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. 15
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 December 17, 1976 (as supplemented by letters dated February 9 and August 17, 1977) and April 15, 1977 (as supplemented by letters dated April 21, August 12 and 24, and September 26, 1977) comply with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The request for exemption from ECCS failure criterion of 10 CFR 50.46, Appendix K, Paragraph I.D.1 dated September 15, 1977 (as supplemented by letter dated October 12, 1977) is authorized by law and will not endanger life or property or the common defense and security and is otherwise in the public interest.

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

   D. 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;

   E. 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

F. 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 paragraphs 2.C(2) and (3) of Facility License No. DPR-6 are hereby amended to read as follows:

(2) Technical Specifications

The Technical Specifications contained in Appendix A, as revised through Amendment No. 15, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

(3) Exemption from 10 CFR §50.46, Appendix K, Paragraph I.D.1

Pursuant to 10 CFR §50.12 the licensee is granted an exemption from the ECCS failure criterion of 10 CFR §50.46, Appendix K, Paragraph I.D.1 as applied to a Loss-of-Coolant Accident followed by concurrent single failure in the redundant core spray system for the 1978 operating fuel cycle.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Victor Stello, Jr., Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 17, 1977

ATTACHMENT TO LICENSE AMENDMENT NO. 15

FACILITY OPERATING LICENSE NO. DPR-6

DOCKET NO. 50-155

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

1. Add the following column to Table 5.1 for Reload G-3:

11 x 11	3
0.577	Zr-2
113	0.034
0	0.449
4	91.5
1	70
3	Helium $\geq$95%

2. Add new Figure 5-8 attached.

3. Replace the following revised tables in Section 5.2.1(b):

   Table 1 and Table 2

4. Add new Figure 1 and Figure 2 following Table 2.

5. Replace Table 8.2 with the revised Table 8.2.

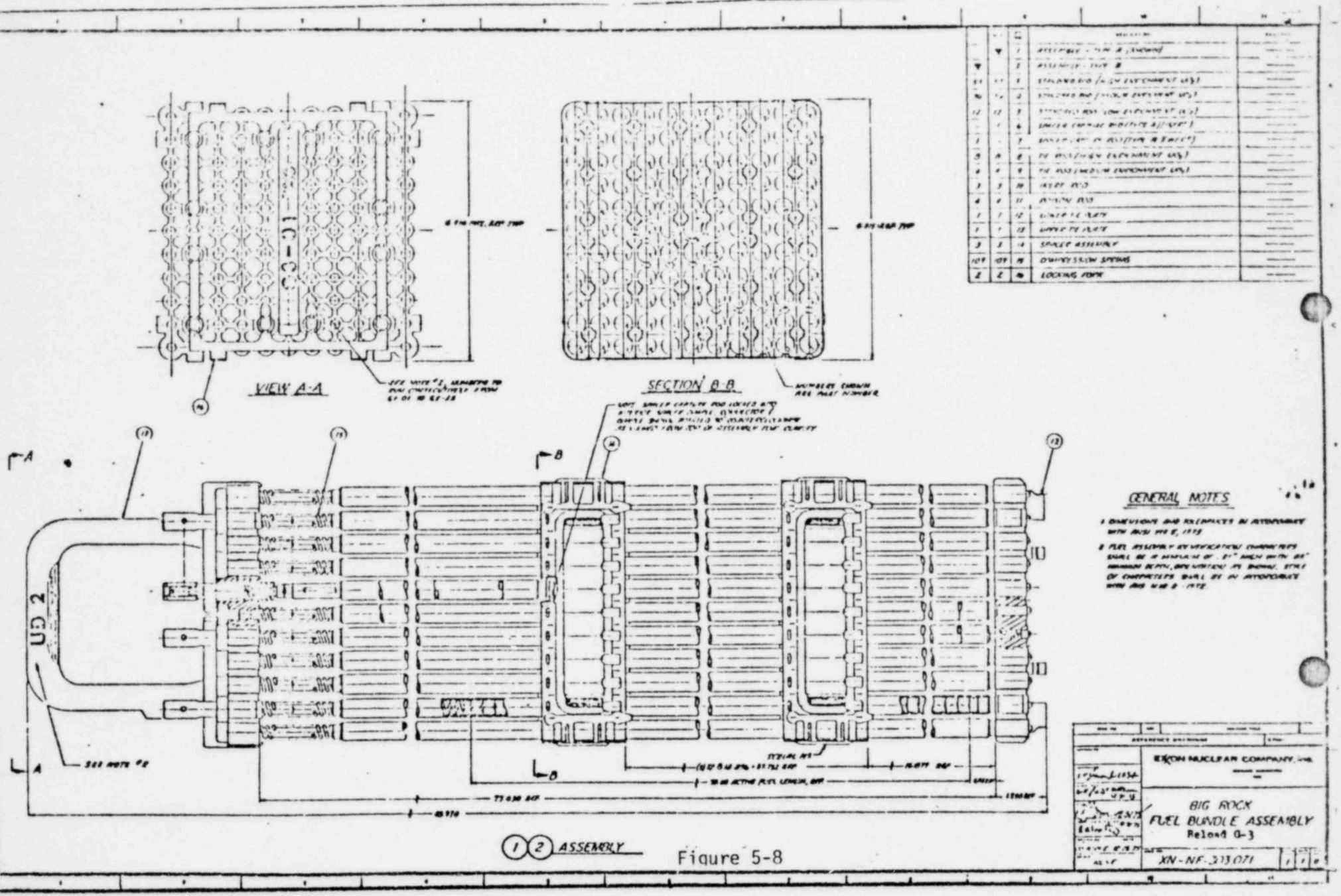
6. Replace Section 11.3.1.4 with the following revised pages:

