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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 10 TO FACILITY LICENSE NO. DPR-6

CONSUMERS POWER COMPANY

BIG ROCK POINT PLANT

DOCKET NO. 50-155

INTRODUCTION

The Consumers Power Company (CPCo) has pending before the Commission a request for exemption from the Commission's ECCS single failure criterion of 10 CFR Part 50, Section 50.46 and Appendix K. The Director of Nuclear Reactor Regulation has provided comments to the Commission in connection with such exemption request[1/].

The following evaluation des ibes the staff's safety evaluation of remaining aspects of the ECCS analysis submitted by CPCo on July 25, 1975 to meet the requirements of the Commission's acceptance criterion in 10 CFR Part 50, Section 50.46 and Appendix K.

This safety evaluation also discusses the staff's evaluation of other changes in Technical Specifications for Core Reload 14 and associated changes submitted by CPCo on August 15, 1974, March 10, 1975, October 13, 1975, December 5, 1975, as amended by filings dated February 4, April 28, May 11, and May 25, 1976. These changes would modify a number of specific limitations related to fuel element geometry, Reactor Depressurization System, and containment Integrated Leak Rate Testing.

1/ 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. NRC Staff Response to Request for Information by the Commission dated May 17, 1976.

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The Big Rock Point core consists of 84 fuel assemblies with both 9x9 and 11x11 arrays. The current core (14th reload) contains the following fuel types:

36 GE type F UO_2 9x9 assemblies

2 EXXON type J-2 MOX 9x9 assemblies

4 NFS type DA MOX 11x11 assemblies

28 EXXON type G MOX 11x11 assemblies (including 8 new assemblies)

14 EXXON type G UO_2 (G-1U) 11x11 assemblies

The July 25, 1975 CPCo analysis considered type F and Modified F fuels manufactured by General Electric (GE). The results were extrapolated to include types J-1* and J-2 fuel manufactured by Exxon Nuclear Company. Exxon Nuclear performed an analysis for CPCo for their Type G fuel and for the Nuclear Fuel Services type DA fuel.

October 13, 1975, CPCo requested changes to the Technical Specifications for the facility to reload using a configuration of all uranium fuel similar in design to the 11x11 type G fuel used for the past several reloads. The core reload for Core 14 will now include 8 new type G mixed oxide assemblies and 14 type G-1U all uranium assemblies. The total quantity of mixed oxide assemblies in use would not exceed the Commission's previously authorized core load limit of 50 kilograms. On February 4, 1976, a fuel heat up analysis for the all-uranium type G-1U fuel was submitted.

*The previous cycle (Core 13) contained Exxon Type J-1 fuel assemblies.

SAFETY EVALUATION

1. Break Spectrum

The July 25, 1975 ECC submittal was in full compliance with the NRC break spectrum requirements. The worst recirculation line break is the DBA which is a postulated double ended break in one of the two 24-inch recirculating suction pipes. These pipes are the largest and lowest in the primary system, located approximately 15 feet below the bottom of the reactor vessel. The Exxon analysis for Type G fuel was performed using the same break spectrum as the CPCo July 25 submittal.

2. Fuels

Approximately 80% of the mixed oxide fuel rods in the fuel cycle beginning in May 1976, will be carried over from the previous cycle. The densification behavior of these mixed oxide rods has been measured in the Big Rock Point reactor (see XN-75-11) and a density change less than 2% of theoretical density was found. This is substantially less than the 4% theoretical density used in the Exxon densification model. The mixed oxide fuel elements to be loaded into Big Rock Point for the next fuel cycle have been sampled and tested with a 1700°C 2-hour resintering anneal, and a 1.77% theoretical density change was reported for this fuel. This resintering test has been found to conservatively estimate in-reactor density changes for UO-2 fuels, and is also expected to be conservative for mixed oxide fuels although application to mixed oxide fuels has not been confirmed. Because of the large margin that has been found (1.77% estimated versus 4% assumed in the model) the Staff believes the Exxon densification model conservatively predicts the behavior of all of the fuel in Big Rock Point.

