Consumers Power

POWERING MICHIGAN'S PROGRESS

Big Rock Point Nuclear Plant, 10269 US-31 North, Charlevoix, MI 49720

William L Beckman
Plant Manager

October 15, 1990

Nuclear Regulatory Commission
Document Control Desk
Washington, DC 20555

DOCKET 50-155 - LICENSE DPR-6 - BIG ROCK POINT PLANT -
OPERATION WITH ONE LOOP OUT OF SERVICE

During a recent feedwater pump trip event and subsequent manual recirculation pump trip at Big Rock Point on March 1, 1990, some questions and concerns were raised dealing with the isolation requirements contained in our plant operating procedures. The current procedures state that extended operation (greater than one hour) with one loop idle requires:

1) The suction, discharge, and discharge bypass valves of the inactive loop to be closed and caution tagged.

2) The inactive pump motor breaker to be opened and lowered.

3) A determination of the maximum allowable reactor power to that permitted by Technical Specifications for one loop operation.

These guidelines were established during resolution of Systematic Evaluation Program (SEP) Topic IV-1.A - Operation With Less Than All Loops in Operation. As discussed in NRC letter dated August 9, 1979 (Attachment 1), the analysis used to justify operating in the N-1 mode assumes that the inactive loop is isolated allowing no bypass flow through the inactive loop. The second restriction dealt with concerns associated with a cold water accident. This led to the recommendation that the power to the inoperative pump be locked out during N-1 loop operation to provide additional assurance that an inadvertent cold water injection accident will not occur. This evaluation also noted that this restriction may be removed by performing a reanalysis of the cold water accident and by obtaining the acceptance of that analysis from the NRC staff.

In a recent review of these operating restrictions, a concern regarding reliability of the recirculation pump seals has been raised. If a pump is removed from service for other than a seal failure, completely isolating the pump can lead to seal degradation during restart. If the plant operators are unable to start the inactive pump within the one hour limit, closing the suction valve in addition to the discharge and discharge bypass valves will cause the pump shaft seals to begin to depressurize. When pump restart is

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conducted, the pressure transient on the pump seals may cause them to seal improperly initiating shaft seal leakage leading to pump isolation and unit derate. Leaving the suction valve open maintains system pressure on the seals during shutdown, minimizing the transient affects during restart.

In order to reduce the potential for seal degradation, the operating procedures for one pump operation have been revised to:

1) Close and caution tag the discharge and discharge bypass valves of the inactive loop.

2) A determination of the maximum allowable reactor power permitted by Technical Specifications for one loop operation.

Justification for these changes is contained in the attached 10CFR50.59 evaluation (Attachment 2); however, it can be summarized as follows:

1) Closing the discharge and discharge bypass valves alone prevents reverse flow through the idled pump. The suction valve would only be redundant to the discharge and discharge bypass valves.

2) The power lockout requirement of the idled loop was based upon not having an accepted analysis for single loop operation with restarting the idled pump (cold water injection). The analysis was performed for SEP Topics XV-1, XV-3, XV-4, XV-5, XV-7 and XV-9 and submitted on July 15, 1981. In a letter dated April 7, 1982 (Attachment 3), the staff reviewed and accepted the analysis.

Although changes made to plant procedures pursuant to 10CFR50.59 do not require NRC approval, Consumers Power Company felt it was appropriate to notify the staff of these changes. Although none of the initial procedural controls discussed earlier appear in the plant Technical Specifications, they do appear in the staff's Safety Evaluation for Amendment No. 48 dated September 21, 1981 (Attachment 4).

W L Beckman

W L Beckman
Plant Manager

CC Administrator, Region III, USNRC
NRC Resident Inspector - Big Rock Point

Attachments

OC0990-0074-NL04

ATTACHMENT 1

Consumers Power Company
Big Rock Point Plant
Docket 50-155

NRC LETTER DATED AUGUST 9, 1979

12 Pages

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Assessment of single loop operation for SEP topic IV-1.A

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

AUG 14 1979
NUCLEAR LICENSING

August 9, 1979

Docket No. 50-155

Mr. David Bixel
Nuclear Licensing Administrator
Consumers Power Company
212 West Michigan Avenue
Jackson, Michigan 49201

Dear Mr. Bixel:

RE: TOPIC IV-1.A - OPERATION WITH LESS THAN ALL LOOPS IN OPERATION

Enclosed is a copy of our revised safety assessment of Topic IV-1.A, Operation With Less Than All Loops In Operation. This revision supersedes the evaluation issued by our letter dated August 17, 1978.

