Consumers Power Company

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August 17, 1979

Director, Nuclear Reactor Regulation
Att Mr Dennis L Ziemann, Chief
Operating Reactors Branch No 2
US Nuclear Regulatory Commission
Washington, DC 20555

DOCKET 50-155 - LICENSE DPR-6 - BIG ROCK POINT PLANT - RESPONSE TO REQUEST FOR INFORMATION FOR NRC STAFF GENERIC REPORT ON EFFECT OF THREE MILE ISLAND ACCIDENT ON BWRs

NRC letter dated July 17, 1979 requested Consumers Power Company to provide information for use in preparing the subject report. Similar information requests were sent to owners of other General Electric Company BWRs. This information has been collected for all operating General Electric Company BWRs, including Big Rock Point, in a generic report. This approach was discussed in a letter from T D Keenan (Chairman, General Electric Operating Plants Owners' Group) to D F Ross (NRC) dated July 24, 1979.

The General Electric Company generic report, NEDE-24708 "Additional Information Required for NRC Staff Generic Report on BWR Reactors," is being transmitted to D F Ross by letter from T D Keenan concurrently with this letter. As discussed in NEDE-24708, meetings between the Owners' Group and NRC staff resulted in agreement that certain items requested by the July 17, 1979 letter need not be provided until November 1979. A future revision of NEDE-24708 will provide this additional information.

General Electric Company report NEDE-24708 should be included in Docket 50-155. This report should be considered as including Consumers Power Company's response to those items of the July 17, 1979 letter for which it was agreed that a short-term response was required.

As one of the oldest General Electric BWRs, Big Rock Point differs in design from newer plants. Some of the differences are particularly applicable to the occurrences discussed in NEDE-24708 (small break loss of coolant accidents and loss of feed-water transients). Particular features which should be

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considered in reviewing NEDE-24708 as it applies to Big Rock Point are briefly discussed in the attachment to this letter.

David A Bixel (Signed)

David A Bixel
Nuclear Licensing Administrator

CC JGKeppler, USNRC

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ATTACHMENT

Specific Design Features of Big Rock Point Nuclear Plant
To Be Considered in Application of NEDE-24708,
"Additional Information Required for NRC Staff Generic Report on BWR Reactors"

1. The Three Mile Island Accident was caused, in part, by failure of a power operated relief valve to close after opening as a result of a plant transient. NRC requests which led to the preparation of NEDE-24708 indicate that a principal purpose of the current NRC review will be to determine the effect of a stuck open relief valve in a BWR. This will apparently include consideration of whether the probability of relief valve failures increases the likelihood of a small break loss of coolant accident above that considered in previous NRC reviews. NEDE-24708 includes information relative to the effect of a stuck open relief valve and future revisions will include information on relief valve performance.

   A design difference between Big Rock Point and newer BWRs of particular significance with respect to this area is the lack of reliance on relief valves for mitigation of plant transients. Big Rock Point has no power operated relief valves (other than those used in the reactor depressurization system discussed below). Six spring-loaded safety valves are provided. The lowest relief valve set point is 1,535 psig compared to operating pressure of 1,335 psig. The plant is also equipped with an emergency condenser which actuates automatically at 1,435 psig and has a design capacity of 4% rated reactor power. Operation of the emergency condenser on high pressure following scram is sufficient to cause a reduction in system pressure such that relief valves are not lifted. Transient analyses have consistently shown that no anticipated transient is expected to result in opening of a relief valve if plant equipment functions as designed. A failure of each of two redundant loops in the emergency condenser (two active failures) would have to occur before a relief valve would be lifted by a plant transient.

2. Big Rock Point design does not include all of the high-pressure coolant injection systems found on newer BWRs. Capability to mitigate small loss of coolant accidents is provided, instead, by a reactor depressurization system (RDS). RDS involves four trains of two valves each (one block valve and one pilot operated relief valve) which are intended to depressurize the reactor vessel to permit operation of the low-pressure core sprays. The system is actuated by low steam drum water level (20 inches below drum center line) which starts a 120-second timer and also initiates fire pump (source of water for core sprays) start signals. Upon indication of low water level in the reactor vessel (2'9" collapsed level above the top of active fuel) RDS valves are opened provided the 120-second timer has expired and there is high pressure in the fire suppression system piping. No coincident high containment pressure signal is required for RDS actuation; thus, RDS would operate for small breaks located outside containment in a manner identical to that for a break inside containment.

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Big Rock Point has a dry containment with large free volume and no pressure suppression pool. RDS discharges to the containment atmosphere. Since introduction of steam to the containment atmosphere is undesirable, RDS is utilized exclusively for mitigation of small break loss of coolant accidents. The system is designed so that a single failure would neither cause the system to operate nor prevent it from operating. None of the RDS trains has ever opened during three years of operation.

3. As discussed in 1 above, the emergency condenser system at Big Rock Point provides principal protection against overpressurization and acts as a heat sink if the primary condenser is unavailable. The condenser consists of a horizontal drum vented to the atmosphere and contains two piping bundles. In operation, steam is admitted to each bundle, or loop, from the main steam header and condensate is returned to the bottom of the steam drum. Valves in the steam inlet lines are open during reactor operation and the outlet valves are normally closed. The outlet valves are operated by d-c motors and thus not affected by loss of a-c power which would render the principal heat sink (main condenser) unavailable. A water supply sufficient for four hours of operation is maintained within the emergency condenser. Makeup water can be provided from two sources (demineralized water and fire suppression water).