train operable when the water level above the vessel flange is allowed to be below 23 feet. If the additional train is unavailable the water level is required to be greater than 23 feet above the vessel flange while in the equivalence of Mode 6. Surveillance requirements consistent with these Technical Specifications should be included.

The licensee responded to this request by submitting PA 55 on August 24, 1983, which along with other TS changes addressed the above concerns.

Prior to the receipt of the above proposed amendment, the staff had completed a safety evaluation of the KNPP decay heat removal system and TS and concluded that although the likelihood of the plant losing both steam generators' decay heat removal capabilities while remaining in hot standby was very remote, the staff requested that KNPP's TS be modified to require both steam generators be operable during this mode. The staff also requested that surveillance requirements consistent with this requirement be included. The licensee, in PA 55, includes a new Section 3.1.a.2, Limiting Conditions for Operations, Decay Heat Removal Capability. Section 3.1.a.2.A covers reactor coolant temperature (RCT) between 200 and 350F, which is comparable to Mode 4, Hot Shutdown, in Westinghouse Standard Technical Specifications (STS), NUREG-0452 Revision 4. This part of the new section requires at least two of the four heat sinks (two steam generators and two RHR trains) be operable in this mode. A new Section 3.1.a.2.B covers RCT below 200F which is comparable to STS Modes 5 and 6 and requires two operable RHR trains below 200F.

In telecons on January 22, 1985 and March 25, 1985, the licensee agreed to add an action statement to Section 3.1.a.2.A and requested that note (a) be deleted since it was redundant with the newly revised Section 3.4. The licensee also requested that while in the refueling mode (below 140F) one RHR system could be inoperable for maintenance. This is consistent with existing TS Section 3.8.a.4 which requires only one operable RHR pump.

The basis for the addition of decay heat removal has been included on pages TS 3.1-2b and 2c along with other wording changes to clarify the basis.

The existing steam generator operability requirements on page TS 3.1-1 have been changed and moved to page TS 3.4-1, Section 3.4, Steam and Power Conversion System. The new Section 3.4.a now requires two operable steam generators when the RCT is above 350F, corresponding to STS Modes 1, 2, 3. This change provides the heat removal redundancy to meet the single failure criterion. The Basis on pages TS 3.4-2 and 3.4-3 have been modified to cover the above changes.

The staff has reviewed the above TS changes and find that they are responsive to the concerns in our letter dated July 11, 1983, the safety evaluation of September 20, 1983, and in general agree with STS requirements; therefore they are acceptable.

B. <u>Changes to Section 3.3, Engineered Safety Features (ESF) and Auxiliary Systems</u> (Pages TS 3.3-1, 3.3-2, 3.3-3, 3.3-4, 3.3-5, 3.3-6, 3.3-7, 3.3-8, 3.3-9, 3.3-10, 3.3-11, 3.3-12, and 3.7-2)

In a letter dated February 14, 1983, the staff requested the licensee to consider modifying the KNPP TS concerning Limiting Conditions of Operations for ESF Systems so that they would be compatible with the STS action statements (3/4 3.0.3).

In response to this request, on May 24, 1983, the licensee indicated that the existing KNPP TS adequately addressed the staff's concerns, but committed to amend the TS to incorporate STS instructions to eliminate certain ambiguities that existed. The PA 55 submittal contains a complete rewrite of TS Section 3.3., Engineered Safety Features and Auxiliary Systems, which includes the Accumulators, Safety Injections/Residual Heat Removal Systems, Containment Cooling Systems, Component Cooling Systems, and Service Water Systems. The new section is consistent with STS with the exception of operating modes in which ESF systems are required to be operable and in the length of time to achieve cold shutdown in case of inoperability. The existing KNPP TS requires both trains of ESF systems to be operable before the reactor is made critical, which compares to STS Modes 1 and 2, while STS in general requires operability in Modes 1, 2, and 3 and in some cases Mode 4. The existing and proposed TS require 48 hours to cold shutdown for inoperability while STS in most cases requires cold shutdown in 36 hours. The operating modes for ESF operability and the 48 hour transition to cold shutdown have been previously reviewed for these systems and approved by the staff and are therefore acceptable. The staff's concerns in the letter dated March 7, 1984 have also been resolved, since the new Containment Cooling Systems TS agree with STS and operability of a system is assumed if the required surveillances have been completed on the system.

PA 55 also adds the STS action statement to Specification 3.7.b, Auxiliary Electrical Systems. The proposed change requires the reactor to be in Hot Standby within 6 hours if operability of the specified systems cannot be restored to operability in the time limits specified.