This revision completes our assessment of Topic IV-1.A which will be used as input to the integrated review of the Big Rock Point Plant. However, it should be noted that the acceptability of this topic evaluation is contingent upon your agreement to (1) include in a procedure for N-1 loop operation a statement that the bypass and isolation valves in the inactive loop be closed during N-1 operation, (2) physically lock-out power to the inactive pump, and (3) incorporate the MAPLHGR limits for N-1 loop operation in the Technical Specifications.

If there are any errors in the facts of this revised assessment, please supply corrected information and your response with respect to items (1) through (3) above within 30 days of the date you receive this letter.

Sincerely,

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosure:
Revised Assessment for
Topic IV-1.A

cc w/enclosure:
See next page

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Enclosure

SYSTEMATIC EVALUATION PROGRAM

Topic IV-1.A Operation With Less Than All Loops In Service

Plant: Big Rock Point

Discussion

The majority of the presently operating BWRs and PWRs are designed to permit operation with less than full reactor coolant flow. That is, if a PWR reactor coolant pump or a BWR recirculation pump becomes inoperative, the flow provided by the remaining loop or loops is sufficient for steady state operation at some definable power level, usually less than full power.

Plants authorized for long term operation with one reactor coolant pump out of service have submitted, and the staff has approved, the necessary ECCS, steady state, and transient analysis. The remaining PWR and BWR licensees have Technical Specifications which require reactor shutdown within 24 hours if one of the operating loops becomes inoperable and cannot be returned to operation within the time period.

In a letter dated August 17, 1978, Consumers Power Company (the licensee) was sent draft evaluations of eight essentially completed Systematic Evaluation Program (SEP) Topics. We requested that the licensee review and verify that the information was factual and that all documentation cited was current. Topic IV-1.A, Operation

With Less Than All Loops In Service, was one of the eight essentially completed for Big Rock Point. The assessment stated that authorization for N-1 loop operation is provided (Technical Specification 4.1.2(b)); however, there is no supporting ECCS analysis to justify this mode of operation.

By letter dated October 24, 1978, the licensee responded to our August 17, 1978 request and provided comments concerning the correctness of our assessments. With regard to Topic IV-1,A, the licensee took exception to our assessment and concluded that operation with less than all loops in service at Big Rock Point was justified. This conclusion was based on an analysis performed by General Electric Company (GE) in early 1977. The licensee derived and implemented operating limits for Big Rock Point based on this analysis.

The October 24, 1978 letter from the licensee contained three enclosures: (1) Consumers Power Company statement concerning N-1 loop operation, (2) the GE Single Loop LOCA Analysis, and (3) a document entitled Operation of the Big Rock Point Reactor With One Loop Out of Service - Impact on MCHFR Limits. Enclosure 2 contained 3 attachments: (1) the General Electric Analysis, (2) Addendum A General Electric's Answers to Consumers Power's questions, and (3) Addendum B - MAPLHGR Limits for Exxon Fuels.

Evaluation

Several factors have to be considered when evaluating N-1 loop operation: (1) the impact on normal operation (i.e., are there adequate thermal margins when one considers the effect of anticipated transients), (2) the potential effect on accidents which are analyzed (principally the LOCA and locked rotor accident), and (3) the potential for a new accident (in this case, a coldwater accident caused by the startup of the inactive pump).

One factor that can affect all three of these considerations is the effect of one loop operation on reactor coolant flow distribution.

