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.54
License No. DPR-6

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

   A. The applications for amendment by Consumers Power Company (the licensee) dated December 15, 1981 and January 7, 1980 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 1;

   B. The facility will operate in conformity with the applications, 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.

8210250327

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and Paragraph 2.C(2) of Facility Operating License NO. DPR-6 is hereby amended to read as follows:

   Technical Specifications

   The Technical Specifications contained in Appendix A as revised through Amendment No. 54 , are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

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

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis M. Crutchfield

Dennis M. Crutchfield, Chief
Operating Reactors Branch #5
Division of Licensing

Attachment:
Changes to the Technical
Specifications

Date of Issuance: October 18, 1982

ATTACHMENT TO LICENSE AMENDMENT NO. 54

FACILITY OPERATING LICENSE NO. DPR-6

DOCKET NO. 50-155

Revise Appendix A Technical Specifications by removing the pages identified below and inserting the enclosed pages. The revised pages include the captioned amendment number and marginal lines indicating the area of change.

Remove Pages	Insert Pages
3-6	3-6
5-9a	5-9a
6-13	6-13
--	6-13a*
--	6-14a
7-9a	7-9a
10-20	10-20

*There are no changes to the provisions contained on this page; it is merely included for pagination purposes.

ness. The sum of leakage reates from these valves and locks shall be less that 0.25% per day of the containment atmosphere (weight basis) at 20 psig.

(b) Each reactor shutdown for refueling, but in no case at intervals greater than two years, the following valves shall be tested for operability from both the manual and automatic modes of operation and, at the same time, shall be tested for leak tightness by means of a pressure test utilizing air or the normal working fluid at a pressure not less than 20 psig:

Main Steam Isolation (MO-7050)
*Main Steam Drain (MO-7065)
Cleanup System Resin Sluice (CV-4091, CV-4092, CV-4093)
Reactor and Fuel Pit Drain Isolation (CV-4027, CV-4117)
Reactor Enclosure Clean Sump Isolation (CV-4031, CV-4102)
Reactor Enclosure Dirty Sump Isolation (CV-4025, CV-4103)

All significant leaks (drops/second) revealed by these tests shall require repair of valve seals and retests.

Automatic controls and instrumentation associated with these valves shall be tested at approximately quarterly intervals; these tests may be conducted with a simulated signal or in such other manner as to obviate plant shutdown.

(c) Each reactor shutdown for refueling, but in no case at intervals greater than two years, the following shall be visually examined for evidence of corrosion, cracking or deterioration:

All Electrical and Accessible Piping Penetration Nipple Welds
All Accessible Piping Welds to Nipples
All Expansion Joints and Welds on Expansion Joints
Potting Compound in All Electrical Penetrations

Insulation at piping penetration welds shall be removed to permit visual examination.

The probable cause of any significant corrosion, cracking or deterioration revealed by such visual examination shall be determined, and evaluated in terms of likelihood of recurrence and probable effect upon other containment sphere penetration components. An individual component leak detection test shall be performed with air at 10 psig on the faulty component prior to its repair or modification. The faulty component, and other components if necessary, shall be repaired or modified, and an individual component leak detection test performed with air at 10 psig upon each repaired or modified component. All components so repaired or modified shall be visually re-examined at appropriate intervals, but not less frequently than once every six months, until the adequacy of annual visual inspection is re-established to the operator's satisfaction.

*Operability, automatic controls, and instrumentation tests required only if valve is opened for use during operation.

TABLE 1

	Reloads: F & Modified F	Reload G	Reload G-1U	Reload G-3/G-4/ H-1/H-2
Minimum Critical Heat Flux Ratio at Normal Operating Conditions*	3.0	3.0	3.0	3.0
Minimum Bundle Dry Out Time**	Figure 1	-	-	-
Maximum Heat Flux at Overpower, Btu/h-Ft2	500,000	395,000	407,000	392,900
Maximum Steady State Heat Flux, Btu/h-Ft2	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
Maximum Bundle Power, MW_t	Figure 2 x 0.95	Figure 2	Figure 2	Figure 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	1,335	1,335	1,335	1,335
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 MW_t and 200 MW_t, and 10 MW_t per minute when power is between 200 MW_t and 240 MW_t.

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

**The actual dry out time for GE 9X9 fuel (based on the General Electric Dry Out Correlation for non-jet Pump Boiling Water Reactors, NEDE-20566) should be above the dry out time shown in Figure 1.

***For operation with only one recirculation loop in service these limits shall be reduced by 5 percent for Reload F and Modified F and reduced by 20 percent for other fuel types.

6.4.2 (Cont'd)

(b) Two of these nineteen area monitors shall be located in the vicinity of the fuel storage areas to provide gamma monitoring of the fuel storage areas and refueling operations. Local alarms shall be provided for these monitors, and alarm settings shall be in accordance with the provision of 10 CFR 70. In the event that both of these monitors become inoperable during power operation or fuel handling activities, the containment ventilation isolation valves shall be closed.