The thermal conductivity for mixed oxide fuel is lower than for UO_2 by a small, but significant amount. WASH-1327 states that the mixed oxide thermal conductivity would be reduced approximately 5% for a 5% Pu fuel compared with UO_2. CPCo performed a revised fuel heatup analysis after making appropriate changes to the thermal conductivity in the GAPEX and HUXY computer codes.

Without changing the MAPLHGR from prior analyses, the change in peak clad temperature resulting from the decrease in thermal conductivity ranges from +7.5°F at zero exposure to +33°F at 28,000 Mwd/MTM. As a result of the re-analysis, new MAPLHGR limits were calculated to account for the decrease in thermal conductivity.

3. Submerged Equipment

On May 2, 1975 CPCo transmitted Special Report Number 21 entitled "Investigation of and Correction of Discrepancies Associated with Equipment Required to Operate During a Postulated Loss of Coolant Accident" to the NRC. CPCo in this report identified the environmental qualification of all safety related items in the containment to post accident conditions including flooding. As a result of this investigation, a number of components were relocated. More recently, in connection with the Staff review of the CPCo exemption request, CPCo was again asked to look at equipment which could be submerged and may affect ECCS equipment. The Staff was informed that five additional items had been identified which would be submerged including core spray valves control and indication. Appropriate corrective action is being required in conjunction with the CPCo exemption request.

4. LOCA Analysis and Fuel Heatup Analysis

As discussed above, the Staff determined that a sufficient number of break sizes and locations have been considered and that the worst case DBA was established.

The Big Rock Point Plant core contains G. E. Fuel in 9x9 bundles having four corner cobalt target rods and 9x9 bundles with the same target rods and a hollow perforated spacer capture rod. The Exxon bundles have an 11x11 array with 4 cobalt target rods and a passive zircalloy rod. Eight of the Exxon reload bundles are composed of mixed oxide fuel. In addition, on April 28, 1976 CPCo informed the staff by letter that Fuel assembly G-21 would have 3 peripheral urania rods replaced with solid zirconium rods of the same configuration. The staff agrees with CPCo that this change does not invalidate previous analyses or compromise the mechanical integrity of the core.

Three G. E. computer codes were used to perform the LOCA analyses:

SAFE, GEGAP-III and CHASTE

Because no credit was taken for reflooding of the core to terminate the heatup transient, use of the REFLOOD code was not required.

SAFE predicted the blowdown transients. The appropriate elevation and proximity to the reactor vessel and steam drum were modeled by SAFE for each break. No credit was taken for frictional flow effects. Output from SAFE are time of core uncovery, rated spray flow, and tables of core pressure versus time.

GEGAP-III predicted the variation of fuel pellet to clad gap conductance as a function of linear heat generation rate (LHGR) and exposure, and the behavior of fission product gas release with exposure and

CHASTE predicted the course of the heatup transient for each break, fuel type, and exposure, using the results of SAFE and GEGAP-III. Output from CHASTE is peak cladding temperature (PCT), local oxidation thickness and heat transfer coefficients as functions of time. The standard G. E. dryout correlation for non-jet pump plants, as amended December 1974, was applied during the CHASTE calculation.

SAFE and CHASTE are described in NEDE-20566, 8-74, "G.E. Analytical Model for LOCA analysis in accordance with 10 CFR 50 Appendix K." GEGAP-III is described in NEDO-20181, 11-73, "GEGAP-III: A Model for the Prediction of Pellet-Cladding Thermal Conductance in BWR Fuel Rods."

Three computer codes, comprising the Exxon Nuclear Company, INC. Non-Jet Pump Boiling Water Reactor Fuel Heatup Model (ENC-NJP-BWR-FHM), were used for the fuel heatup analysis.

