DCS MS-016
CP S1

NOV 8 1982

Docket Nos. 50-266
and 50-301

Mr. C. W. Fay
Assistant Vice President
Wisconsin Electric Power Company
231 West Michigan Street
Milwaukee, Wisconsin 53201

Dear Mr. Fay:

The Commission has issued the enclosed Amendment No. 66 to Facility Operating License No. DPR-24 and Amendment No. 71 to Facility Operating License No. DPR-27 for the Point Beach Nuclear Plant, Unit Nos. 1 and 2, respectively. The amendments consist of changes to the Technical Specifications in response to your application transmitted by letter dated November 16, 1981 as modified by letter dated May 3, 1982.

These amendments upgrade the existing Technical Specifications in order to provide for redundancy of decay heat removal capability in all modes of operation for Point Beach Units 1 and 2.

Your May 3, 1982 submittal addressed the staff concerns with your previous November 16, 1981 submittal as identified in our January 22, 1982 letter. However, your May 3, 1982 submittal included additional proposed Technical Specifications (TS) which the staff feels do not meet the intent of providing redundancy for decay heat removal. Proposed TS 15.3.1.A.3.a(5) allows one of the two operable means of decay heat removal to be temporarily out of service to meet surveillance requirements.

This proposed TS was not part of your November 16, 1981 submittal. Nor does the basis provided adequately justify this proposed TS. The removal from service of the associated RHR loop to perform surveillance has no associated time limit. Thus, if two RHR loops were the redundant methods of decay heat removal being used, removing one from service for surveillance testing would allow for a temporary loss of all decay heat removal capability given a single failure of the operating RHR loop. For this reason, and because reliance on a reactor coolant loop, reactor coolant pump and associated steam generator is allowed as a method of decay heat removal in both modes 4 and 5, the staff feels that adequate flexibility would exist to perform RHR system surveillance testing without issuance of this proposed TS. Therefore, as discussed with your staff, we are not approving this proposed change.

OFFICE ▶							
SURNAME ▶							
DATE ▶							

Additionally, your proposed TS 15.3.3.A.3 states that if one RHR loop is inoperative during power operation and repairs are not completed within an additional 48 hours, the reactor shall be maintained between 500°F and 350°F in order to allow two steam generators and associated reactor coolant loops to provide the redundant methods of decay heat removal. As discussed with your staff, we are modifying your proposed TS to require maintenance of reactor coolant temperature between 350°F and 140°F where the remaining RHR loop must be relied on in conjunction with a steam generator and associated reactor coolant loop to provide the required redundancy in decay heat removal capability.

The basis statements have also been modified to be consistent with the above changes.

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

Sincerely,

Original signed by

Timothy G. Colburn, Project Manager
Operating Reactors Branch #3
Division of Licensing

Enclosures:
1. Amendment No. 66 to DPR-24
2. Amendment No. 71 to DPR-27
3. Safety Evaluation
4. Notice of Issuance

cc: w/enclosures
See next page

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AMENDMENT

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

WISCONSIN ELECTRIC POWER COMPANY

DOCKET NO. 50-266

POINT BEACH NUCLEAR PLANT, UNIT NO. 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 66
License No. DPR-24

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

   A. The application for amendment by Wisconsin Electric Power Company (the licensee) dated November 16, 1981 as modified May 3, 1982, 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 3.B of Facility Operating License No. DPR-24 is hereby amended to read as follows:

   B. Technical Specifications

      The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 66 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective 20 days from the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: November 8, 1982

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

WISCONSIN ELECTRIC POWER COMPANY

DOCKET NO. 50-301

POINT BEACH NUCLEAR PLANT, UNIT NO. 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 71
License No. DPR-27

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

   A. The application for amendment by Wisconsin Electric Power Company (the licensee) dated November 16, 1981 as modified May 3, 1982, 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 3.B of Facility Operating License No. DPR-27 is hereby amended to read as follows:

   B. Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 71 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective 20 days from the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark

