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

CONSUMERS POWER COMPANY

DOCKET NO. 50-155

BIG ROCK POINT PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 10
License No. DPR-6

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

   A. The applications for amendment by the Consumers Power Company (the licensee) dated July 25, 1975 and supplements thereto dated August 22, 1975, September 8, 1975, November 26, 1975, February 4, 1976, February 27, 1976, March 26, 1976, April 30, 1976, May 10, 1976 and May 11, 1976; October 13, 1975, as modified by letters dated April 28, 1976 and May 11 and 25, 1976; August 15, 1974, as modified by letters dated November 14, 1974, December 17, 1974, March 10, 1975, April 29, 1975 and October 9, 1975; and December 5, 1975, comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and, that in view of the exemption granted by the Commission on May 26, 1976, comply with 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. After weighing the environmental aspects involved, 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. On May 26, 1976, the Commission by Memorandum and Order granted to the Consumers Power Company certain exemptions from the requirements of 10 CFR 50.46 for the captioned facility.

3. Accordingly, Facility Operating License No. DPR-6 is hereby amended as follows:

   A. Change the Technical Specifications as indicated in the attachment to this license amendment.

   B. Revise item 2.C(2) of the license to read:

      (2) Technical Specifications

      The Technical Specifications contained in Appendix A as issued May 1, 1964, as revised, are hereby incorporated in this license. The licensee shall operate the facility in accordance with the Technical Specifications as revised.

   C. Add the following as item 2.C(3) of the license:

      (3) The licensee is subject to the conditions set forth in Section III.d of the Commission's Memorandum and Order dated May 26, 1976.

4. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Karl R. Goller

Karl R. Goller, Assistant Director
for Operating Reactors
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: June 4, 1976

ATTACHMENT TO LICENSE AMENDMENT NO. 10

FACILITY OPERATING LICENSE NO. DPR-6

DOCKET NO. 50-155

The Technical Specifications attached to Facility Operating License No. DPR-6 are hereby changed as follows:

1. Replace Contents page iv with the attached new page.

2. Delete Section 3.5.2(a) in its entirety.

3. Delete Section 3.5.2(c) in its entirety.

4. Replace Section 3.7(e) with the following:

   "An integrated leakage rate test shall be conducted on the containment sphere at approximately three equal intervals during each 10-year service period."

5. Delete the following phrases in chronological order from Section 4.1.2(b):

   ", core spray and backup core spray systems"

   and

   "core spray system,"

   and

   "and the fire water makeup system to the condenser hotwell"

   and

   "core spray system and".

6. Delete Section 4.2.1(a) in its entirety.

7. Delete Section 4.2.1(b) in its entirety.

8. Delete the last sentence in Section 4.2.6.

9. Delete the columns entitled Reload B & C and Reload E and add the following column to Table 5.1.

General	Reload G-1U
Geometry, Fuel Rod Array	11 x 11
Rod Pitch, Inches	0.577
UO_2 Rods	109
Cobalt - Bearing Corner Rods	4
Gadolinium - Bearing UO_2 Rods	4
Inert Spacer Capture Rod (Zr-2)	1
Zircaloy Rods	3
Spacers per Bundle	3
Fuel Rod Cladding	
Material	Zr-2
Wall Thickness, Inches	0.034
Fuel Rods	
Outside Rod Diameter, Inches	0.449
Fuel Stacked Density, Percent Theoretical	91.6
Active Fuel Length, Inches Standard Rod	70
Fill Gas	Helium $\geq$ 95%

10. Delete the references to Reloads B, C and E in Note 1 of Table 5.1.

11. Delete the reference to Reload E in Note 3 of Table 5.1.

12. Change Section 5.2.1(b) to read as follows:

5.2.1(b) Reactor Operation

The reactor operation shall be so limited as to be consistent with the most conservative of the following:

TABLE 1

	Reload E-G and Modified E-G F & J-2	Reload G	Reload G-1U
Minimum Core Burnout Ratio at Overpower	1.5*	1.5**	1.5**
Transient Minimum Burnout Ratio in Event of Loss of Recirculation Pumps From Rated Power	1.5	1.5	1.5
Maximum Heat Flux at Overpower, $Btu/h\text{-}ft^2$	500,000	395,000	407,000
Maximum Steady State Heat Flux, $Btu/h\text{-}ft^2$	410,000	324,000	333,600
Maximum Average Planar Linear Heat Generation Rate, Steady State, kW/Ft	***	***	***
Stability Criterion: Maximum Measured Zero-to-Peak Flux Amplitude, Percent of Average Operating Flux	20	20	20
Maximum Steady State Power Level, MW_t	240	240	240
Maximum Value of Average Core Power Density @ 240 MW_t, kW/L	46	46	46
Nominal Reactor Pressure During Steady State Power Operation, psig	1335	1335	1335
Minimum Recirculation Flow Rate, Lb/h (Except During Pump Trip Tests or Natural Circulation Tests as Outlined in Section 8)	6×10^6	6×10^6	6×10^6
Maximum MWd/T of Contained Uranium for an Individual Bundle	23,500	23,500	23,500

Rate-of-Change-Of-Reactor Power During Power Operation:

Control rod withdrawal during power operation shall be such that the average rate-of-change-of-reactor power is less than 50 MW_t per minute when power is less than 120 MW_t, less than 20 MW_t per minute when power is between 120 and 200 MW_t, and 10 MW_t per minute when power is between 200 and 240 MW_t.

*Based on correlation given in "Design Basis for Critical Heat Flux Condition in Boiling Water Reactors," by J. M. Healzer, J. E. Hench, E. Janssen and S. Levy, September 1966 (APED 5286 and APED 5286, Part 2).