The General Electric blowdown analysis with the GE SAFE code provided the necessary thermal-hydraulic input data for the Exxon models. Input data supplied from SAFE is (1) core uncovering time (2) time of rated spray and (3) fluid temperature versus time, for each break to be analyzed. All other input parameters for the Exxon codes were derived by Exxon. Although the GE SAFE code is a Staff approved evaluation model and the Exxon codes are also Staff approved, the Staff has not previously reviewed or approved their use in combination. The Staff finds the combination of codes used for the Big Rock Point application acceptable.

The GAPEX code predicted the variation of fuel pellet-to-clad gap conductance and gap size as a function of linear heat generation rate (LHGR) and exposure.

The BULGEX code predicted fuel rod failure temperature during the course of a postulated LOCA.

The HUXY code predicted the peak clad temperature (PCT), local clad oxidation, and heat transfer coefficient as function of time.

These computer codes are described in more detail in XN-235,"Exxon Nuclear Evaluation Model for BWR LOCA," October 1974. These codes were used in a manner consistent with Staff requirements, as described in "Report Regarding the Exxon Nuclear Company ECCS Non-Jet Pump BWR Fuel Heatup Model by the Office of Nuclear Reactor Regulation," March 6, 1975.

The GE Computer Code SAFE output needed by ENC-NJP-FHM as input are a) core uncovery time; b) time of rated spray; c) fluid temperature versus time; d) end of blowdown time; and e) fission heat vs time, for each break analyzed.

Staff review of the LOCA and heatup analyses resulted in a number of technical concerns all of which have been resolved to staff satisfaction.

The following discussion presents the resolution of major Staff concerns identified during the LOCA analysis review. These include 1) determination of Core Spray heat transfer coefficients for 9x9 and 11x11 fuel assemblies, 2) use of GE blowdown analysis with 9x9 fuel as input to ENC analysis with 11x11 fuel assemblies, 3) resolution of potential difference in Volumetric average fuel temperature between HUXY and GAPEX codes 4) applicability of the fission power curve used for small and intermediate size LOCAs, 5) fraction of locally generated gamma energy deposited in fuel and 6) thermal conductivity used for mixed oxide fuel.

The core spray heat transfer coefficients used in the General Electric fuel heatup analysis are shown below for one octant of the 9 x 9 array:

3.0	3.5	3.5	3.5	3.5
	1.5	1.5	1.5	1.5
		1.5	1.5	1.5
			1.5	1.5
				1.5

Units: $\frac{\text{Btu}}{\text{h Ft}^2\ {}^\circ\text{F}}$

Octant Symmetry

Channel Box HTC= 5.0

The method of assignment of these heat transfer coefficients (HTCs) is taken from the Appendix K prescription for 7 x 7 fuel (Section I.D.6.b):

"During the period after core spray reaches rated flow but prior to reflooding, convective heat transfer coefficients of 3.0, 3.5, 1.5, and 1.5 BTU-hr^{-1}-ft^{-2}-$^{\circ}$F^{-1} shall be applied to the fuel rods in the outer corners, outer row, next to outer row, and to those remaining in the interior, respectively, of the assembly."

The extension of this prescription to the 9 x 9 array is considered conservative and has been used in the past for Big Rock Point. It yields a lower rod average heat transfer coefficient for the 9 x 9 array when compared to that for the 7 x 7 array:

$$\bar{h}_{7x7} = \frac{1}{49}\,(4(3.0) + 20(3.5) + 25(1.5)) = 2.439$$

$$\bar{h}_{9x9} = \frac{1}{81}\,(4(3.0) + 28(3.5) + 49(1.5)) = 2.265$$

Since the use of lower HTCs in this portion of the heatup analysis produces higher peak clad temperatures, the choice of the above HTCs for the 9 x 9 array is justified. Similar calculations lead to the conclusion that the rod average heat transfer coefficient for the 8 x 8 array is lower than that for the 7 x 7 array. The use of these heat transfer coefficients was shown to be conservative for use in the prediction of the peak clad temperature for full-length bundles during simulated LOCA transient experiments (Ref: J. D. Duncan and J. E. Leonard, "Modeling the BWR/6 LOCA: Core Spray and Bottom Flooding Heat Transfer Effectiveness," NEDE-10801, March 1973).