The licensee in a telecon on March 25, 1985 requested to add Specification 3.7.c as follows:

"When its normal or emergency power source is inoperable, a system, train, or component may be considered operable for the purpose of satisfying the requirements of its applicable limiting condition for operation, provided:

1. Its corresponding normal or emergency power source is operable; and

2. Its redundant system, train, or component is operable."

A specification similar to the above was suggested in the Technical Evaluation forwarded by the Staff in the letter of December 19, 1983.

We have reviewed the above changes to TS Sections 3.3 and 3.7 and find them to be acceptable.

In addition to revising Section 3.3 to be compatible with STS format and action statements, the following changes were also proposed in PA 55.

1. Low power physics testing has been excluded from the operability requirements in Section 3.3 and this exception has been removed.

2. A new Subsection 3.3.a, Accumulators, has been included using STS format and applicable existing specifications found in Subsection 3.3.a, Safety Injection/Residual Heat Removal Systems.

3. Subsection 3.3.a, Safety Injections/Residual Heat Removal Systems, has been renumbered to 3.3.b and all subsequent subsections were also renumbered. In making the KNPP TS comparable to STS, the inoperative action limits have been increased from 24 hours for components to 72 hours for a train. An independent surveillance valve position verification was deleted since it is required in existing surveillance procedures.

4. As in 3.3.b above, the inoperative action limits for containment cooling systems have been increased to 7 days for 1 of 4 trains and to 72 hours for 2 of 4 trains inoperative. In addition the volume of required NaOH solution is now permitted to be contained in the containment spray system and spray additive tank rather than in the spray additive tank alone.

5. Both the Component Cooling Water (3.3.d) and Service Water (3.3.e) inoperative action limits have also been increased to 72 hours to make them compatible with STS.

6. The Basis for Section 3.3 has been expanded to include some of the above changes and expand the Basis in other areas.

The staff has reviewed the above miscellaneous changes to Section 3.3 and find them to be consistent with STS requirements or with original licensing basis and therefore they are acceptable.

C. Refueling (TS pages 3.8-1, 3.8-2, and 3.8-2a)

Section 3.8, Refueling, has been modified to make containment integrity requirements consistent with other KNPP TS and the STS. Other minor changes were made to delete obsolete requirements and clarify wording.

The staff finds these changes to be acceptable since they are consistent with STS.

D. Emergency Ventilation Systems (Pages TS 3.12-1, 3.12-2, 4.4-5, 4.4-6, 4.4-7, 4.4-11, 4.4-12, 4.12-1, 4.12-2, 4.12-3, 4.17-1, 4.17-2, and 4.17-3).

The licensee in PA 59 responded to Generic Letter 83-37, NUREG-0737, TMI Technical Specifications by submitting new TS covering the Control Room Post-accident Recirculation System (CRPRS), Section 3.12 by letter dated March 30, 1984 and as amended March 19, 1985. In PA 55 the licensee had submitted changes to the surveillance TS for the Shield Building Ventilation System (SBVS), the Auxiliary Building Special Ventilation System (ABSVS), and the Spent Fuel Pool Sweep System (SFPSS).

The new CRPRS TS (Pages TS 3.12-1, 3.12-2) provide requirements and Basis that are essentially identical with the present TS for the SBVS, ABSVS and SFPSS and in general are consistent with STS 3/4.7.7.

The surveillance TS for the SBVS, ABSVS, and SFPSS submitted in PA 55 and the new TS for the CRPRS in PA 55a are identical except where the systems differ; that is, electrical heater testing in SBVS and door interlock and heater testing in ABSVS. Other than rearrangement to make the TS consistent, the only change from existing TS was the deletion of the required DOP testing for the HEPA filters following painting, fire, or chemical release in any ventilation zone communicating with the system. All testing requirements for the charcoal filters remain the same. The licensee states that this TS is overly restrictive since no matter how well the HEPA filters are protected during painting, or welding, etc., DOP testing would still be required. There are TS requirements for differential pressure and design flow. Any degradation of the HEPA filters from plugging would be clearly evident and plugging is the only probable HEPA filter degradation if fire is not in close proximity to the filters. Therefore, differential pressure would be a more reasonable test than a DOP test. In a telecon on April 2, 1985, the licensee committed to including monthly HEPA differential pressure readings in plant surveillance procedures for the SFPSS which runs continuously and to take differential pressure readings during the required monthly operational tests of the SBVS, ABSVS, and CRPRS.

The staff has reviewed the above change and concludes that the monthly differential pressure readings would be more effective in predicting HEPA filter degradation from painting, welding, fire, and the release of chemical vapors than a DOP test. The DOP test requirement is still required if any work is conducted on the HEPA filter system and at least every 18 months.