Robert A. Clark, Chief
Operating Reactors Branch #3
Division of Licensing

ent:
s to the Technical
Specifications

Date of Issuance: November 8, 1982

ATTACHMENT TO LICENSE AMENDMENTS

AMENDMENT NO. 66 TO FACILITY OPERATING LICENSE NO. DPR-24

AMENDMENT NO. 71 TO FACILITY OPERATING LICENSE NO. DPR-27

DOCKET NOS. 50-266 AND 50-301

Revise Appendix A as follows:

Remove Pages	Insert Pages
15.3.1-1	15.3.1-1
15.3.1-2	15.3.1-2
15.3.1-3	15.3.1-3
15.3.1-3a	15.3.1-3a
-	15.3.1-3b
15.3.3-2	15.3.3-2
15.3.3-2a	15.3.3-2a
15.3.3-8	15.3.3-8
15.3.3-9	15.3.3-9
15.3.8-1	15.3.8-1

15.3 LIMITING CONDITIONS FOR OPERATION

15.3.1 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.

Specification

A. OPERATIONAL COMPONENTS

1. Coolant Pumps

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

   b. When the reactor is critical and above 1% of rated power except for natural circulation tests, at least one reactor coolant pump shall be in operation.

   c. (1) Reactor power shall not be maintained above 10% of rated power unless both reactor coolant pumps are in operation.

      (2) If either reactor coolant pump ceases operating, immediate power reduction shall be initiated under administrative control as necessary to reduce power to less than 10% of rated power.

2. Steam Generator

   a. One steam generator shall be operable whenever the average reactor coolant temperature is above 350°F.

3. Components Required for Redundant Decay Heat Removal Capability

   a. Reactor coolant temperature less than 350°F and greater than 140°F.

      (1) At least two of the decay heat removal methods listed shall be operable.

         (a) Reactor Coolant Loop A, its associated steam generator and either reactor coolant pump

         (b) Reactor Coolant Loop B, its associated steam generator and either reactor coolant pump

(c) Residual Heat Removal Loop (A)*

(d) Residual Heat Removal Loop (B)*

(2) If the conditions of specification (1) above cannot be met, corrective action to return a second decay heat removal method to operable status as soon as possible shall be initiated immediately.

(3) At least one of the above decay heat removal methods shall be in operation except when required to be secured for testing.

(4) If no decay heat removal method is in operation, all operations causing an increase in the reactor decay heat load or a reduction in reactor coolant system boron concentration shall be suspended. Corrective actions to return a decay heat removal method to operation shall be initiated immediately.

b. Reactor Coolant Temperature Less Than 140°F

(1) Both residual heat removal loops shall be operable except as permitted in items (3) or (4) below.

(2) If no residual heat removal loop is in operation, all operations causing an increase in the reactor decay heat load or a reduction in reactor coolant system boron concentration shall be suspended. Corrective actions to return a decay heat removal method to operation shall be initiated immediately.

(3) One residual heat removal loop may be out of service when the reactor vessel head is removed and the refueling cavity flooded.

(4) One of the two residual heat removal loops may be temporarily out of service to meet surveillance requirements.

4. Pressurizer Safety Valves

a. At least one pressurizer safety valve shall be operable whenever the reactor head is on the vessel.

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

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*Mechanical and electrical design provisions of the residual heat removal system afford the necessary flexibility to allow an operable residual heat removal loop to consist of the RHR pump from one loop coupled with the RHR heat exchanger from the other loop and to allow the normal or emergency power source to be inoperable or tied together when the reactor coolant temperature is less than 200°F.

5. Pressurizer Power Operated Relief Valves (PORV) and PORV Block Valves
   a. Two PORVs and their associated block valves shall be operable.
      (1) If a PORV is inoperable, the PORV shall be restored to an operable condition within one hour or the associated block valve shall be closed.
      (2) If a PORV block valve is inoperable, the block valve shall be restored to an operable condition within one hour or the block valve shall be closed with power removed from the block valve; otherwise the unit shall be in hot shutdown within the next six hours.
6. The pressurizer shall be operable with at least 100 KW of pressurizer heaters available and a water level greater than 10% and less than 95% during steady-state power operation. At least one bank of pressurizer heaters shall be supplied by an emergency bus power supply.