The methods used in calculating the spray coefficients for the 11x11 fuel geometries are identical to the methods provided by ENC to the NRC Staff in responses to Staff questions of February 28, 1975. The method and justification for calculating the spray coefficients based on the 10 CFR 50 values for 7x7 fuel using the Oyster Creek 8x8 assemblies is presented as an example. The details pertaining to the 11x11 fuel geometries follow below:

The core spray heat transfer coefficients (HTCs) used in the Exxon Nuclear Company, Inc fuel heatup analysis are shown below for one octant of the 11 x 11 array:

2.60	3.00	3.00	3.00	3.00	3.00	Units: $\frac{\text{Btu}}{\text{h Ft}^2\ {}^\circ\text{F}}$
	1.34	1.34	1.34	1.34	1.34	Octant Symmetry
		1.34	1.34	1.34	1.34	
			1.32	1.32	1.32	Channel Box HTC=5.00
				1.27	1.27	
					1.16	

The method for assigning the above HTCs to an 11 x 11 array is an extension of the method described in the ENC report "ECCS Spray Coefficients for ENC 8 x 8 BWR Fuel Assemblies," February 28, 1975.

The Staff has reviewed the calculations presented to determine HTC's for the 11x11 fuel and conclude that they are acceptable and will result in conservative predictions of cladding temperature.

The NSSS (GE) Blowdown analysis was used as input for the ENC fuel heatup analysis. The Blowdown analysis was donducted with GE fuel in the core and the Fuel heatup analysis included ENC fuel.

The NSSS supplied results used in the ENC-NJP-BWR-FHM consist of three items of information for each LOCA transient:

1. Time of core midplane uncovery.

2. Time of rated spray.

3. Pressure versus time table.

These results describe the gross behavior of each blowdown transient. The gross behavior of a blowdown transient is governed by the energy gains and losses. The dominant energy transport is that energy lost by fluid expelled from the break. Energy added by the core to the primary fluid during blowdown is of very minor significance, as demonstrated by the following table:

Comparison of Energy Transport for Big Rock Point Limiting Break (DBA)

Time Seconds	Pressure Psia	Energy Out Break Btu/s	Energy From Core Btu/s
0.15	1321	2.03×10^7	2.33×10^5
5.66	164	1.77×10^6	1.44×10^5
10.19	17.6	1.56×10^5	3.45×10^3

Thus the blowdown transient is insensitive to energy input from the core, as this energy input is approximately 1 to 2% of the energy carried out the break.

Differences between the ENC and GE fuel would affect only the energy added by the core to the primary fluid. Other aspects of the transient would be unaffected. The major difference regarding energy added by the core is the heat transfer area difference between the ENC and GE fuel assemblies. The ratio of heat transfer area of the ENC assemblies to the GE assemblies is:

$$\frac{(116 \text{ Fuel Rods})\ (\Pi)\ (.3810 \text{ in OD})\ (h)}{(77 \text{ Fuel Rods})\ (\Pi)\ (.5625 \text{ in OD})\ (h)} = 1.020$$

Thus, if the core were entirely ENC fuel (instead of GE fuel for which the blowdown calculations were done), the heat transfer area would increase by about 2%, and the energy added by the core during blowdown would increase by about 2%.

However, a 2% increase in energy added by the core is only 1-2% times 2% = .0004 change when compared to the energy lost out the break, which is the dominant energy transport mechanism in the transient.

The staff concurs that use of the NSSS supplied blowdown results in the ENC-NJP-BWR- for this application is valid.

The heatup calculations were performed in accordance with the ENC-NJP-BWR-FHM (XN-CC-33(A)). This requires the use of the Brassfield correction of uranium conductivity for fuel other than 95% theoretical density. This correction is identical to that used in GAPEX (XN-73-25). In addition, the gap coefficients were corrected in HUXY to be consistent with the definition of gap coefficient used in HUXY. Both of the above requirements are not by input into HUXY. A comparison of the HUXY and GAPEX volumetric average fuel temperatures is included in the attached table and shows the values to be equal from the two codes. The HUXY value is directly read from the initialization of the heatup while the GAPEX value is interpolated from the tabular GAPEX results.