   11-1 through 11-5

7. Delete Page 11-6 of Section 11.3.1.4.

8. In Paragraph 4.1.2(b) revise the first sentence to read: "A minimum of one reactor recirculating loop or its equivalent shall be used during all reactor power operations."

9. Replace pages 11-14 and 11-16 of Section 11.3.3.4 with attached revised pages.



(1)(2) ASSEMBLY

Figure 5-8

TABLE 1

	Reloads: Modified F & J-2	Reload G	Reload G-1U	Reload G-3
Minimum Critical Heat Flux Ratio at Normal Operating Conditions*	3.00	3.00	3.00	3.00
Minimum Bundle Dry Out Time**	Figure 1	Figure 2	Figure 2	Figure 2
Maximum Heat Flux at Overpower, Btu/h-ft^2	500,000	395,000	407,000	392,900
Maximum Steady State Heat Flux, Btu/h-ft^2	410,000	324,000	333,600	322,100
Maximum Average Planar Linear Heat Generation Rate, Steady State, kW/ft***	Table 2	Table 2	Table 2	Table 2
Stability Criterion: Maximum Measured zero-to-Peak Flux Amplitude, Percent of Average Operating Flux	20	20	20	20
Maximum Steady State Power Level, MW_t	240	240	240	240
Maximum Value of Average Core Power Density @240 MW_t, kW/L	46	46	46	46
Nominal Reactor Pressure During Steady State Power Operation, psig	1335	1335	1335	1335
Minimum Recirculation Flow Rate, Lb/h	6×10^6	6×10^6	6×10^6	6×10^6

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.

* The bundle Minimum Critical Heat Flux Ratio (MCHFR), based on the Exxon Nuclear Corporation Senthesized Hench-Levy Correlation, must be above this value.

** The actual dryout time for each bundle (based on the General Electric Dry Out Correlation for Nonjet Pump Boiling Water Reactors, NEDE-20566) should be above the dryout time shown in Figure 1 or 2, as appropriate.

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

TABLE 2

MAPLHGR (kW/Ft) LIMITS

Planar Average Exposure (MWd/STM)	Reload Modified F	Reload F,J-2	Reload G	Reload G-1U	Reload G-3
0	-	-	6.453	6.491	6.554
200	9.5	9.4	-	-	-
214	-	-	6.750	6.758	-
216	-	-	-	-	6.807
437	-	-	6.887	6.888	-
443	-	-	-	-	6.973
884	-	-	-	6.960	-
885	-	-	6.978	-	-
893	-	-	-	-	7.033
1,758	-	-	6.929	-	-
1,769	-	-	-	6.970	-
1,773	-	-	-	-	6.984
3,494	-	-	6.885	-	-
3,509	-	-	-	-	6.913
3,545	-	-	-	6.983	-
5,000	9.9	9.7	-	-	-
6,939	-	-	6.838	-	-
6,970	-	-	-	-	6.865
7,085	-	-	-	6.978	-
10,000	9.9	9.7	-	-	-
10,422	-	-	6.847	-	-
10,482	-	-	-	-	6.882
10,690	-	-	-	7.019	-
13,938	-	-	6.867	-	-
14,019	-	-	-	-	6.904
14,355	-	-	-	7.069	-
15,000	9.8	9.6	-	-	-
20,000	8.7	8.6	-	-	-
21,022	-	-	6.905	-	-
21,194	-	-	-	-	6.958
21,843	-	-	-	7.171	-
25,000	8.4	8.3	-	-	-
27,778	-	-	6.843	-	-
28,035	-	-	-	-	6.903
29,084	-	-	-	7.161	-
34,013	-	-	6.703	-	-
35,147	-	-	-	-	6.923
35,322	-	-	-	6.958	-