Basis

When the boron concentration of the reactor coolant system is to be reduced, the ess must be uniform to prevent sudden reactivity changes in the reactor. of the reactor coolant will be sufficient to maintain a uniform boron concentration if at least one reactor coolant pump or one residual heat removal pump is running while the change is taking place. The residual heat removal pump will circulate the primary system volume in approximately one-half hour. The pressurizer is of little concern because of the lower pressurizer volume and because pressurizer boron concentration normally will be higher than that of the rest of the reactor coolant.

Specification 15.3.1.A.1 requires that a sufficient number of reactor coolant pumps be operable to provide core cooling in the event a loss of power occurs. The flow provided in each case will keep DNBR well above 1.30 as discussed in FFDSAR, Section 14.1.9. Therefore, cladding damage and release of fission products to the reactor coolant will not occur. Heat transfer analyses[(1)] show that reactor heat equivalent to 10% of rated power can be removed with natural circulation only; hence the specified upper limit of 1% rated power without operating pumps provides a substantial safety factor.

Item 15.3.1.A.1.c.(2) permits an orderly reduction in power if a reactor coolant pump is lost during operation between 10% and 50% of rated power.

Above 50% power, an automatic reactor trip will occur if either pump is lost. The power-to-flow ratio will be maintained equal to or less than 1.0, which ensures that the minimum DNB ratio increases at lower flow since the maximum enthalpy rise does not increase above its normal full-flow maximum value.[(2)]

Specification 15.3.1.A.3 provides limiting conditions for operation to ensure that redundancy in decay heat removal methods is provided. A single reactor coolant loop with its associated steam generator and a reactor coolant pump or a single residual heat removal loop provides sufficient heat removal capacity for removing the reactor core decay heat; however, single failure considerations require that at least two decay heat removal methods be available. Operability of a steam generator for decay heat removal includes two sources of water, water level indication in the steam generator, a vent path to atmosphere, and the Reactor Coolant System filled and vented so thermal convection cooling of the core is possible. If the steam generators are not available for decay heat removal, this Specification requires both residual heat removal loops to be operable unless the reactor system is in the refueling shutdown condition with the refueling cavity flooded and no core alterations in progress. In this condition, the reactor vessel is essentially a fuel storage pool and removing a RHR loop from service provides conservative conditions should operability problems develop in the other RHR loop. Also, one residual heat removal loop may be temporarily out of service due to surveillance testing, calibration, or inspection requirements. The surveillance procedures follow administrative controls which allow for timely restoration of the residual heat removal loop to service if required.

Each of the pressurizer safety valves is designed to relieve 288,000 lbs. per hour of saturated steam at setpoint. If no residual heat is removed by any of the means available, the amount of steam which could be generated at safety valve relief pressure would be less than half the valves' capacity. One valve, therefore, provides adequate defense against overpressurization. Below 350°F and 400 psig in the Reactor Coolant System, the residual heat removal system can remove decay heat and thereby control system temperature and pressure.

A PORV is defined as OPERABLE if leakage past the valve is less than that allowed in Specification 15.3.1.D and the PORV has met its most recent channel test as specified in Table 15.4.1-1. The PORVs operate to relieve, in a controlled manner, reactor coolant system pressure increases below the setting of the pressurizer safety valves. These PORVs have remotely operated block valves to provide a positive shutoff capability should a PORV become inoperable.

The requirement that 100 KW of pressurizer heaters and their associated controls be capable of being supplied electrical power from an emergency bus provides assurance that these heaters can be energized during a loss of offsite power condition to maintain pressure control and natural circulation at hot standby.

References

(1) FSAR Section 14.1.6

(2) FSAR Section 7.2.3

safety injection system are in the open position.

g. All valves, interlocks, and piping associated with the above components and required to function during accident conditions are operable.

h. During conditions of operation with reactor coolant system pressure in excess of 1,000 psig, the source of AC power shall be removed from the accumulator isolation valves MOV-841A and B at the motor control center and the valves shall be open.

i. Power may be restored to MOV-841A and B for the purpose of valve testing or maintenance providing the testing and maintenance is completed and power is removed within four hours.