Volumetric Average
Fuel Temperatures

Exposure (MWd/MTM)	kW/Ft	Temperatures HUXY	GAPEX
0	5.109	1339	1343
2250	5.506	1156	1159
11000	5.810	1102	1106
21000	5.829	975	978

Based on sensitivity studies and comparative analysis, as reported and discussed in NEDO-20566, the NSSS supplier concluded that the normalized core power versus time can be adequately represented by a single power versus time curve regardless of plant or fuel design differences. The NSSS supplier power versus time envelope curve representing the most conservative shape was used in the 11 x 11 fuel heatup analysis.

A value of 0.96 was used for the fraction of locally generated gamma energy deposited in fuel pins and represents the historical value for Big Rock Point (Ref: CPCo submittal of 11/1/72, "Small and Intermediate Break LOCA Analysis for the Big Rock Point Reactor with ADS and JNC Reload G Fuel").

GE licensing Topical Report, NEDO-20214 presents calculations of gamma heating distributions in a BWR core. This report has been reviewed by the Staff. On the basis of the data presented, Big Rock Point calculations using a .96 value for locally generated gamma energy deposited in fuel pins is acceptable.

The February 4, 1976, CPCo submittal transmitted Exxon report, "Heatup Analysis for Exxon Nuclear Company, Inc, All-uranium G fuel in the Big Rock Point Plant in Conformance with 10 CFR 50, Appendix K" dated January 1976. Also included in the February 4th submittal was MAPLHGR limits for all fuel types in the Big Rock core. These limits are included in Table 2.

The ENC reload fuel contains assemblies which differ in two respects from G fuel now in the BRP core. First, these new assemblies will have no fuel rods containing mixed-oxide material. Therefore, this new fuel is called "all-uranium G fuel" to distinguish it from the "mixed-oxide G fuel" analyzed in the July 25, 1975 submittal. Second, the new fuel assembly has four inert Zircaloy rods in the central region of the rod array, rather than just one inert rod as exists in the

TABLE 2

MAPLHGR (kW/Ft)

Planar Average Exposure (MWd/STU)	Modified F	E-G, F, J-1*, J-2	Reload G & NFS-DA	Reload G-1U
0	-	-	6.38	6.40
200	9.5	9.4	-	-
1,000	-	-	-	6.86
2,205	-	-	-	6.87
2,480	-	-	6.79	-
5,000	9.9	9.7	-	-
5,511	-	-	6.76	6.90
10,000	9.9	9.7	-	-
11,023	-	-	-	7.05
12,125	-	-	6.86	-
15,000	9.8	9.6	-	-
16,534	-	-	6.97	-
17,637	-	-	-	7.25
20,000	8.7	8.6	-	-
22,046	-	-	-	7.25
23,148	-	-	6.95	-
25,000	8.7	8.3	-	-
30,864	-	-	7.05	-
33,069	-	-	-	7.28

*The previous cycle (Core 13) contained Exxon Type J-1 fuel assemblies.

center of the mixed-oxide G fuel rod array. The all uranium fuel is called G-1U fuel to distinguish it from the mixed oxide G-1 fuel. This analysis was performed under the same assumptions as the mixed-oxide G fuel analysis in regard to ground rules, single failures and NSSS vendor blowdown calculations.

The all-uranium G fuel has the following characteristics:

Array: 11x11
4 Corner Cobalt Target Rods
4 Inert Zircaloy rods
Rod OD: 0.449 Inches
Clad Thickness: 0.034 Inches
Active Fuel Length: 70 Inches
Rod Pitch: 0.577 Inches
Design Axial Peaking: 1.51
Maximum LHGR: 14.0 kW/Ft

The July 25, 1975 CPCo submittal identified the Design Basis Accident (DBA) as the worst break for both 9x9 and 11x11 fuel types. The NSSS vendor blowdown results for the DBA were input to the ENC heatup codes for this analysis. Other breaks were not analyzed.