However, notwithstanding the requirements of Section 70.24(a)(1), alarm settings may be raised above 20 mR/hr as long as the overall detection criterion in Section 70.24(a)(1) is satisfied and the requirements specified in paragraph 6.4.3(e) below are met.

(c) At least five environmental film or TLD monitoring stations shall be provided for determining the integrated gamma dose rate in the site environs. These stations shall be placed on an arc of about 1,350 meters from the stack.

(d) Four narrow range water level monitors are provided in the main control room as part of the Reactor Depressurizing System to be used for detection of adequate core cooling during accident situations.

(e) The containment atmosphere shall be monitored by two high range gamma monitors. The monitors are designed to measure gamma radiation in containment under accident conditions from 1 R/hr to 1E+06 R/hr. The monitors are located external to the containment sphere. The readouts of the monitors are located in the control room.

6.4.3 <u>Operating Requirements</u>

(a) At least one of the two air ejector off-gas monitoring systems shall be in service during power operation and set to initiate closure of the off-gas isolation valve as described below. Alarms normally shall be set to annuciate in the control room if the off-gas radioactivity reaches a level that corresponds to a stack release of 0.1 curie per second. At stack releases above 0.1 curie per second, the alarm shall be set approximately a factor of two above the expected off-gas release rate but in no event above that level corresponding to a stack release of $\frac{0.47}{\overline{E}}$ curie per second where $\overline{E}$ is the average gamma energy per disintegration (MEV/dis). If the limit of $\frac{0.47}{\overline{E}}$ curie per second is exceeded, reactor power shall be immediately reduced such as to meet the limits. The monitors shall be set to initiate closure of the off-gas isolation valve (after a time adjustable from 0 to 15 minutes) if the off-gas radioactivity reaches a level that would correspond to a stack release rate of ten curies per second. Off-gas samples shall be taken monthly during power operation and analyzed for calibration of the off-gas radiation monitors. The automatic closure function of the monitors shall be tested monthly during power operation.

(b) The stack-gas monitoring system shall normally be in service. Adequate spare parts shall be on hand to allow necessary repairs to be made promptly. The alarm normally shall be set to annunciate in the control room at level that corresponds to a stack release rate of 0.1 curie per second. At stack release rates above 0.1 curies per second. the alarm shall be set approximately a factor of two above the expected stack release rate, but in no event above $\frac{0.47}{\bar{E}}$ curies per second.

The calibration of the system shall be checked at least monthly. The particulate filter and iodine filter shall be analyzed at least weekly.

(h) Both high range containment atmosphere gamma monitors shall normally be in service during power operation. If either monitor is inoperable, restore to operable status within 72 hours or, in lieu of any other report required by Specification 6.9.2, prepare and submit a Specifal Report to the Commission pursuant to Specification 6.9.4 within the next 14 days outlining the cause of the inoperablility and the plans for restoring the system to OPERABLE status. A channel check shall be performed for each monitor at least once per month, and channel calibration shall be performed at each refueling outage. The channel calibration for all ranges above 10R/hr may be performed by electronic signal substitution.

(Cont'd)

Channel comparison check of reactor level indicating instruments in the Reactor Depressurizing System	One month or less	Section 6.4.3
Calibration of reactor level indicating instruments in the reactor Depressurizing System	At each major refueling shutdown	Section 6.4.3
Steam drum safety valve position monitor check	One month or less	Section 4.1.2
Calibration of steam drum safety valve position monitors	At each major refueling shutdown	Section 4.1.2
High Radiation Trip Closure of the Containment Ventilation Isolation Valves	At each major refueling shutdown	Section 6.4.2
Channel Check [illegible] High Range Containment Gamma Monitors	One month or less	Section 6.4.3
Channel calibration of High Range Containment Gamma Monitors	At each major refueling shutdown	Section 6.4.3

Amendment No. ~~49~~, ~~51~~, 54

6.9.3 (Cont'd)

(b) Total number of samples.

(c) Number of locations at which levels are found to be significantly above local backgrounds.

(d) Highest, lowest, and the annual average concentrations or levels of radiation for the sampling point with the highest average and description of the location of that point with respect to the site.

(2) If levels of radioactive materials in environmental media indicate the likelihood of public intakes in excess of 1% of those that could result from continuous annual exposure to the concentration values listed in Appendix B, Table II, Part 20, estimates of the likely resultant exposure to individuals and to population groups and assumptions upon which estimates are based shall be provided.

(3) If statistically significant variations of offsite environmental concentrations with time are observed, correlation of these results with effluent release shall be provided.

6.9.4 Special Reports

Special Reports shall be submitted to the Director of the appropriate Regional office within the time period specified for each report. These reports shall be submitted covering the activities identified below pursuant to the requirements of the applicable technical specification section:

a. In-service inspection reports.
b. Fire system reports.
c. High range containment gamma monitoring system reports.

6.10 RECORD RETENTION

(Records not previously required to be retained shall be retained as required below commencing January 1, 1976.)

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

a. Records and logs of facility operation covering time interval at each power level.