Big Rock Point is a 2 loop, General Electric design, non-jet pump boiling water reactor. The coolant flows through two inlet nozzles (one per loop) which lie 72 degrees apart on the vessel lower head. The flow entering through each nozzle impinges on a diffuser plate (one plate per nozzle). A flow diffuser baffle connected to the core support plate surrounds the fuel channel support tubes and causes the pressure at the inlet to the core support tubes to be relatively uniform. The fact that the vessel entrance region acts as a plenum has been supported by test ("Core Performance and Transient Flow Testing - Big Rock Point Boiling Water Reactor", GEAP-4496, July 1965, USAEC contract AT (04-3)-361). The test showed that the frictional pressure drop between the vessel nozzles and the support tube inlets to be nearly 5 times the velocity head in the support tubes. Instrumented fuel assembly measurements during forced-circulation tests (Figure 4-7 of the above reference) have shown relative assembly power to be insensitive to the number o loops in operation, further indicating that the relative flow to the assemblies is not substantially effected by the number of loops in operation. When considering the high losses due to flow resistance caused by the orifices in the assemblies, a small pressure difference in the lower plenum at the support tube entrance elevation should have a negligible effect on the core flow distribution.

Since the physical arrangement of the forced circulation systems at Big Rock Point, flow through the core, and supportive testing indicate that flow perturbations will not be introduced to the system, it is expected that the reactor will not discern the difference between one pump and two pump operation. The staff, therefore, concludes that uneven or asymmetric flow conditions will have a negligible affect on Big Rock Point during N-1 loop operation.

With regard to the effects of anticipated transients: The licensee has provided (Enclosure 3 to the October 24, 1978 submittal) a discussion on the effects of transients on the minimum critical heat flux ratio (MCHFR) when operating in the reduced flow configuration. The licensee stated that the 3.0 MCHFR limit derived for full flow steady state operation is valid for single loop operation. Our review of the codes used to predict the MCHFR limit support this conclusion. Although the critical heat flux correlation used by the licensee (synthesized ENC Hench-Levy) does not, per se, have a flow term which would directly support the licensee's statement that MCHFR is insensitive to flow changes; it does have a fluid quality factor. Since we have indicated above that the reactor does not see the difference between one or two pump operation, the quality of the fluid does not change; therefore, the computed MCHFRs for two loop operation are bounding for the one loop operation. Furthermore, as discussed in the Cycle 15 reload analysis, a transient MCHFR of 2.15 was established; to this limit was added additional margin to account for the worst case transients. The staff further added conservatism to the limit which yielded the steady state MCHFR of 3.0. The total steady state MCHFR provides assurance that under the worst case transient the resulting MCHFR will not be below the safety limit of 2.15. This margin of safety precludes operation of the reactor in a region conducive to fuel failure.

In addition to the operating restrictions of MCHFR, high neutron flux and high reactor pressure trips are maintained within the same proximity for single loop operation as for two loop operation. That is, the reactor protective system is realigned to cause trips within the same tolerance at the reduced power level as they would at the full power level (e.g., high flux trip - 120% of maximum allowed operating power level, at reduced power, say 50%, in the event of a transient the reactor would trip if power reached or exceeded the 60% power level). The same reduction in setpoint would relate to the overpressure trip.

General Electric (GE) has performed for Big Rock Point an ECCS-LOCA calculation at 102% of rated power, with one loop out-of-service, with the out-of-service loop isolated (pump suction, discharge, and bypass valve closed), and has compared the results to calculations for the two pump in service conditions. The calculations and comparisons performed demonstrate all effects of one loop operation that might significantly affect peak clad temperature (PCT). The analysis of the break spectrum revealed that the worst case break would result from a break in the recirculation discharge line of the inactive loop (0.500 ft^2 break size). This analysis predicts the PCT for this break size to be 2192 degrees F and a peak local oxidation fraction of 0.072. The Appendix K to 10 CFR 50.46 PCT limit is 2200 degrees F and the Big Rock Point oxidation upper limit is 0.17. The calculated PCTs and oxidation fractions for all breaks analyzed for single loop operation are reported in Table 3 of Enclosure 2 to the October 24, 1978 submittal by the licensee.