**Based on Exxon Nuclear Corporation Synthesized Hench Levy.

***To be determined by linear extrapolation from Table 2 attached.

13. Following the table in Section 5.2.1(b), add the attached Table 2 - MAPLHGR (kw/ft).

14. Delete Figures 5.2 and 5.3 and renumber existing Figures 5.4 through 5.8 as 5.2 through 5.6, respectively. Add new Figure 5.7 attached.

15. Delete Section 6.1.4(a) in its entirety.

16. Delete Section 6.1.4(b) in its entirety.

17. Delete Section 6.1.4(c) in its entirety.

18. Change Section 6.1.5(b) to read as follows:

    "The emergency condenser system control initiation sensors shall be functionally tested not less frequently than once every 12 months."

19. Delete Section 6.1.6 in its entirety.

20. Delete the entire entry of "Post-Incident Spray System - Automatic Control Operation" in the table contained in Section 7.6.

21. Change the words "Reactor emergency cooling systems trip circuits" in the table in Section 7.6 to "Emergency Condenser Trip Circuits."

22. Replace Table 8.2 with the attached new Table 8.2.

23. Incorporate additional pages 11-1 through 11-20. These pages are the new format of the Specifications to be issued in the future and are identified as Section 11.0 but contain the numbering sequence of the new format, i.e., 3.1.5/4.1.5, etc. Amendment No. 10 is identified for each new page in the lower corner.

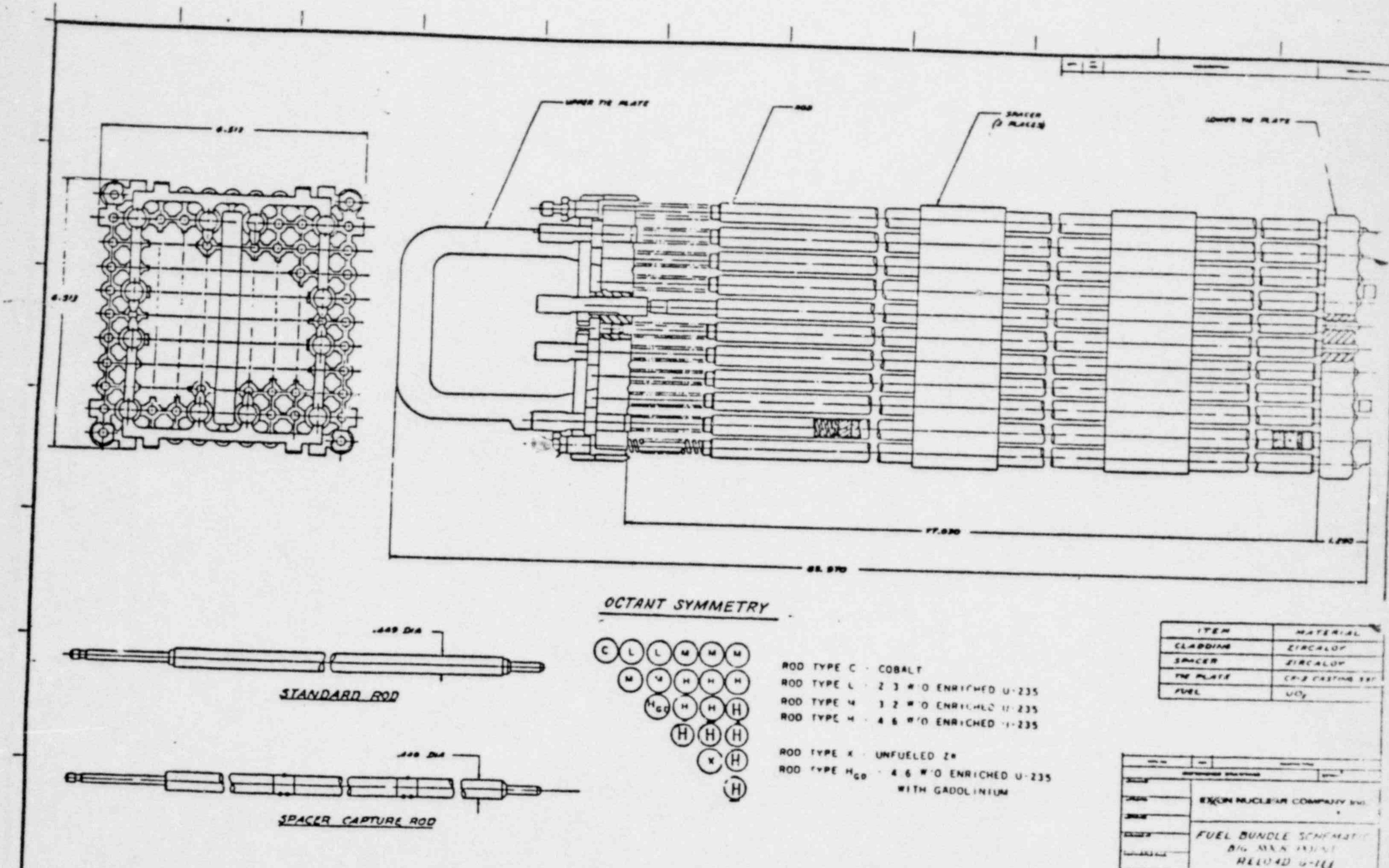
CONTENTS (Contd)

(NOTE: This table should follow Table 1 of Section 5.2.1(b))

TABLE 2

MAPLHGR (kW/Ft)

Planar Average Exposure (MWd/STU)	Modified F	E-G, F, J-2	Reload G & NFS-DA	Reload G-1U
0	-	-	6.38	6.40
200	9.5	9.4	-	-
907	-	-	-	6.86
1,814	-	-	-	6.87
2,041	-	-	6.79	-
4,536	-	-	6.76	6.90
5,000	9.9	9.7	-	-
9,072	-	-	-	7.05
9,979	-	-	6.86	-
10,000	9.9	9.7	-	-
13,608	-	-	6.97	-
14,515	-	-	-	7.25
15,000	9.8	9.6	-	-
18,144	-	-	-	7.25
19,051	-	-	6.95	-
20,000	8.7	8.6	-	-
25,000	8.7	8.3	-	-
25,401	-	-	7.05	-
27,216	-	-	-	7.28