The GAPEX code (version DEC74A) was used to predict the variation of fuel pellet-to-clad gap conductance and gap size as a function of linear heat generation rate and exposure.

The HUXY9 code (version JUL75A) predicted the peak clad temperature (PCT) and local clad oxidation as a function of time.

These are the same versions of GAPEX and HUXY that were used in the mixed-oxide G fuel analysis. These versions are approved by the Staff for use in the ENC ECCS Non-Jet-Pump BWR Fuel Heatup Model.

Based on the criterion of limiting the PCT to 2200°F or less, the MAPLHGR limits as a function of exposure for the G-1U Fuel is presented in Table 2. These limits result in a small derating (average about 5%) for Big Rock Point.

Local clad oxidation was always less than the limiting fraction of 0.17. A core-wide metal-water oxide fraction calculation was not performed as this calculation for mixed-oxide G fuel in the July 25, 1975 submittal predicted that only 0.0018 of the total clad reacted in the DBA. The minor differences in the all-uranium G fuel design as compared with the mixed-oxide G fuel design should not lead to a core-wide oxide fraction which exceeds 0.01.

5. Nuclear Design

A review of the nuclear design was conducted by the Staff. Available data on some physics parameters are several years old. Although it is still generally applicable and no serious deficiencies were found, the Staff considers it desirable to update all physics information in the near future. We therefore conclude that prior to startup following the next refueling, CPCo should provide more definitive nuclear design information.

6. Reactor Depressurization Systems (RDS)

A. Discussion

By letter dated August 15, 1974, CPCo submitted a report entitled "Big Rock Point Plant Reactor Depressurization System Description, Operation, and Performance Analysis." The NRC staff review resulted in questions relating to: 1) the safety margin provided in the design and construction of the system structural elements, 2) pressure drop calculations, and 3) physical separation criteria to prevent a hot short in the trip logic modules. Subsequent letters from CPCo dated November 14, 1974 and December 17, 1974 responded to the staff concerns.

On March 10, 1975, CPCo proposed technical specifications for periodic testing of the RDS. In addition, their March 10 letter provided answers to other staff concerns relating to the consequences of an inadvertent blowdown through a system bypass line. Additional CPCo letters dated April 29, 1975 and October 9, 1975 provided revisions to the August 1974 report and a revised fluid system failure mode effects analysis.

B. Evaluation

The licensee's analyses show that the calculated peak clad temperatures for small and medium size pipe breaks with the RDS are lower than for a maximum recirculation line break. These calculations were based on three functioning blowdown legs so that the failure of a single valve would not incapacitate the system.

A small bypass line around the inside isolation valve is used to maintain system pressure and temperature in the piping between the two isolation valves in each leg. There is a manual shutoff valve in each bypass line and a remotely operated shutoff valve in the steam header that feeds the four lines. These valves are normally open, however, the remote operated shutoff valve will be closed when the isolation valves in the depressurization lines are tested to limit the amount of blowdown to the containment. Similarly, the bypass line shutoff valve can be used to minimize any inadvertent blowdown caused by the spurious opening of an outer isolation valve.

The main isolation valve on the depressurization line is air operated. In the event of failure of the instrument air supply, an unwanted actuation of the valve would depend on the air leakage from the valve operator. The licensee will perform periodic system tests to assure that the air leakage rates are within specifications. An inadvertent activation of the main isolation valve due to loss of instrument air would not result in a blowdown because the second depressurization valve in the line would not be affected.

The depressurization system is designed to perform its function even with a single failure and the design contains features to minimize the effects of an inadvertent blowdown in the containment. Nevertheless, we have analyzed the radiological consequences of an inadvertent blowdown through the bypass line around the inside isolation valve. Our analysis assumes that, in the event the containment is being purged at the time of the blowdown, the offgas system, which is used for purging, can be isolated within a short time. We have determined that the two hour thyroid dose for I-131 equivalent is small (on the order of 0.2 REM) and is acceptable.

