Mr. Patrick M. Donnelly, Plant Manager
Big Rock Point Plant
Consumers Power Company
10269 U.S. 31 North
Charlevoix, MI 49720

January 29, 1997

SUBJECT: REVIEW OF INDIVIDUAL PLANT EXAMINATION - INTERNAL EVENTS - BIG ROCK POINT PLANT (TAC NO. 74381)

Dear Mr. Donnelly,

By letters dated May 5, and May 27, 1994, you responded to Generic Letter (GL) 88-20, "Individual Plant Examinations for Severe Accident Vulnerabilities," and associated supplements. By letter dated April 4, 1996, you responded to both of our requests for additional information (RAI) dated January 29, and March 1, 1996. The staff has completed its review of the Big Rock Point Individual Plant Examination (IPE) submittals for internal events and internal flooding. The results of the review are enclosed and consist of a Staff Evaluation Report (Enclosure 1) and a Technical Evaluation Report (Enclosure 2). Your submittals also addressed the Individual Plant Examination for External Events (IPEEE) in response to Supplement 4 to GL 88-20. Our review of the external events portion will be addressed in separate correspondence.

The Big Rock Point IPE submittals did not identify any severe accident vulnerabilities associated either with core damage or poor containment performance. Based on our review of the IPE submittals and associated documentation, we conclude that Consumers Power Company has met the intent of GL 88-20, including Supplements 1, 2, and 3.

Generic Letter 88-20 suggested that licensees could use their IPE submittal to address other safety issues. In your submittals, you had proposed to resolve Unresolved Safety Issue A-45, "Shutdown Decay Heat Removal Requirements," and Unresolved Safety Issue A-43, "Containment Sump Emergency Performance." Your IPE submittals adequately resolve these issues for Big Rock Point Plant.

If you have any questions or comments, please contact me at (301) 415-1361.

Sincerely,
Original signed by
Linh N. Tran, Project Manager
Project Directorate III-1
Division of Reactor Projects - III/IV
Office of Nuclear Reactor Regulation

Docket No. 50-155

Enclosures: 1. Staff Evaluation Report
2. Technical Evaluation Report

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cc w/encl: See next page

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NUCLEAR REGULATORY COMMISSION
WASHINGTON, D.C. 20555-0001

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cc:
Mr. Thomas A. McNish
Vice President & Secretary
Consumers Power Company
212 West Michigan Avenue
Jackson, Michigan 49201

Judd L. Bacon, Esquire
Consumers Power Company
212 West Michigan Avenue
Jackson, Michigan 49201

Jane E. Brannon, County Clerk
County Building Annex
203 Antrim Street
Charlevoix, Michigan 49720

Office of the Governor
Room 1 - Capitol Building
Lansing, Michigan 48913

Regional Administrator, Region III
U.S. Nuclear Regulatory Commission
801 Warrenville Road
Lisle, Illinois 60532-4351

Drinking Water and Radiological
Protection Division
Michigan Department of
Environmental Quality
3423 N. Martin Luther King Jr Blvd
P. O. Box 30630 CPH Mailroom
Lansing, Michigan 48909-8130

U.S. Nuclear Regulatory Commission
Resident Inspector's Office
Big Rock Point Plant
10253 U.S. 31 North
Charlevoix, Michigan 49720

Mr. Robert A. Fenech
Vice President-Nuclear Operations
Palisades Plant
27780 Blue Star Memorial Hwy.
Covert, Michigan 49043

Michigan Department of Attorney
General
Special Litigation Division
630 Law Building
P.O. Box 30212
Lansing, Michigan 48909

BIG ROCK POINT NUCLEAR POWER PLANT INDIVIDUAL PLANT EXAMINATION

STAFF EVALUATION REPORT

Enclosure 1

I. INTRODUCTION

On May 5, 1994, and on May 27, 1994 (consisting of a reissue of Section 3.0, "Results and Screening Process"), Consumers Power Company submitted the Big Rock Point Nuclear Power Plant (BRPNPP) individual plant examination (IPE) in response to Generic Letter (GL) 88-20 and associated supplements. On January 29, 1996 and on March 1, 1996, the staff requested additional information from the licensee. The licensee responded to both requests in a letter dated April 4, 1996.