ITEM	MATERIAL
CLADDING	ZIRCALOY
SPACER	ZIRCALOY
TIE PLATE	CF-8 CASTING SST
FUEL	UO_2

Figure 5.7

TABLE 8.2

	EEI UO_2-PuO_2	Centermelt Intermediate	Centermelt Advanced	NFS-DA
Minimum Core Burnout Ratio at Overpower	1.5*	1.5*	1.5*	1.5
Transient Minimum Burnout Ratio in Event of Loss of Recirculation From Rated Power	1.5	1.5	1.5	1.5
Maximum Heat Flux at Overpower, Btu/h-Ft^2	500,000	-	-	402,000
Maximum Steady State Heat Flux, Btu/h-Ft^2	410,000	500,000	500,000	329,000
Maximum Average Planar Linear Heat Generation Rate, Steady State, kW/Ft	**	**	**	(See Note 1)
Stability Criterion: Maximum Measured Zero-to-Peak Flux Amplitude, Percent of Average Operating Flux	20	-	-	20
Maximum Steady State Power Level, MW_t	240	-	-	240
Nominal Reactor Pressure During Steady State Power Operation, Psig	1,335	-	-	1,335
Minimum Recirculation Flow Rate, Lb/h (Except During Pump Trip Tests or Natural Circulation Tests as Outlined in Sec 8)	6×10^6	-	-	6×10^6
Maximum MWd/T of Contained Uranium for an Individual Bundle	23,500	-	-	23,500
Number of Bundles:				
Pellet UO_2	-	1	3	-
Powder UO_2	-	1	2	-

Rate-of-Change-of-Reactor Power During Power Operation:

Control rod withdrawal during power operation shall be such that the average rate-of-change-of-reactor power is less than 50 MW_t per minute when power is less than 120 MW_t, less than 20 MW_t per minute when power is between 120 and 200 MW_t, and 10 MW_t per minute when power is between 200 and 240 MW_t.

*Based upon critical heat flux correlation, APED 5286.
**No longer used in reactor.

Note 1: MAPLHGR (kW/Ft)

Planar Avg Exposure (MWd/STU)	NFS-DA
0	6.38
2041	6.79
4536	6.76
9979	6.86
13608	6.97
19051	6.95
25401	7.05

Limiting Conditions for Operation

11.3.1.4 EMERGENCY CORE COOLING SYSTEM

Applicability:

Applies to the operating status of the emergency core cooling system.

Objective:

To assure the capability of the emergency core cooling system to cool reactor fuel in the event of a Loss of Coolant Accident.

Specification:

A. The two core spray systems (original and redundant) shall be operable whenever the plant is in a power operation condition. The original core spray system shall also be operable during refueling operations.

B. The core spray recirculation system shall be operable whenever the plant is in a power operation condition.

C. The core spray recirculation heat exchanger shall not be taken out of service during power operation for periods exceeding four (4) hours. The heat exchanger shall be considered inoperable and out of service if tube bundle leakage exceeds 0.2 gpm.

Surveillance Requirement

11.4.1.4 EMERGENCY CORE COOLING SYSTEM

Applicability:

Applies to periodic testing requirements for the emergency core cooling systems.

Objective:

To verify operability of the emergency core cooling systems.

Specification:

A. Each month the following shall be performed:

Verify the operability of MO-7051, -7061, -7066, -7073 and -7074 by remote manual actuation.

Leak testing of the core spray heat exchanger.

Automatic actuation of both fire pumps.

Verify that the check valve between MO-7051 and -7061 is not stuck shut.

B. At each shutdown the following shall be performed.

Verify the operability of MO-7070 and -7071 by remote manual actuation.

11.3.1.4 EMERGENCY CORE COOLING SYSTEM (Contd)

D. Both fire pumps (electric and diesel) and the piping system to the core spray system tie-ins shall be operable whenever the plant is in a power operation condition and refueling.

E. If Specifications A, B, C, and D are not met, a normal orderly shutdown shall be initiated within 24 hours and the reactor shall be shut down as described in Section 1.2.5(a) within twelve (12) hours and shut down as described in Section 1.2.5(a) and (b) within the following 24 hours. No work shall be performed on the reactor or its connected systems when irradiated fuel is in the reactor vessel which could result in lowering the reactor water level below elevation 610'5".

F. Until such time as the effectiveness of redundant core spray nozzle has been proven, the fire water makeup system to the condenser hct well shall be operable and ready for service during power operation. If the fire water makeup system becomes inoperable and not corrected, a normal orderly shutdown shall be initiated within one (1) hour and the reactor shall be shut down as described in Section 1.2.5(a) within twelve (12) hours and shut down as described in Section 1.2.5(a) and (b) within the following 24 hours.

G. Instrument set points shall be as specified in Table 11.3.1.4(a).

11.4.1.4 EMERGENCY CORE COOLING SYSTEM (Contd)

C. At least once every six (6) months, except for periods of continuous shutdown when the following shall be performed prior to startup:

Automatic actuation of the core spray system valves with water flow manually blocked (MO-7051, -7061, -7070 and -7071).

Operability check of the Core Spray Recirculation System.

D. At each major refueling outage, the following shall be performed:

Calibration of core spray system actuation and pressure and flow instrumentation.

Verify that the two core spray system containment isolation check valves are not stuck shut.

Operability of the check valves between MO-7051 and MO-7061 and MO-7070 and MO-7071.

Calibration of fire system basket strainer differential pressure switches.