2. During power operation, the requirements of 15.3.3.A.1, Items b and c, may be modified to allow one of each of the following components to be inoperable at any one time. If the system is not restored to meet the requirements of 15.3.3.A.1 within the time period specified, the reactor shall be placed in the hot shutdown condition. If the requirements of 15.3.3.A.1 are not satisfied within an additional 48 hours, the reactor shall be placed in the cold shutdown condition.

   a. One accumulator may be isolated for a period of up to one hour to permit a check valve leakage test. Before isolating an accumulator, the other accumulator isolation valve shall be checked open.

   b. One safety injection pump may be out of service, provided the pump is restored to operable status within 24 hours. The other safety injection pump shall be tested to demonstrate operability prior to initiating repair of the inoperable pump.

   c. Any valve in these systems required to function during accident conditions may be inoperable provided repairs are completed within 24 hours. Prior to initiating repairs, all valves in the system that provide the duplicate function shall be tested to demonstrate operability.

3. During power operation, the requirements of 15.3.3.A.1, Items d and e, may be modified to allow one of each of the following components to be inoperable at any one time. If the component is not restored to meet

the requirements of 15.3.3.A.1 within the time specified, the reactor shall be placed in the hot shutdown condition. If the requirements of 15.3.3.A.1 are not satisfied within an additional 48 hours, the reactor shall be maintained in a condition with reactor coolant temperatures between 500 and 350°F, unless one residual heat removal loop is being relied upon to provide redundancy for decay heat removal. In this case the reactor shall be maintained between 350° and 140°F.

a. One residual heat removal pump may be out of service, provided the pump is restored to operable status within 24 hours. The other residual heat removal pump shall be tested to demonstrate operability prior to initiating repair of the inoperable.pump.
b. One residual heat exchanger may be out of service for a period of no more than 48 hours.
c. Any valve in the system, required to function during accident conditions, may be inoperable provided repairs are completed within 24 hours. Prior to initiating repairs, all valves in the system that provide the duplicate function shall be tested to demonstrate operability.

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.	3.6
30 min.	1.55
1 hour	1.25
8 hours	0.7
48 hours	0.4

*Based on ANS 5.1-1979, "Decay Heat Power in Light-Water Reactors"

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 system components in order to effect repairs.

Failure to complete safety injection system repairs within 48 hours of going to the hot shutdown condition is considered indicative of a requirement for major maintenance and, therefore, in such a case, the reactor is to be put into the cold shutdown condition. When the failures involve the residual heat removal system, in order to insure redundant means of decay heat removal, the reactor system may remain in a condition with reactor coolant temperatures between 500 and 350°F so that the reactor coolant loops and associated steam generators may be utilized for redundant decay heat removal. However, when the remaining RHR loop must be relied upon for redundant decay heat removal capability, reactor coolant temperatures shall be maintained between 350°F and 140°F.

With respect to the core cooling function, there is some functional redundancy for certain ranges of break sizes.[2]

The containment cooling function is provided by two independent systems: (a) fan coolers and (b) containment spray which, with sodium hydroxide addition, provides the iodine removal function. During normal power operation, only three of the four fan coolers are required to remove heat lost from equipment and piping within the containment.[3] In the event of a Design Basis Accident, any one of the following combinations will provide sufficient cooling to reduce containment pressure: (1) four fan coolers, (2) two containment spray pumps, (3) two fan coolers plus one containment spray pump.[4] Sodium hydroxide addition via one spray pump reduces airborne iodine activity sufficiently to limit off-site doses to acceptable values. One of the four fan coolers is permitted to be inoperable when the reactor is made critical and during power operation.

The component cooling system is different from the other systems discussed above in that the components are so located in the Auxiliary Building as to be accessible for repair after a loss-of-coolant accident. One component cooling water pump together with one component cooling heat exchanger can accommodate the heat removal load on one unit either following a loss-of-coolant accident, or during normal plant shutdown. If during the post-accident phase the component cooling water supply is lost, core and containment cooling could be maintained until repairs were effected.[5]

A total of six service water pumps are installed, only three of which are required to operate during the injection and recirculation phases of a postulated loss-of-coolant accident,[6] in one unit together with a hot shutdown condition in the other unit.