The staff performed a "Step 1" review of the BRPNPP IPE submittal. That Step 1 review focused on whether the licensee's method was capable of identifying vulnerabilities. Therefore, the review considered (1) the completeness of the information and (2) the reasonableness of the results given the BRPNPP's design, operation, and history. A more detailed review, a "Step 2" review, was not performed for this IPE submittal. Brookhaven National Laboratory (BNL) supported the staff in this review. Details of BNL's findings are given in the technical evaluation report (Appendix A) attached to this staff evaluation report (SER). The staff encourages the licensee to consider the contractor's findings in its future updates of the BRPNPP probabilistic risk assessment (PRA).

As requested by the staff in GL 88-20, the licensee proposed to resolve Unresolved Safety Issue (USI) A-45, "Shutdown Decay Heat Removal Requirements," as part of the BRPNPP IPE. In addition, the licensee proposed that USI A-43, "Containment Sump Emergency Performance", be resolved as part of the BRPNPP IPE. No other specific USIs or generic safety issues (GSIs) were proposed for resolution by the staff or by the licensee as part of the BRPNPP IPE.

II. EVALUATION

BRPNPP is a boiling water reactor (BWR) with a large, dry containment. The licensee estimated a total core damage frequency (CDF) of about 5E-5/reactor-year, including a 1E-9/reactor-year contribution from internal flooding. Loss-of-coolant accidents (LOCAs) contribute about 73 percent; transients, about 12 percent; anticipated transients without scram (ATWS), about 7 percent; steamline breaks inside containment, 6 about percent; station blackout (SBO), about 1 percent; and internal flooding, a very small fraction of 1 percent. Important contributors to the CDF are failures of the post-incident system (a system analogous to a low pressure recirculation system in a PWR), the reactor depressurization system, and the core spray system.

It is noted that, although the CDF compares reasonably with the CDF of other BWR plants, the risk profile for BRPNPP does not look like that of a typical BWR, where SBO and ATWS usually dominate the CDF. The SBO contribution at BRPNPP is small (1 percent) due to the existence of a 100 percent load rejection capability, an emergency condenser, the ac-independent makeup to the emergency condenser, the long life of the alternate shutdown battery, and the existence of two diesel generators (albeit with limited capability). The relatively small ATWS contribution at BRPNPP (7 percent) is governed by two

opposing forces: less time than at other BWRs is available for injection of the standby liquid control system (SLCS) because there isn't a high-pressure, high-volume emergency core cooling system (ECCS) at BRPNPP; however, the SLCS at BRPNPP is a fast-acting one that ensures subcriticality in about 1 minute after operator actuation. There are several reasons for the high LOCA contribution: a portion of the primary system's piping is located below the level of the core, which leads to a more severe class of LOCAs; there is paucity of high-pressure, high-flow-rate makeup systems; for larger LOCAs, makeup to the condenser hotwell is inadequate, which leaves the two fire pumps as the only low-pressure system available; some important systems would be disabled by the harsh environments caused by LOCAs or steam-line breaks; lack of a suppression pool means that at some point recirculation must be brought into play; and finally, no credit is given for proceduralized action to flood the spherical containment structure, with its passive cooling features, if recirculation fails.

On the basis of the licensee's IPE process used to search for decay heat removal (DHR) vulnerabilities, and the review of BRPNPP plant-specific features, the staff finds the licensee's DHR evaluation consistent with the intent of the resolution of USI A-45. In addition, the licensee proposed that USI A-43 be resolved as part of the BRPNPP IPE. GL-88-20 states that if a licensee "concludes that no vulnerability exists at its plant that is topically associated with any USI or GSI, the staff will consider the USI or GSI resolved for a plant upon review and acceptance of the results of the IPE." The staff notes that no such vulnerability was identified at BRPNPP and, therefore, concludes that USI A-43 is resolved for BRPNPP.

The licensee performed a human reliability analysis (HRA) to document and quantify potential failures in human-system interactions and to quantify human-initiated recovery from failure events. The licensee identified the following operator actions as important in the estimate of the CDF: manual loading of the emergency diesel generator (EDG) or the standby diesel generator (SDG) onto the emergency bus, manual starting of the SDG, alignment and starting of the portable diesel pump for emergency condenser makeup, manual depressurization or pressure control, actuation of the liquid poison system, recovery of the recirculation system (referred to as the post-incident system (PIS) - although it is noted that credit for operation of this system was not taken in the IPE - and proceduralized action to flood the spherical containment structure, with its passive cooling features, if recirculation fails (also not credited in the IPE).