The analysis has demonstrated that a correction factor must be applied to the all-loops-in-service maximum average planar linear heat generation rate (MAPLHGR) limit for conservative one-loop-out-of-service operation. The MAPLHGR limits for double loop operation were calculated for the design basis accident (DBA) for the cycle 15 reload. New MAPLHGRs were calculated by the licensee considering the worst case break (0.5 ft^2 recirculation discharge line) and compared them with those for the two loop worst case break. The maximum offset in MAPLHGRs was 0.5 kW/ft. The licensee proposed to operate the plant in the N-1 loop configuration with MAPLHGR limits calculated by subtracting 0.5 kW/ft from the double loop MAPLHGR numbers. The analysis yielding approximately a 5% reduction in MAPLHGR applies to the GE fuel. Big Rock Point also employs fuel fabricated by Exxon; however, an analysis for the behavior of the Exxon fuel in single loop operation has not been performed. The licensee states that since MAPLHGR is insensitive to the number of loops in service, since they changed only by at most 5% in single loop for the GE fuel, MAPLHGRs can be conservatively derived by reducing the existing two-loop Exxon MAPLHGR limits by 10%. Based on our review of the methodology employed to calculate the MAPLHGR limits associated with Cycle 15 and our review of the analysis performed for the single loop mode of operation, we conclude that the new limits are conservatively derived and are therefore acceptable. However, since these limits are subject to change with each reload review, we recommend that the single loop

MAPLHGR limits be made part of the Technical Specifications and subsequent changes to them be evaluated by the staff in the same manner as any other change to safety limits.

Regarding the locked roter accident, Big Rock Point has not provided an analysis of the effects that this accident might have when operating in the N-1 loop configuration. However, since operation of the facility with less than all loops in service is a relatively low likelihood event (based on operating experience with several reactors of this design) we conclude that an event such as a locked rotor while in this mode is even more remote. Furthermore, the Systematic Evaluation Program in the course of reviewing design basis events will review the locked rotor accident in both the N loop and N-1 loop conditions. Thus, we conclude that in the interim this deficiency is acceptable for Big Rock Point.

With regard to the potential for a new accident the staff considered the potential for a cold water injection accident caused by the startup of the inactive loop. Staff criteria requires that an analysis of this event be performed to determine the potential consequences. Technical Specifications prohibiting startup of an inactive loop are not considered by the staff to be an acceptable alternative to analyzing the event. However, in lieu of an analysis, reducing the credibility of the event is an acceptable alternative. Methods such as the use of temperature differential interlocks, which prevent opening a valve or starting a pump unless a predetermined minimum temperature differential exists between the active and inactive loop, or requiring the isolation valves to remain open when the pump is inactive, thereby maintaining the idle loop in thermal equilibrium with the operating loop, are examples of effective measures to reduce the likelihood of the event which the staff would review for acceptability.

The analysis performed by the licensee to justify the operating MCFR and the limiting MAPLHGR in the N-1 loop configuration assumes that the inactive loop is completely isolated allowing no bypass flow through the inactive loop. Although operating with the isolation valves open will establish a thermal equilibrium between the loops and resolve the cold water injection accident, the diversion of bypass flow (backflow through the loop) may impact the previously discussed limits in such a way that they can no longer be considered conservative without additional analysis. On this basis the staff cannot permit Big Rock Point to operate with an idle loop (non-isolated) when in the N-1 configuration. Based on the above, we recommend that the licensee establish a procedure that administratively dictates the closing of the isolation and bypass valves in the inactive loop if operation is to continue. Section 12.8 of the Big Rock Point Final Hazards Summary Report presents the cold water accident analysis performed by the licensee. A comparison of this analysis to the methods used in current criteria (Standard Review Plan 15.4.4) reveal some deviations. The SRP method is more detailed than that of the licensee in that it discusses the effects of the cold water accident on operating limits (MCFR), linear heat generation rate (LHGR), and the potential for overpressurization. However, re-analysis of the cold water accident need not be performed if the potential for occurrence is substantially reduced or removed. Therefore, we recommend that the power to the inoperative pump be physically removed (locked out) during N-1 loop operation. This action will provide additional assurance that an inadvertent cold water injection accident will not occur. It should further be noted that this administrative operating restriction may be removed by performing a re-analysis of the cold water accident in accordance with SRP 15.4.4 and the acceptability of that analysis determined by the NRC staff.