References

(1) FSAR Section 3.2.1

(2) FSAR Section 6.2

(3) FSAR Section 6.3.2

(4) FSAR Section 6.3

(5) FSAR Section 9.3.2

(6) FSAR Section 9.6.2

15.3.8 REFUELING AND SPENT FUEL ASSEMBLY STORAGE

Applicability:

Applies to operating limitations during refueling operations and to operating limitations concerning the movement of heavy loads over or into the spent fuel storage pools.

Objective:

To ensure that no incident could occur during refueling operations, or during auxiliary building crane operations that would affect public health and safety.

Specifications:

A. During refueling operations:

1. The equipment hatch shall be closed and the personnel locks shall be capable of being closed. A temporary third door on the outside of the personnel lock shall be in place whenever both doors in a personnel lock are open (except for initial core loading).
2. Radiation levels in fuel handling areas, the containment and spent fuel storage pool shall be monitored continuously.
3. 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 available 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 loop shall be in operation. However, if refueling operations are affected by the residual heat removal loop flow, the operating residual heat removal loop may be removed from operation for up to one hour per eight hour period.
5. During reactor vessel head removal and while loading and unloading fuel from the reactor, a minimum boron concentration of 1800 ppm shall be maintained in the primary coolant system.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 66 TO FACILITY OPERATING LICENSE NO. DPR-24

AND AMENDMENT NO. 71 TO FACILITY OPERATING LICENSE NO. DPR-27

WISCONSIN ELECTRIC POWER COMPANY

POINT BEACH NUCLEAR PLANT, UNIT NOS. 1 AND 2

DOCKET NOS. 50-266 AND 50-301

Introduction

On April 19, 1980, a loss of decay heat removal (DHR) capability occurred at Davis-Besse Unit 1. This was the subject of IE Information Notice 80-20 dated May 8, 1980. This incident also prompted issuance of IE Bulletin 80-12 transmitted to Wisconsin Electric Power Company (licensee) on May 12, 1980 which required licensees of pressurized water reactors (PWRs) to conduct reviews of the susceptibility of decay heat removal capability for their facilities and implement immediate procedural and administrative controls where needed to reduce the likelihood of such an event.

The licensee responded to the above bulletin by letter dated June 5, 1980 outlining their procedural changes and administrative controls effected to achieve redundancy of DHR capability in all modes of operation.

Subsequent to the licensee's response, the NRC staff transmitted to licensees of all PWRs by letter dated June 11, 1980, a request that they amend the Technical Specifications (TS) for their facilities to ensure redundancy of DHR capability in all modes of operation. Attached to the staff's letter were sample standard TS.

The licensee responded to this request by letter dated October 14, 1980. By letter dated August 14, 1981, the staff transmitted their review of the licensee's response to this issue and again requested that the licensee amend the Point Beach Unit 1 and 2 TS. The licensee responded to the staff's request by letter dated November 16, 1981 as modified by letter dated May 3, 1982.

Discussion and Evaluation

The intent of IE Bulletin 80-12 was to improve nuclear power plant safety by reducing the likelihood of losing DHR capability in operating PWRs. PWRs are most susceptible to losing DHR capability when their steam generators or other diverse means of removing decay heat are not readily available. Such conditions often occur when the plants are in a refueling or cold shutdown mode, and during which time concurrent maintenance activities are being performed.

There is a need to assure that all reasonable means have been taken to provide redundant or diverse means of DHR during all modes of operation. (Note: A redundant means could be provided by having DHR Train A AND Train B operable; a diverse means could be provided by having either DHR Train A OR Train B operable AND a steam generator available for DHR purposes.) There is also need to assure that all reasonable means have been taken to preclude the loss of DHR capability due to common mode failures during all modes of operation.

The licensee's November 16, 1981 letter requested changes to the Point Beach Units 1 and 2 TS which the licensee believed would satisfy the staff's concerns regarding redundancy of DHR capability in operational modes 4 and 5. In their August 14, 1981 letter to the licensee the NRC staff had concluded that the existing Point Beach Units 1 and 2 TS adequately addressed this issue in all but operational modes 4 and 5. The staff further evaluated the licensee's administrative controls as adequately providing interim assurance of redundancy of DHR capability until final resolution of this issue.