The licensee evaluated and quantified the results of the severe-accident progression through the use of a containment event tree and considered uncertainties in containment response through the use of sensitivity analyses. The licensee's back-end analysis appeared to have considered important severe-accident phenomena. Of the 6 percent conditional containment failure probability, early containment failure is about 4 percent with containment overpressure failure before vessel breach in ATWS events the primary contributor; late containment failures are negligible due to the large containment volume and the use of a 36-hour (after vessel failure) mission time; and bypass is about 2 percent with failure to isolate the process lines

that connect to the primary system the principal contributor. The containment remains intact about 94 percent of the time. Early radiological releases are dominated by ATWS and LOCA sequences with enclosure spray available and, as noted previously, late releases are negligible. The licensee's response to containment performance improvement program recommendations is consistent with the intent of GL 88-20 and its Supplement 3.

The licensee identified the following insights and unique plant safety features at BRPNPP:

(1) The primary water inventory is large relative to core thermal power and decay heat levels.

(2) The emergency condenser (EC) for high-pressure core cooling and makeup enables passive cooling of the reactor without reliance on ac power (diesel-driven pumps can be used for shell-side makeup).

(3) The firewater system, consisting of one diesel-driven pump and one electric pump, takes suction from Lake Michigan and can be used as part of the ECCS.

(4) Should recirculation be unavailable for continued core cooling, continuing in the injection mode will result in almost filling the spherical containment with water, which will then provide core cooling via natural circulation and air cooling of the containment steel shell.

(5) The emergency ac power consists of one 200-kW emergency diesel generator (EDG) and one 250 kW standby diesel generator (SDG). The absence of additional diesel capacity would be detrimental, compared to other BWRs, were it not for the fact that the plant relies extensively on passive features. The core can be cooled without ac power. Two battery banks, i.e., the normal and the alternate shutdown battery bank, supply dc power. The alternate shutdown battery bank supplies the post-initiator loads of interest, and is sized so that a blackout can be survived for about a week. The emergency power system requires no support function.

(6) Control rod drive (CRD) pumps cannot be used in conjunction with safety valve cycling or actuation of the reactor depressurization system because of high temperature in the CRD pump room, a negative plant feature. In general, this plant seems to be more vulnerable to environmental conditions (some of the other systems vulnerable to harsh conditions are the reactor cooling water system, emergency condenser outlet valves, primary core spray valves, the reactor pressure and level instrumentation, and some operator actions) than newer BWRs.

(7) The instrument air system has three air compressors; one is sufficient for system success.

(8) The plant has a recently improved "100 percent load rejection capability." However, this positive feature has not been entirely proven in practice.

(9) The plant has no high-flow-rate, high-pressure ECCS pumps, a negative feature, except in some ATWS sequences where this feature prevents containment failure.

(10) The plant has a fast-acting, passive, manually initiated liquid poison system in case of ATWS.

(11) The plant has a single, two-train, low-pressure ECCS for LOCA evolutions

(12) A portion of primary system piping is located below the core midplane.

(13) The plant has no sign of the classic symptoms of intergranular stress corrosion cracking (IGSCC) found at other BWRs.

(14) The plant utilizes a spherical steel containment vessel to provide a large, dry containment, which effectively decouples containment considerations from the Level 1 analysis. In comparison with other plants that use large, dry containments (pressurized-water reactors), the containment volume-to-thermal power ratio for BRPNPP is significantly (about four times) higher.

(15) The sump beneath the reactor vessel is large enough to hold the entire core debris.

The licensee defined a vulnerability as "new or unusual means of reaching a situation in which core damage or containment failure would occur, or if PRA results indicated BRPNPP would prevent the industry from meeting published safety goals." On the basis of this definition, the licensee did not find any vulnerabilities. No plant improvements were identified nor are any planned, other than minor emergency operating procedure (EOP) modifications for SBO.

III. CONCLUSION

On the basis of these findings, the staff notes that (1) the licensee's IPE is complete with regard to the information requested by GL 88-20 (and associated guidance in NUREG-1335) and (2) the IPE results are reasonable given BRPNPP's design, operation, and history. As a result, the staff concludes that the licensee's IPE process is capable of identifying the most likely severe accidents and severe accident vulnerabilities and, therefore, that the BRPNPP IPE has met the intent of GL 88-20.

It should be noted that the staff focused its review primarily on the licensee's ability to determine whether severe-accident vulnerabilities exist at BRPNPP. Although certain aspects of the IPE were explored in more detail than others, the review is not intended to validate the accuracy of the licensee's detailed findings (or quantification estimates) that stemmed from the examination. Therefore, this SER does not constitute NRC approval or endorsement of any IPE material for purposes other than those associated with meeting the intent of GL 88-20.

BROOKHAVEN NATIONAL LABORATORY
TECHNICAL EVALUATION REPORT

Enclosure 2