We therefore conclude, based on our review of the docketed and submitted material, that operation with less than all loops in service is acceptably resolved. However, it should be noted that the acceptability of this topic (IV-1-A) is contingent upon the licensee's agreement to (1) include in a procedure for N-1 loop operation a statement that the bypass and isolation valves in the inactive loop be closed during N-1 operation, (2) physically lock-out power to the inactive pump, and (3) incorporate the MAPLHGR limits for N-1 loop operation in the Technical Specifications.

ATTACHMENT 2

Consumers Power Company
Big Rock Point Plant
Docket 50-155

10CFR50.59 EVALUATION DATED JUNE 15, 1990

5 Pages

NUCLEAR OPERATIONS DEPARTMENT
SAFETY EVALUATION FORM

Item To be Evaluated: Procedure changes regarding Requirements for one reactor loop power operations.	Item Identification: NUCLEAR STEAM SUPPLY SYSTEM No. SOP-29 Rev 144

I. 10 CFR 50.59 DETERMINATION

1. Is the item safety-related, or can it affect another safety-related item? [X] Yes [ ] No
   List affected item(s). Fuel Thermal Hydraulic Limits
2. Is the item changed from its description in the FSAR/FHSR? [X] Yes [ ] No
   List affected section(s). Section 4.5 Operation with Less Than All Loops In Service (attached)
3. Does the item involve a test or experiment not previously described in FSAR/FHSR? [ ] Yes [X] No
4. Does the item require a change to Technical Specifications? [ ] Yes [X] No
   List affected section(s).

II. 10 CFR 50.59 EVALUATION

1. Is the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report increased? [ ] Yes [X] No

   Basis:

   See basis attached.

2. Is the possibility of an accident or malfunction of a different type than any evaluated previously in the Safety Analysis Report created?

Basis: See basis, attached ☐ Yes ☒ No

3. Is the margin of safety as defined in the basis for any Technical Specification reduced?

Basis: See basis attached ☐ Yes ☒ No

III. NRC NOTIFICATION

1. Should this be included in FSAR/FHSR update?	☒ Yes	☐ No
2. Should item be included in Annual Report to NRC?	☒ Yes	☐ No
3. Is prior NRC approval and an application for amendment to License required?	☐ Yes	☒ No

IV. APPROVALS

Prepared by	Date	Reviewed by	Date
DCMcIntosh	3/14/90	G Burs	6/15/90

4.5 OPERATION WITH LESS THAN ALL LOOPS

Topic IV-1.A of the Systematic Evaluation Program deals with operating the reactor at power with one of the recirculation loops out of service. NRC letter of October 9, 1979 to David Bixel from DLZiemann presents the safety assessment of this topic. The acceptability of operating with one loop out of service was contingent upon satisfying certain conditions. The bypass and isolation valves in the inactive loop must be closed. This requirement was to be controlled by procedure. The power to the recirculating pump in the inactive loop must be physically locked out. MAPLHGR limits for N-1 loop operations must be incorporated into the Technical Specifications.

The proposed Technical Specifications to incorporate MAPLHGR limits for N-1 loop operation was included in Consumers Power Company letter of February 25, 1980.

NRC letter dated June 9, 1981 included a safety evaluation of the revised MAPLHGR limits for Exxon fuel for a one loop operation. At that time evaluation of the one loop MAPLHGR limits for Exxon fuel was not complete. A follow-up letter in response to NRC questions was issued by Consumers Power Company to DMCrutchfield on June 19, 1981. A new MAPLHGR limit for Exxon fuel with a one loop operation was proposed. This change is further documented by letter to DMCrutchfield from GCWithrow on July 22, 1981.