We have reviewed the procedures for testing the RDS valves and specifically the effects of this testing in the containment system for the Big Rock Point Plant and found the procedures acceptable. The bypass isolation valve will terminate an inadvertent blowdown through the 1 1/2-inch bypass line. Prior to the test, the bypass isolation valve, which is normally open, will be closed. With this valve closed, a continuous blowdown through the bypass line cannot occur during the test. Therefore, steam release will be limited to the steam volume trapped between the depressurizing valve and the bypass isolation valve. We have calculated the trapped steam to be about five pounds. We conclude that there would be no adverse effect on the containment integrity from this release.

The staff has reviewed the structural aspects of the RDS design to determine its compliance with the provisions of Document "B" entitled "Structural Design Criteria for Evaluating the Effects of High-Energy Pipe Breaks in Category I Structures Outside the Containment" dated June 1973. The seismic design has been reviewed for compliance with Regulatory Guides 1.60 and 1.61. Our review indicates that the new elements of the RDS will not affect the capability of existing structures or components to perform the necessary safety functions and that the new structural elements will have at least the same margin of safety as the existing structural elements.

We have reviewed the instrumentation, controls, and electrical power supply portions of the RDS design for conformance to the single failure criterion. Based on the information provided, we conclude the following:

(1) No single failure within the instrumentation, controls, or electrical power supplies of the system will prevent the operation of at least three of the four depressurization channels. Therefore, the single failure criterion is satisfied since operation of only three of the four depressurization channels is required.

(2) A single failure within the instrumentation, controls, or electrical power supplies of the system could not result in spurious operation of a depressurization channel. In addition, the system design is such that the output circuits of the two trip logic modules in each channel are physically separated by a minimum distance of 15 inches.

C. Conclusion

Based on our review of the information provided by CPCo and on the discussion contained in this section and in the ECCS evaluation above, we have concluded that the proposed RDS and periodic test requirements, as modified after discussion with CPCo, are acceptable.

7. Integrated Leak Rate Test (ILRT)

A. Discussion and Evaluation

CPCo's letter of December 5, 1975, proposed changes to upgrade the entire Big Rock Point Technical Specifications. Since NRC review of the proposed new Technical Specifications has not been completed, one area has been identified by CPCo which has a direct impact on the startup schedule from the present refueling outage. The item of concern is the ILRT test interval, now specified as once every two years. Appendix J to 10 CFR 50 allows a test interval of three years. Thus, Big Rock Point's existing Specification 3.7(e) requires additional ILRT not currently required by NRC.

Since Appendix J requires the ILRT on the containment sphere at approximately three equal intervals during each 10-year service period, we have concluded that the proposed specification is acceptable on the basis that it conforms with the test interval of the current regulation. Furthermore, the proposed specification would conform with the specifications for plants being licensed today.

8. Environm al Considerations

The Commission's staff has evaluated the potential for environmental impact associated with operation of Big Rock Point in the proposed manner. From this evaluation, the staff had determined that there would be no change in effluent types or total amounts, no change in authorized power level and no significant environmental impact attributable to the proposed action. Having made this determination, the Commission has further concluded pursuant to 10 CFR Section 51.5(c)(1) that no environmental impact statement need be prepared for this action. A Negative Declaration and supporting Environmental Impact Appraisal are being issued with this amendment to the license. As required by Part 51, the Negative Declaration is being filed with the Office of the Federal Register for publication.

9. Conclusion

Based on our evaluation of reactor operation with Reload-14 fuel, we have concluded that because the proposed changes do not involve a significant increase in the probability or consequences of accidents previously considered and do not involve a significant decrease n a safety margin, the changes do not involve a significant hazards consideration and that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner. Based on our evaluation of operating limits using an acceptable ECCS evaluation model, we have concluded that there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner.

In light of the exemption granted, we have concluded, based on the considerations discussed in the evaluation that all of the activities discussed herein will be conducted in compliance with the Commission's regulations. We also conclude that the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: June 4, 1976