Operability check of the core spray recirculation system.

E. Instruments shall be checked, tested and calibrated at least as frequently as listed in Table 11.4.1.4(a).

TABLES 11.3.1.4a AND 11.4.1.4a

Instrumentation That Initiates Cores Spray

Parameter	11.3.1.4a Limiting Conditions for Operation: Trip System Logic	Limiting Set Point	Conditions for Operability	11.4.1.4a Surveillance Requirement: Instrument Trip Test Including Valve Actuation	Instrument Calibration
Open Core Spray Valves					
Low Reactor Water Level (b)	One of Two for Each of Two Valves in Series	≥610'5" Elev (≥2'8" Above Core)	Power Operation and Refueling Operations (a)	Once Every Six Months of Operation Other Than Cold Shutdown	Each Major Refueling
Steam Drum Pressure Low (b)	One of Two for Each of Two Valves in Series	>200 Psig	Power Operation and Refueling Operations (a)	Once Every Six Months of Operation Other Than Cold Shutdown	Each Major Refueling

Notes for Tables 11.3.1.4a and 11.4.1.4a

(a) Initiation of valve operation requires both low reactor water level coincident with low steam drum pressure.

(b) The primary core spray system shall be available for use during refueling operations and the backup system shall be closed and operation of the backup core spray valves shall be blocked or otherwise defeated while the piping section from the valves to the reactor head is dismantled.

Bases:

The core spray system consists of two automatically actuated independent double capacity piping headers capable of cooling reactor fuel for a range of Loss of Coolant Accidents. Either system by itself is capable of providing adequate cooling for postulated large breaks in all locations. When adequate depressurization rates are achieved in the postulated small-break situation, either core spray system provides adequate cooling. For the largest possible pipe break, a flow rate of approximately 400 gpm is required after about 20 seconds.

Each core spray system has 100% cooling capacity from each spray header and each pump set. Thus, specifying both systems to be fully operational will assure to a high degree core cooling if the core spray system is required. In addition, the original core spray is required to be operable during refueling operations to provide fuel cooling in the unlikely event of an inadvertent draining of the reactor vessel.

The core spray systems receive their water supply from the plant fire system. The plant fire system supply is from Lake Michigan via two redundant 1,000 gpm fire pumps, one electric and one diesel driven. These pumps start automatically on decaying fire system pressure.

The core spray recirculation system is provided to prevent excessive water buildup in the containment sphere and to provide for long-term, post-accident cooling. The system consists of two pumps (400 gpm each) and a heat exchanger. The pumps take a suction from the lower levels of containment and discharge to the core spray headers. The system is actuated manually when the water level in the containment rises to elevation 587 feet. The 587-foot elevation will be achieved between 6 to 24 hours operation of one core spray and one containment spray system.

Bases: (Contd)

A test tank and appropriate valving is provided in the core spray recirculation system so the pump suction conditions and the flow characteristics of the system can be periodically tested.

One core spray recirculation pump has adequate capacity to provide fuel cooling at anytime following a Loss of Coolant Accident. Continuous containment spray operation is not required during the post-accident recirculation phase if only one recirculation pump is available.

The operable status of the various systems and components is to be demonstrated by periodic tests. Some of these tests will be performed while the reactor is operating in the power range. If a component is found to be inoperable, it will be possible in most cases to effect repairs and restore the system to full operability within a relatively short time. For a single component to be inoperable does not negate the ability of the system to perform its function, but it reduces the redundancy provided in the reactor design and thereby limits the ability to tolerate additional equipment failures. If it develops that (a) the inoperable component is not repaired within the specified allowable time period; or (b) a second component in the same or related system is found to be inoperable, the reactor will initially be removed from service which will provide for a reduction of the decay heat from the fuel and consequential reduction of cooling requirements after a postulated Loss of Coolant Accident. If the malfunction cannot be rapidly corrected, the reactor will be cooled to the shutdown condition using normal cooldown procedures. In this condition, release of fission products or damage of the fuel elements is not considered possible.

The plant operating procedures will require immediate action to effect repairs of an inoperable component and, therefore, in most cases, repairs will be completed in less than the specified allowable repair times. The limiting times to repair are intended to: (1) Assure that operability of the component will be restored promptly and yet, (2) allow sufficient time to effect repairs using safe and proper procedures.

The leakage rate limit for the core spray recirculation system heat exchanger has been established to assure detection of any degradation of the integrity of the heat exchanger.

Bases: (Contd)

As a result of an evaluation of the effect testing frequency on emergency core cooling system reliability[9] and because of a lack of test data to prove the effectiveness of the redundant (nozzle) core spray system spray distribution[10] the surveillance requirements for the original core spray system have been increased. In addition, time periods allowed for operation with the original (ring) core spray system out of service have been reduced significantly. Further changes in surveillance and operability requirements will be requested prior the refueling outage presently scheduled for Spring 1977 based on modifications to make the core spray systems more testable and following proof of nozzle spray effectiveness.

The fire water makeup system to the condenser hot well was provided as a temporary means of reducing peak fuel clad temperature under postulated small and intermediate sized pipe breaks until the Reactor Depressurization System could be completed. It is still required until nozzle spray distribution patterns are demonstrated.

References:

1. Consumers Power Company letter to Directorate of Licensing, USAEC, dated May 18, 1972.

2. Technical Specifications Change No 26 dated July 27, 1971.

3. FHSR, Section 12.

4. FHSR, Section 3.

5. FHSR, Section 5.

6. Consumers Power Company letter to Directorate of Licensing, USAEC, dated September 22, 1972.

7. FHSR, Section 13.

8. "Big Rock Point Plant Hydrological Survey," Great Lakes Research Division, Special Report No 9, Ayer, J. C., et al, Nov 1961.