The NRC staff reviewed the licensee's proposed TS and found them unacceptable for reasons identified in the staff's January 22, 1981 letter. In addition to concerns relating to the licensee's proposed TS, the staff identified two additional concerns. One of these related to checking operability of a component prior to taking its redundant component out of service to conduct repairs or tests. Specifically, this concern related to accumulator check valve leakage tests.

The other additional staff concerns related to the ability of a single residual heat removal (RHR) loop to provide sufficient heat removal capacity immediately following shutdown from extended operation at full power. Inability of a single RHR loop to adequately remove reactor decay heat immediately following shutdown would mean that initially either two steam generators and their associated reactor coolant loops or both RHR loops and one steam generator and its associated reactor coolant loop would be required to meet the redundancy criteria.

The licensee modified their proposed TS to address the NRC staff concerns by letter dated May 3, 1982. The staff has reviewed the licensee's proposed TS, as modified, and finds that they adequately address the staff concerns regarding redundancy of DHR capability in operational modes 4 and 5. Additionally, they address the NRC staff's concerns regarding accumulator check valve leakage testing and the ability of a single RHR loop to provide adequate decay heat removal capacity following extended operation at full power. However, the licensee's May 3, 1982 submittal included additional proposed Technical Specifications (TS) which the staff feels do not meet the intent of providing redundancy for decay heat removal. Proposed TS 15.3.1.A.3.a(5) allows one of the two operable means of decay heat removal to be temporarily out of service to meet surveillance requirements.

The proposed TS was not part of the licensee's November 16, 1981 submittal. Nor does the basis provided adequately justify this proposed TS. The removal from service of the associated RHR loop to perform surveillance has no associated time limit. Thus, if two RHR loops were the redundant methods of decay heat removal being used, removing one from service for surveillance testing would allow for a temporary loss of all decay heat removal capability given a single failure of the operating RHR loop. For this reason, and because reliance on a reactor coolant loop, reactor coolant pump and associated steam generator is allowed as a method of decay heat removal in both modes 4 and 5, the staff feels that adequate flexibility would exist to perform RHR system surveillance testing without issuance of this proposed TS. Therefore, the staff is not approving this proposed change.

In light of their more recent analysis of the ability of a single RHR loop to provide adequate decay heat removal capability, the licensee proposed modification of the table in the basis of TS 15.3.3 to include the predicted decay heat vs. time values in the American National Standard ANS 5.1, 1979 "Decay Heat Power in Light Water Reactors". The staff finds this acceptable as clarification to support the ability of a single RHR loop to adequately remove reactor decay heat.

Environmental Consideration

We have determined that the amendments do not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendments involve an action which is insignificant from the standpoint of environmental impact and, pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of these amendments.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendments do not involve a significant increase in the probability or consequences of an accident previously evaluated, do not create the possibility of an accident of a type different from any evaluated previously, and do not involve a significant reduction in a margin of safety, the amendments do not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Date: November 8, 1982

Principal Contributor:

T. Colburn

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NOS. 50-266 AND 50-301

WISCONSIN ELECTRIC POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENTS TO FACILITY OPERATING LICENSES

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 66 to Facility Operating License No. DPR-24, and Amendment No. 71 to Facility Operating License No. DPR-27 issued to Wisconsin Electric Power Company (the licensee), which revised Technical Specifications for operation of Point Beach Nuclear Plant, Unit Nos. 1 and 2 (the facilities) located in the Town of Two Creeks, Manitowoc County, Wisconsin. The amendments are effective 20 days from the date of issuance.

The amendments upgrade the existing Technical Specifications for Point Beach Units 1 and 2 in order to provide for redundancy of decay heat removal during all modes of operation.

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

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

For further details with respect to this action, see (1) the application for amendments dated November 16, 1981 as modified by letter dated May 3, 1982, (2) Amendment Nos. 66 and 71 to License Nos. DPR-24 and DPR-27, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C. 20555, and at the Joseph Mann Library, 1516 16th Street, Two Rivers, Wisconsin 54241. A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 8th day of November, 1982.

FOR THE NUCLEAR REGULATORY COMMISSION

Robert A. Clark

Robert A. Clark, Chief
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