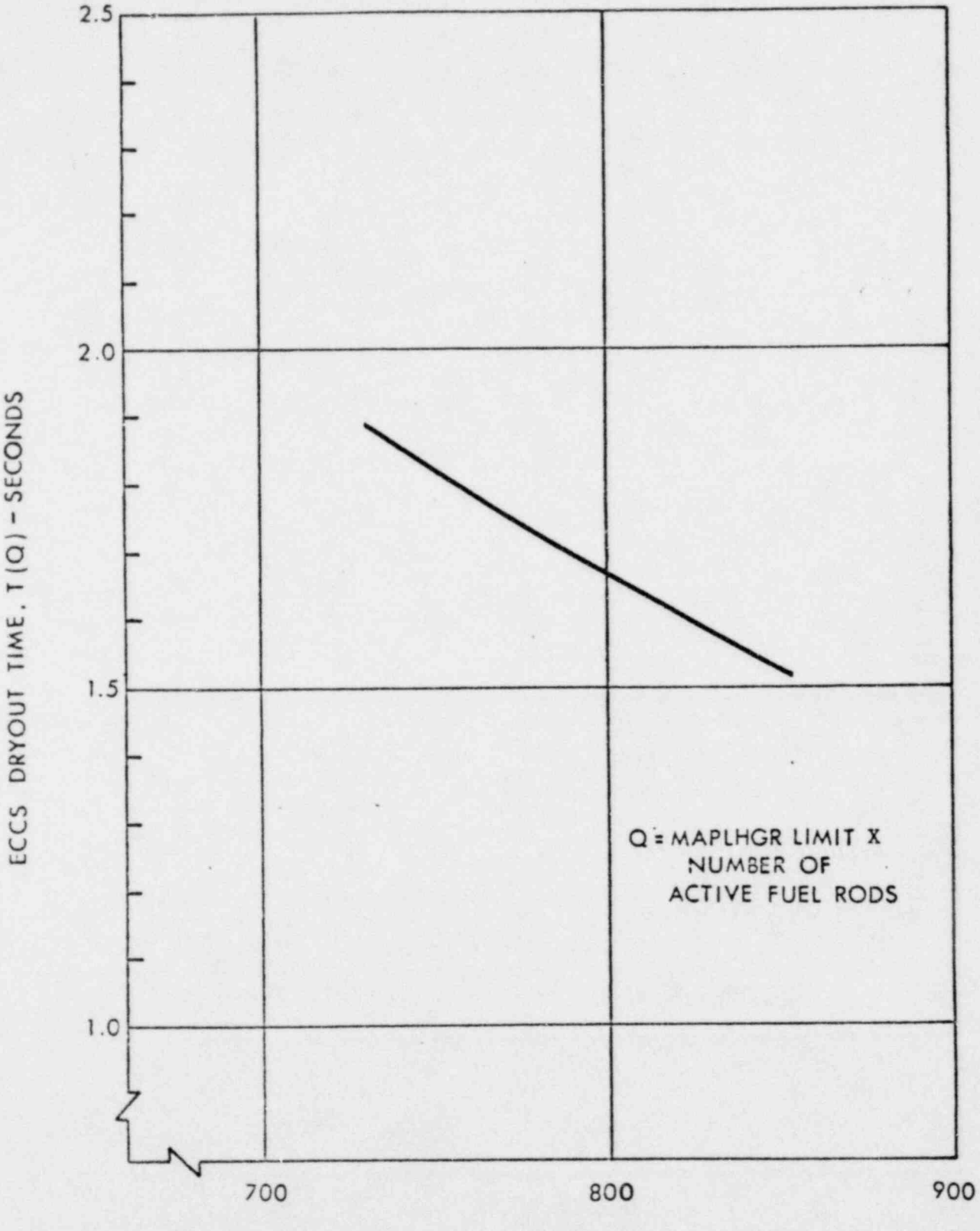


Figure 1

FUNCTION T(Q) FOR ENC 11X11 FUEL POWER VOID RELATION

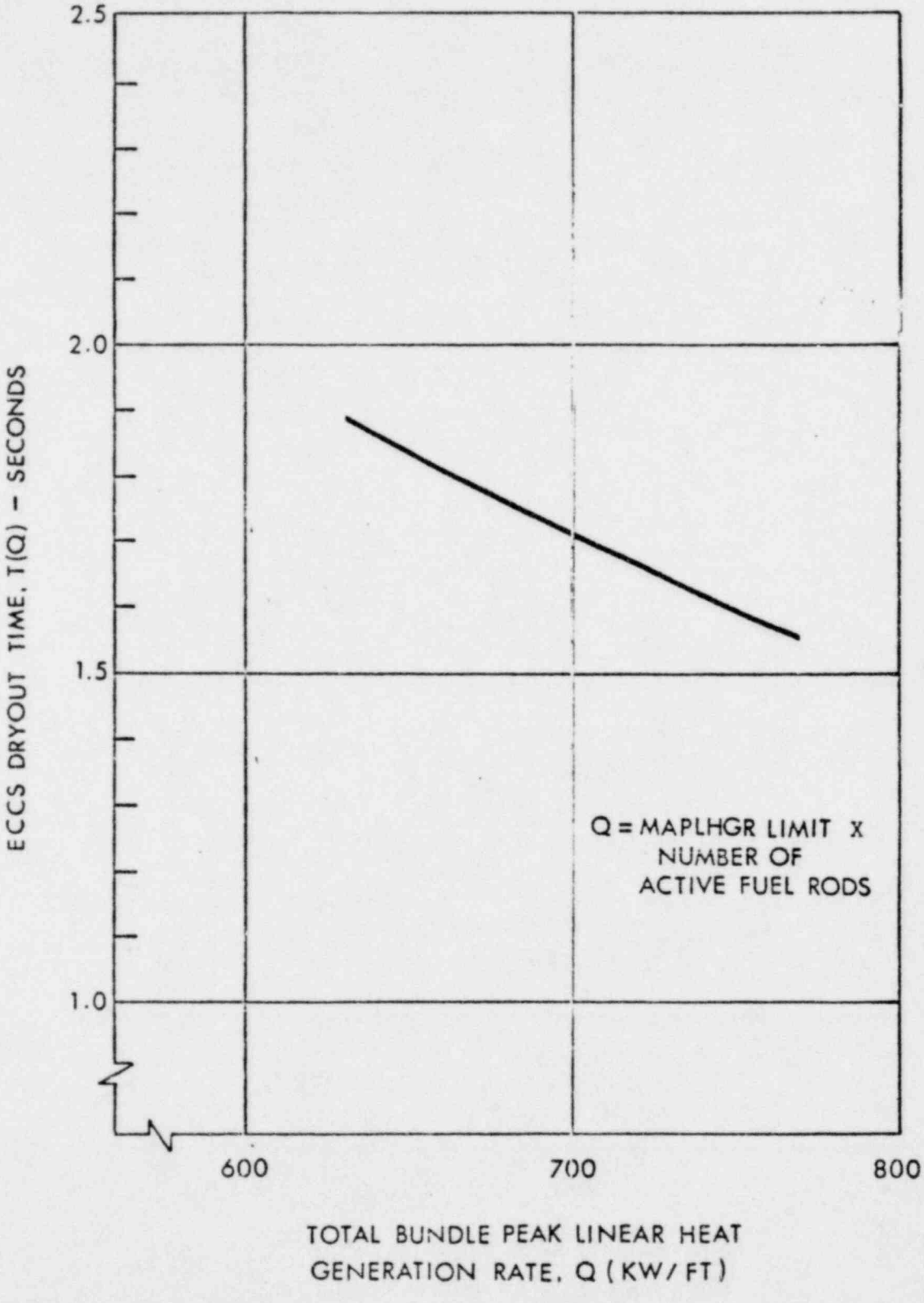


Figure 2

FUNCTION T(Q) FOR GE 9X9 FUEL
POWER VOID RELATION

TABLE 8.2

	EEI UO_2-PuO_2	Centermelt		NFS-DA
		Intermediate	Advanced	
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, $Btu/h\text{-}Ft^2$	500,000	-	-	402,000
Maximum Steady State Heat Flux, $Btu/h\text{-}Ft^2$	410,000	500,000	500,000	329,000
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
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 Circultion Tests as Outlined in Sec 8)	6×10^6	-	-	6×10^6
Number of Bundles:				
Pellet UO_2	-	1	3	-
Power 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.