9. Consumers Power Company letter to the Secretary of the Commission, USNRC dated March 26, 1976.

10. Comments by the Director, Nuclear Reactor Regulation Relating to the Request for Exemption of the Big Rock Point Nuclear Power Plant From the Requirements of 10 CFR 50.46 dated April 19, 1976.

(NOTE: This is the new format of the Specifications to be issued in the future. Therefore, the numbering system may conflict with existing sections. Both are still applicable.)

Limiting Conditions for Operation

3.1.5 REACTOR DEPRESSURIZATION SYSTEM

Applicability:

Applies to the operating status of the Reactor Depressurization System (RDS).

Objective:

To assure the operability of the RDS and when working in conjunction with the emergency core cooling system to allow cooling of the reactor fuel in the event of a Loss of Coolant Accident.

Specification:

A. The RDS shall be operable at all power levels and when the reactor is critical with the head on or when in hot shutdown conditions.

B. The limiting conditions for operation of the instrumentation and actuating circuitry which initiates and controls the RDS are given in Table 3.5.2.h.

Surveillance Requirement

4.1.5 REACTOR DEPRESSURIZATION SYSTEM

Applicability:

Applies to periodic testing requirements for the RDS.

Objective:

To verify operability of the RDS.

Specification:

A. The isolation valves shall be test-operated at least once every three months per Section IWV-3410 Summer 1973 Addenda of the ASME B&PV Code Section XI.

B. The depressurizing valves shall be test-operated during each cold shutdown; however, in the case of frequent cold shutdowns, these valves need not be exercised more often than once every three months per Section IWV-3410 Summer 1973 Addenda of the ASME B&PV Code Section XI.

C. The instrumentation shall be functionally tested, calibrated and checked as indicated in Table 4.5.2.h.

3.1.5 REACTOR DEPRESSURIZATION SYSTEMS (Contd)

Action:

1. Should one depressurizing valve or isolation valve become inoperable in the closed position, the reactor may remain in operation for a period not to exceed seven (7) days. The remaining valves and actuating circuitry shall be demonstrated to be operable within 4 hours and at least once each 72 hours until the system is restored to operable status.

2 Should one isolation valve or depressurizing valve become inoperable in the open position, during power operation, the plant will be brought to the cold shutdown condition within 12 hours.

3. Only one RDS valve train, one input channel, one output channel and one UPS power supply may be out of service at any one time. If these components are not returned to operable status within seven (7) days, a normal orderly shutdown shall be initiated within one (1) hour and the reactor shall be shutdown as described in Section 1.2.5(a) within twelve (12) hours and shutdown as described in Section 1.2.5(a) and (b) within the following 24 hours.

4.1.5 REACTOR DEPRESSURIZATION SYSTEM (Contd)

D. System Logic shall also be functionally tested as indicated in Table 4.5.2.h.

E. Should one input or output channel fail, the remaining three channels shall be tested within 4 hours and at least once each 72 hours until the system is restored to normal.

F. The UPS battery surveillance is as described in Section 11.4.5.3.

G. The RDS containment penetration assemblies seal pressure shall be examined at six-month intervals.

Tables 3.5.2.h and 4.5.2.h

Instrumentation That Initiates RDS Operation

Parameter	Minimum Operable Channels	Limiting Set Point	Conditions for Operability	Instrument Trip Test	Instrument Calibration	Protective Channel Trip
3.5.2.H - Limiting Conditions for Operation				4.5.2.H - Surveillance Requirement		
Low Steam Drum Level	3	Above or Equal to 25" Below Center Line	At Power Levels Whenever the Reactor Is Critical With the Head On or when in Hot Shutdown	Monthly	Each Major Refueling	-
Fire Pump(s) Discharge Pressure	3	≥ 100 Psig	Ditto	Monthly	Each Major Refueling	-
Low Reactor Water Level	3	≥ 2'8" Above Top of Active Fuel	"	Monthly	Each Major Refueling	-
120-Second Time Delay	3	≤ 120 Seconds Following Low Steam Drum Level Signal	"	Monthly	Each Major Refueling	-
Input Channels A Through D	3	-	"	Monthly	-	-
Output Channels I Through IV	3	-	"	-	-	Monthly
*Fire Pump Start	1	-	"	Monthly	-	Monthly

*Reference Specifications 3.1.4 and 4.1.4 for Bases.

Limiting Conditions for Operation	Surveillance Requirement
3.1.5 REACTOR DEPRESSURIZATION SYSTEM (Contd)	4.1.5 REACTOR DEPRESSURIZATION SYSTEM (Contd)

Bases:

The RDS provides for both manual and automatic depressurization of the primary system to allow injection of the core spray following a small-to-intermediate size break in the primary system. This will allow core cooling with the objective of preventing excessive fuel clad temperatures. The design of this system is based on the specified initiation set points described in Table 3.5.2.h. Transient analyses reported in Section 6 of the RDS Description, Operation and Performance Analysis submitted August 15, 1974 to the Directorate of Licensing USAEC, to demonstrate that these conditions result in adequate safety margins for both the fuel and the system pressure. Performance analysis of the RDS is considered only with respect to its depressurizing effect in conjunction with core spray. Therefore, no credit is taken for steam cooling of the core which provides further conservatism to the emergency core cooling system.

These specifications ensure the operability of the RDS under all conditions for which the automatic or manual depressurization of the system is an essential response to the transient described above.

Bases: (Contd)

One RDS valve can remain out of service in the closed position for seven days because of redundancy, provided the remaining RDS valves are test-operated immediately as described in Specification 4.1.5.C. When conditions for the actuation on the depressurizing system are reached, all the valves in the four blowdown paths are opened. Each blowdown path is designed to pass 144 lb/second of steam at 1350 psig which is a third of the required total flow rate. Therefore, failure of one flow path to open upon actuation does not preclude achieving the required rate of depressurization.