The staff agrees that with the commitment to take monthly differential pressure readings the above changes are acceptable.

The licensee's submittal of March 19, 1985, was made as a result of a NRC staff request to delete an ambiguity in the TS and does not contain substantive changes.

Miscellaneous Changes (Tables TS 3.5-3 [pages 1 and 2], TS 3.5-4 [page 2], TS 4.1-2 [pages 1 and 2), TS 4.1-3 [pages 1 and 2], and TS pages 3.1-2, 3.2-2, 6-4, 6-5, 6-23, i, ii, iii, and v)

In Table TS 3.5-3 the licensee added the requirement for four operable level channels for the selected Boric Acid Storage Tank. This requirement was added since these liquid level instruments have an automatic safety function of switching SI suction from the Boric Acid Storage Tank to the Refueling Water Storage Tank when the Boric Acid is near depletion. This addition also required renumbering other items in this table and in Table TS 3.5-4 (page 12). In Table TS 4.1-2 Note (1) was added which deletes Reactor Coolant Boron samples and analysis when there is no fuel in the reactors and Note (2) was added to delete sampling and analysis of the refueling water storage tank when it is empty during refueling outages. Other minor wording changes were made to correct a previous TS change in Amendment 13 that was accidentally deleted by Amendment 18.

In PA 59a, a revision to PA 59 dated March 19, 1985, the licensee requested to delete Item 6 and the associated note in Table TS4.1-3 concerning the testing requirements of the emergency ventilation system filters. Since these testing requirements are now included in the individual system surveillance sections, they were redundant in the table.

The staff has reviewed the above changes and find that they increase the effectiveness and clarity of the TS or are administrative changes and are therefore acceptable.

Proposed changes to Page TS 3.1-2 in PA 55 included a clarification that the pressure relief of the RCS during hydrotesting with the safety valves blocked pertained to both the power operated relief valves as well as the charging pump discharge. Note (a) was revised to require the power breakers for the motor operated RCS pressure isolation valves be locked out with the valves closed during operation. This change gives valve position indication in the control room and at the same time maintains the valves in the correct position.

A minor word change was made on page TS 3.2-2 for clarification.

The responsibilities of the Plant Operations Review Committee (PORC) were changed on Page TS 6-4 to include the review of all proposed changes to the Security Plan and the Emergency Plan and their respective implementing procedures. Subsequent paragraphs in the section and Page TS 6-5 were renumbered to agree with the new review requirements. Organization name changes were also corrected on Page TS 6-5 and 6-23 and an obsolete requirement on TS 6-23 was deleted.

The Table of Contents page i, ii, iii, and v were updated to reflect the changes in PA 55, PA 55a, PA 59, and PA59a.

The staff has reviewed the changes and conclude that they are acceptable based on the above evaluations.

Evaluation of Definition of "OPERABLE" (PA 55A)

The changes proposed by the licensee, discussed in detail below, implement the guidance from the Commission regarding the definition of OPERABLE, and, do not adversely affect the safety of the plant nor the general public. The changes consist of revising the definition of OPERABLE, and pagination changes to accommodate same.

The staff concludes that the changes as described in the proposed amendment are therefore acceptable.

Proposed Change to TS i

Description of Change

The Table of Contents has been revised to reflect the proposed change.

Evaluation

This change is purely administrative in nature and has no effect on safety. This change is acceptable.

Proposed Change to TS 1.1-2, TS 1.1-2a

Description of Change

The definition of OPERABLE is revised to reflect the guidance transmitted to the licensee by letter dated December 19, 1983. TS sections 1.0.g and 1.0.h are moved to new page 1.1-2a as a result of the revised definition.

Evaluation

The revised definition is consistent with the Model Technical Specification, and ensures that necessary support equipment and normal and emergency electrical power supplies are available to maintain operability of Engineered Safeguards equipment within the single failure criterion. Incorporation of the new definition implements the guidance of the December 19, 1983 letter. The changes to TS sections 1.0.g and 1.0.h are simply pagination changes and are purely administrative in nature. This change is acceptable.

Environmental Consideration

This amendment involves a change in the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20. The staff has determined that the amendment involves no significant increase in the amounts, and no significant change in the types, of any effluents that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously issued a proposed finding that this amendment involves no significant hazards consideration and there has been no public comment on such finding. Accordingly, this amendment meets the eligibility criteria for categorical exclusion set forth in 10 CFR Sec 51.22(c)(9). Pursuant to 10 CFR 51.22(b) no environmental impact statement or environmental assessment need be prepared in connection with the issuance of this amendment.

Conclusion

We have 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, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Dated: July 5, 1985

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