Limiting Conditions for Operation	Surveillance Requirement

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.

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, -7070, -7071 and -7066 by remote manual actuation.

Leak testing of the core spray heat exchanger.

Automatic actuation of both fire pumps.

Verify that valve MO-7069 is locked or sealed in open position.

Verify that the fire system transformer deluge valve is shut and its upstream isolation valve is locked or sealed in the shut position.

Verify that the hose required for backup cooling water to the core spray recirculation heat exchanger is installed on a designated rack in the screen house.

Verify operability of the condensate fill valve to the condenser hotwell.

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Limiting Conditions for Operation	Surveillance Requirement
1.4 EMERGENCY CORE COOLING SYSTEM (Contd)	11.4.1.4 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.	B. 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. Calibration of fire system basket strainer differential pressure switches. Operability check of the core spray recirculation system through the test flow tank flow path. Verify manual and automatic actuation of the core spray system valves MO-7051, -7061, -7070 and -7071 with water flow normally blocked. Verify manual actuation of MO 7066. Verify that the hose used for backup cooling water to the core spray recirculation heat exchanger is operable and free of obvious defects. Perform a leak check and flow check of the backup cooling water hose when connected between the screen house fire water connection and the core spray recirculation heat exchanger.
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 spray effectiveness of primary core spray nozzles have been proven: (1) two condensate pumps must be operating during power operation except during startup and whem power is ≤50%, (2) the condensate storage tank level shall be ≥25% during power operation, and (3) the condensate fill valve to the condenser hotwell is operable. If (1), (2) and (3) cannot be maintained, 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.	C. 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 Core Spray

	11.3.1.4a Limiting Conditions for Operation			11.4.1.4a Surveillance Requirements	
Parameter	Trip System Logic	Limiting Set Point	Conditions for Operability	Instrument Trip Test	Instrument Calibration
Open Core Spray Valves					
Low Reactor Water Level (a)	One of Two for Each of Two Valves in Series	≥ 610'5" Elev	Power Operation and Refueling Operations (b)	Quarterly	Each Major Refueling
Primary Pressure Low (a)	One of Two for Each of Two Valves in Series	≥200 Psig	Power Operation and Refueling Operations (b)	Quarterly	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 primary system pressure.

(b) The primary core spray system shall be available for use during refueling operations. The redundant core spray system shall be inoperable during refueling operations with the valves 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 primary 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. If a passive failure of underground fire main piping should occur during the long-term cooling phase, the capability exists to bypass the affected portion of piping utilizing a fire hose to ensure the continuation of long-term ECCS cooling.

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.

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.

Bases: (Contd)

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 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.

By Commission Memorandum and Order dated May 26, 1976, Consumers Power Company was granted a plant life exemption from the single failure criterion requirements of 10 CFR Part 50, 50.46 and Appendix K, Paragraph I.D.1 for the specific case of a Loss of Coolant Accident (LOCA) caused by a break in either core spray line. This exemption was based on conditions specified in the Memorandum and Order and supporting NRC staff documents with which Consumers Power Company has complied.

Consumers Power Company has requested an exemption for Cycle 15 operation from the single failure criterion Paragraph I.D.1, Appendix K to 10 CFR 50.46. The NRC staff has granted the exemption for one cycle of operation pending completion of tests of the original ring spray nozzles.

Limiting Conditions for Operation

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, except that the power supply breaker (52-2B45) must be locked open to preclude inadvertent operation of MO-7068.

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.

Surveillance Requirement

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 11.4.1.4.

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

E. Each month verify that power supply breaker 52-2B45 for MO-7068 is locked open.

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. The backup containment spray valve (which may be normally operated upon failure of the primary containment spray valve) is disabled to preclude a single failure or inadvertent opening during a LOCA.

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.