In addition to reactor protection instrumentation, which initiates a reactor scram, protective instrumentation has been provided for the RDS which initiates action to mitigate the consequences of the Loss of Coolant Accident. This set of specifications provides the limiting conditions of operation for the RDS. The objectives of the specifications are (i) to assure the effectiveness of the protective instrumentation when required even during periods when portions of such systems are out of service for maintenance and (ii) to prescribe the trip settings required to assure adequate performance. To conduct the required input channel maintenance or functional tests and calibrations, any one channel may be bypassed. If the input channel is not bypassed when functional tests and calibrations are performed, actual trip signals supersede test and calibration conditions.

The minimum functional testing frequency used in this specification is based on a frequency that has proven acceptable and conforms to that of the existing reactor protection system.

Four plant variables are monitored and used as inputs to the actuation system. These are (1) steam drum water level, (2) reactor water level, (3) motor-driven fire pump discharge pressure and (4) diesel engine driven fire pump discharge pressure. These variables are jointly processed by the four independent actuation system input channels which are physically and electrically isolated from one another. A failure in one channel cannot propagate into another channel. Each of the four plant variables is monitored by four separate sensors. One sensor in each of the four variables is associated with each of the four input channels. The actuation of the RDS is enabled when two of the four input channels are in a tripped state.

The input channel is in a tripped state upon coincidence and subsequent processing of the following inputs: (1) Low steam drum level (delayed for two minutes), (2) high fire pump discharge pressure (either diesel- or motor-driven) and (3) low reactor water level. A low steam drum level signal is generated when the steam drum level sensor associated with the input channel indicates a level of 25" below steam drum center line.

This low steam drum level signal initiates a two-minute delay which allows a containment evacuation interval prior to system blowdown and also permits the incorporation of operator input to the system initiation logic specified in the design basis (Reference Section 3.3.D of the August 15, 1974 RDS Description, Operation and

Bases: (Contd)

Performance Analysis). For the latter, the operator is provided with manual timer reset capability for each of the four input channels at the control panel. The low steam drum level signal is also used to generate a fire pump start signal. Verification of a fire pump start and thus verification that a source of core spray water is available at the core spray valves is obtained when the pressure switch associated with the input channel at either fire pump discharge has tripped, corresponding to a pressure equal to or exceeding 100 psig. This variable is used as an enabling input to the actuation system to prevent depressurizing the reactor coolant system when the source of coolant required to cool the core is not available. A low reactor water level signal is generated when the input channel reactor water level sensor indicates a level $\geq$ 2'8" above the top of active fuel. Low reactor water level is confirmation of the LOCA and with the other two inputs present (time delayed low drum level and core spray water availability) causes the automatic trip of the input channel. These trip level settings were chosen to be low enough to prevent spurious actuation but high enough to initiate RDS operation so that post-accident cooling can be accomplished.

Upon failure of an uninterruptible power supply (UPS) or a channel power supply, the affected channel fault condition is alarmed as "channel 'X' unavailable." Power failures associated with input channels cause the coincidence trip conditions for the input channels to change from 2-out-of-4 to 2-out-of-3. The output channel actuation coincidence reverts to 3-of-3 upon failure of an output channel power supply.

Input channel bypass capability is provided to permit bypassing any one input channel at a time. The bypass feature is used to bypass a channel when the channel has failed to the "trip" state and/or when channel maintenance is required. Bypassing of an input channel in the "trip" state or for maintenance causes the coincidence trip condition of the input channel to be changed from 1-out-of-3 or 2-out-of-4, respectively, to 2-out-of-3. The input channel bypassed condition is alarmed as "channel 'X' unavailable" and "bypassed."

Should an output channel require maintenance or should a single fault cause an output channel subchannel trip (two independent subchannels operate in 2 of 2 coincidence), the output channel actuation capability can be disabled by removing the associated 125 V DC supply. The 125 V DC supply to an output channel is disabled via a circuit breaker in its respective UPS. The disabling of an output channel is alarmed as "channel 'X' unavailable."

Since 3-out-of-4 output channels are required to assure design requirements are met (one output channel operates one depressurizing valve and one isolation valve), the failure of one output channel will not preclude achieving the required rate of depressurization. This redundancy also enables maintenance to be performed on one output channel while the plant is in operation.

Bases: (Contd)

Once the RDS actuation system output channels are enabled (at least two input channels are in a tripped state or a manual trip is initiated) and tripped, they remain in that condition until they are manually reset. This reset can be accomplished only after the initiating signals (ie, input channel trips or manual trip) have been restored to levels at which RDS operation is not required.

Separate, independent and one-hour sources of electrical power are provided, through four divisions, to accomplish the detection of the LOCA and the completion of the depressurization. Each of the divisions (1, 2, 3 and 4) is supplied with electrical power from one of four independent uninterruptible power supplies (UPS) consisting of a battery charger, a battery and an inverter.

Each UPS has output of 120 V AC, 60 Hz and 125 V DC. Divisions 1 and 2 normally receive power from the existing 480 V AC Bus 1A. Divisions 3 and 4 are supplied by 480 V AC Bus 2A. Normal station power to Busses 1A and 2A can be provided by one of three sources: (1) The station turbine generator, (2) the 138 kV transmission line or (3) the 46 kV transmission line. Should none of these sources be available, provision is included for supplying input power from the 480 V AC Bus 2B which is tied to the emergency diesel. If all 480 V AC power is lost, the UPS is capable of sustaining its output for one hour.

Since only 3-out-of-4 blowdown paths are required to assure adequate depressurization, the single system failure of one UPS division will not preclude achieving the required rate of depressurization. This redundancy also enables maintenance to be performed on the UPS while the plant is in operation.

Four new containment penetration assemblies are used in transmitting electrical power, control and instrumentation signals between equipment located inside the containment building and facilities located external to the containment building. These electrical penetrations are welded into spare containment penetration sleeves. The penetration assemblies are designed in accordance with IEEE 317 and are seismically and environmentally qualified to the RDS design conditions.

The pressure retaining portion of the assemblies is designed and fabricated to the requirements of Subsection NE, Class MC vessels, of Section III of the ASME Code. The penetration assemblies include a single aperture seal and a double electrical conductor seal and are designed to operate with the internal cavity pressurized with nitrogen at approximately 27 psig. The relatively maintenance-free seal assemblies dictate a minimum inspection frequency of twice annually.

Limiting Conditions for Operation	Surveillance Requirement

11.3.3.4 CONTAINMENT SPRAY SYSTEM

Applicability:

Applies to the operating status of the containment spray system.

Objective:

To assure the capability of the containment spray system to reduce containment pressure in the event of a Loss of Coolant Accident.

Specification:

A. During power operation each of the two containment spray systems shall be operable.

B. If Specification A is not met, a normal orderly shutdown shall be initiated within 24 hours and the reactor shall be shut down as described in Section 1.2.5(a) within 12 hours and shutdown as described in Section 1.2.5(a) & (b) within the following 24 hours.

C. Operability of the fire water supply and recirculation systems is governed by Specification 11.3.1.4.

11.4.3.4 CONTAINMENT SPRAY SYSTEM

Applicability:

Applies to the testing of the containment spray system.

Objective:

To verify the operability of the containment spray system.

Specification:

A. Once each operating cycle, the following shall be performed:

   1. Automatic actuation of the containment spray valve MO-7064 (with water flow manually blocked).

   2. Calibration of flow instrumentation.

B. At least once every six (6) months, except for periods of continuous shutdown when the following shall be performed prior to startup:

   Verify operability of power-operated valves required for proper system actuation.

C. Surveillance of fire water supply and recirculation systems is governed by Specification 4.1.4.

D. Instrument channels shall be tested and calibrated as listed in Table 11.4.3.4(a)

TABLE 11.4.3.4

Instrumentation That Initiates Enclosure Spray

Parameter	11.3.3.4 Limiting Conditions for Operation Trip System Logic	Set Point	Conditions for Operability	11.4.3.4 Surveillance Requirement Instrument Trip Test Including Valve Actuation	Instrument Calibration
Enclosure High Pressure	1 of 2	≤2.2 Psig (a)	Power Operation and Refueling Operation	Once Every Six Months of Operations Other Than Shutdown	Each Major Refueling
Time Delay (b)	1 of 1	≥13 Min, ≤15 Min (a)	Power Operation and Refueling Operation	Once Every Six Months of Operations Other Than Shutdown	Each Major Refueling

(a) Primary enclosure spray setting.

(b) The time delay device requires power to perform the tripping function. This supply is provided by the valve control circuit.

Bases:

The containment spray systems are provided to reduce pressure in the containment following a Loss of Coolant Accident. They are initially supplied from the fire water system and later by the core spray recirculation system. They are not required to be in service at reactor coolant temperatures of 212°F or below because the resultant Loss of Coolant Accident pressure is not sufficient to pressurize the containment.

Operation of only one system is sufficient to provide the required containment spray flow. The specified flow of approximately 400 gpm is sufficient to remove post-accident core energy releases including a substantial chemical reaction involving hydrogen generation to below design values.

The operable status of these systems and components is demonstrated by periodic tests. 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. If a single system becomes inoperable, a redundant system has been provided with the ability to perform the spray function, but it reduces the redundancy provided by plant design and limits the ability to tolerate additional equipment failures.

Initiation of the containment spray system assures that containment design pressure will not be exceeded due to hydrogen generation assuming the core spray systems do not function. It has been conservatively calculated that the energy release following a complete core meltdown (assuming no containment spray systems or core spray systems operate) would bring the containment pressure to approximately the design value (27 psig) about 15 minutes after the postulated accident had occurred. Subsequent LOCA analysis system modifications and regulations have limited H_2 generation such that it is no longer significant and calculations show that containment sprays are not required to prevent containment design pressures from being exceeded. Thus, the automatic actuation time of the primary containment spray system has been established at 15 minutes so as to allow the operator adequate time to evaluate and block actuation, if system operation is not required.

References:

1. FHSR, Section 3.

2. Additional information in support of Proposed Technical Specification Change No 8 dated March 17, 1966.

3. Safety Evaluation by the Research and Power Reactor Safety Branch, Division of Reactor Licensing, Consumers Power Company, Proposed Change No 8 dated April 14, 1966.

Limiting Conditions for Operation	Surveillance Requirement

11.3.5.3 EMERGENCY POWER SOURCES

Applicability:

Applies to the operational status of the emergency power sources.

Objective:

To assure the capability of the emergency power sources to provide power required for emergency equipment in the event of a Loss of Coolant Accident.

Specification:

A. For all reactor operating conditions except cold shutdown, there shall normally be available one 138 kV line, one 46 kV line, one diesel generator system, one station battery system, and four RDS uninterruptible power supplies including batteries, except as specified below:

   1. Refueling operations and related testing may be conducted with the 138 kV line de-energized.

   2. The 46 kV line or the diesel generator may be out of service for repair for periods up to three (3) days during reactor operation and for extended periods during refueling or shutdown operations

11.4.5.3 EMERGENCY POWER SOURCES

Applicability:

Applies to the periodic testing requirements for the emergency power sources.

Objective:

To assure the operability of the emergency power sources to provide emergency power in the event of a Loss of Coolant Accident.

Specification:

A. The emergency power system surveillance will be performed as indicated below. In addition, components on which maintenance has been performed will be tested.

   1. During each operating cycle -

      (a) Test of automatic initiation sensors and load test the emergency diesel to 180-200 kW generator output for at least 20 minutes.

      (b) Test and calibrate the following instruments and controls associated with diesel generator:

         (1) Fuel oil level.

         (2) Oil Pressure tripping.

         (3) Water temperature tripping.

11.3.5.3 EMERGENCY POWER SOURCES (Contd)

3. The diesel generator fuel supply shall be adequate for one-day operation.

4. If Specifications A.2 or A.3 are not met, a normal orderly shutdown shall be initiated within one (1) hour and the reactor shall be shut down as described in Section 1.2.5(a) within twelve (12) hours and shut down as described in Section 1.2.5(a) and (b) within the following 24 hours. During refueling operations cease all changes which could affect reactivity.

5. The station battery system shall be operable under all conditions except during cold shutdown. If the station battery is inoperable, no actions shall be taken which result in a reactivity addition, except cooldown, or which might result in the primary coolant system being drained.

6. If Specification A.5 is not met a normal orderly shutdown of the reactor shall be initiated within one (1) hour and the reactor shall be shut down as described in Section 1.2.5(a) within twelve (12) hours and shut down as described in Section 1.2.5(a) and (b) within the following 24 hours.

7. One RDS uninterruptible power supply including battery may be out of service as described in Section 3.1.5 Action 3.

11.4.5.3 EMERGENCY POWER SOURCES (Contd)

(4) Overspeed tripping.

(5) Battery undervoltage alarm.

(c) Verify the automatic transfer of station power from the 138 kV line to the 46 kV line.

(d) Verify the automatic transfer of power sources for the 1Y and 2Y instrument and control panels.

(e) Verify the cells cell plates, and battery racks show no visual indication of physical damage or abnormal deterioration.

(f) Verify the cell-to-cell and terminal connections are clean, tight, free of corrosion and coated with anti-corrosion material.

(g) Verify that the battery charger will supply at least 30 amperes at a minimum of 135 volts for at least 4 hours. See note below.

(h) Verify that the battery capacity is adequate to supply and maintain in OPERABLE status all of the actual emergency loads for the design time interval when the battery is subjected to a battery service test. The design time interval for the RDS batteries is one hour and eight hours for the station battery. See note below.

11.4.5.3 EMERGENCY POWER SOURCES (Contd)

2. Monthly -

(a) Test start diesel generator and operate at least the fire pump as a load to 480 V Bus 2B for at least 20 minutes.

(b) Verify that the cell voltage is $\geq$ 2.0 volts and specific gravity is $\geq$ 1.2 of each cell of the station battery and the RDS batteries.

(c) Test operate the rod position motor generator set.

3. Weekly - The electrolyte level of each pilot cell is between the minimum and maximum level indication marks.

The pilot cell specific gravity, corrected to (77)°F, is $\geq$ 1.2.

The pilot cell voltage is $\geq$ 2.0 volts.

The overall battery voltage is $\geq$ 125 volts.

Test start the diesel generator and run for warn-up period.

4. Sixty Months - At least once per 60 months during shutdown verify that the battery capacity is at least 80% of the manufacturers rating when subjected to a performance discharge test. This performance discharge test shall be performed subsequent to the satisfactory completion of the required battery service test. See Note below.

Note: These surveillance items shall become effective prior to startup following the 1977 refueling outage.

Bases:

Normal station power can be provided by the station turbine generator, the 138 kV transmission line or the
46 kV line. These sources are adequate to provide emergency a-c power. When none of these sources is avail-
able, a single emergency diesel generator rated at 200 kW starts automatically to provide emergency a-c power
to 480 V Bus 2B. The weekly starting test is based on Manufacturer's Bulletin 33743-1 for relubrication pro-
tection of moving parts. Diesel generator initiation and output circuit breaker closure is accomplished by two
voltage sensors, one to detect loss of normal power on Bus 1A and the other to provide assurance of generator
output prior to automatic closure of the generator output breaker. Additional breaker interlocks are provided
to assure that the normal Buses 1A and 2A are isolated prior to closing the generator output breaker. This
prevents overloading of the generator at the switching period.

The diesel fuel oil tank is sized for two-day full load operation. One-day supply is considered adequate to
provide fuel makeup.

A single station battery supplies power for normal station services and is sized for emergency uses including valves and controls for Loss of Coolant Accidents. The battery can be charged from the emergency diesel generator output if normal station power sources are not available.

The primary core spray valves and the primary containment spray valve are operated and controlled by power from the station battery. The backup core spray valves and backup containment spray valves are operated by power from normal station power sources or the emergency diesel generator.

RDS uninterruptible power supplies (UPS) A, B, C, and D, each consisting of a battery, battery charger and an inverter, supply each division (except division 5) with electrical power. Each UPS has outputs of 120 VAC, 60 Hz, and 125 VDC. One of these batteries supplies control power for the emergency diesel generator. Divisions 1 and 2 and 3 and 4 normally receive power from 480 VAC busses 1A and 2A, respectively. In the event of loss of power to either or both busses, provision is included for supplying input power from 480 VAC bus 2B which is tied to the emergency diesel generator. If all 480 VAC power is lost, the RDS UPS is capable of sustaining its outputs for one hour. The station battery has adequate capacity to carry normal loads plus an assumed failure (locked rotor current) of the DC lube oil pump for 54 minutes without the battery charger and still provide sufficient power for equipment required to operate during a LOCA. If steps are taken to reduce nonessential loads during a loss of off-site power (such as part of the emergency lights) additional time (up to five hours) can be gained from the time of the loss of the charger until the battery would no longer have sufficient power for equipment required to operate during a LOCA. The station battery and the four (4) RDS batteries will be considered operable if they are essentially fully charged and the battery charger is in service. Additionally prior to the startup following the 1977 refueling outage, successful completion of service testing and performance discharge testing within each operating cycle and each sixty months, respectively, will further establish battery reliability.