The incorporation of these contingencies pending approval of the Technical Specification change was reported to the NRC by letter dated September 3, 1981 from TCBordine to DMCrutchfield.

Topic IV-1A was acceptably resolved and documented by letter to DPHoffman from DMCrutchfield on October 8, 1981. With the conditions previously mentioned, it is permissible to operate the Big Rock Point reactor with only one recirculation loop in service.

Response to Safety Evaluation Report - Part 2

Question 1

Is the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report increased?

No, the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety has not been increased. Section 4.5 of the Final Hazards Summary Report includes the requirements for plant operations with less than all loops in service. These requirements include:

1) closing the bypass and isolation valves for the idled loop and,
2) the power to the inactive pump must be physically locked out.

These requirements have different bases. The requirement to isolate the inactive loop is based upon not having an acceptable analysis for single loop operation with backflow assuming ECCS conditions. In lieu of performing the analysis MAPLHGR limits were reduced and the idled loop isolated. The power lockout requirement of the idled loop is based upon not having an acceptable analysis for single loop operation with restarting the inactive loop (cold water addition). Complete details are contained in the 10/7/79 safety assessment written by the NRC on SEP Topic IV-4.A "Operation With Less Than All Loops In Service". This assessment was also used in the basis for Technical Specification Amendment No.48 which placed the MAPLHGR limits for single loop operation into the Tech Specs. The NRC stated in their assessment that the administrative limits regarding single loop operation could be removed provided the cold water accident analysis was re-performed in accordance with Standard Review Plan 15.4.4.

The analysis was performed for SEP Topics XV-1, XV-3, XV-4, XV-5, XV-7 and XV-9 and submitted on 7/15/81. Startup of an inactive loop was acceptable and SOP-29 was changed to allow restart providing power had been reduced to a level specified by the reactor engineer. Therefore, with the new analysis, the administrative requirement to lockout power for an idled recirc pump can be removed from SOP-29.

With respect to isolating the idled pump, SOP-29 allows the operator one hour from the time of pump trip until the valves must be closed. Closing of the suction valve to the idled pump is not a prudent action from a reliability standpoint. If the operators are unable to start the inactive pump within the one hour limit, closing the suction valve will cause the pump shaft seals to begin to depressurize. If at some point after the one hour expires and pump restart is performed the pressure transient on the pump seals may cause them to seal improperly causing potential shaft seal leakage which may lead to pump isolation and unit derate. By allowing the operators to keep the suction valve open, pressure is maintained on the seals, thus when the pump is restarted less of a transient is experienced by the seals. Reverse flow is prevented by having the bypass and discharge valves closed.

Therefore, by eliminating the requirements to 1) lockout power to the idled recirc pump and 2) closing the suction valve of the idled pump during one loop operation, the probability of an accident or malfunction previously evaluated in the Safety Analysis Report are not increased.

Question 2

Is the possibility of an accident or malfunction of a different type than evaluated previously in the Safety Analysis Report created?

No, changing the requirements on the system configuration during single loop operation does not create a new possibility of accident or malfunction of a different type than previously evaluated in the Safety Analysis Report. The changes (not locking power out and suction valve open) do not place the plant in an un-analyzed configuration. Previous single loop analysis assumed an isolated inactive loop, no change is being made to this requirement, the idled recirc pump discharge valves still must be closed within one hour. Power lockout was based solely on not having an analysis which has subsequently been performed showing the acceptability of restarting an inactive loop. In addition power level requirements are currently in place in SOP-29 prior to restarting an inactive loop.

Question 3

Is the margin of safety as defined in any Technical Specification reduced?

No, the basis for Tech Spec change No.48 refers to the administrative requirements imposed in the SEP Topic IV-4.A safety assessment (refer to the 10/9/79 NRC to CPCo letter). This assessment allowed for changing the administrative limits should the proper re-analysis be performed. The analysis was subsequently performed and restart of an inactive loop was found acceptable provided the power level was reduced. Administrative limits currently in SOP-29 require the Reactor engineer to provide an acceptable power level prior to pump restart. In addition the requirement to isolate an inactive loop within one hour is maintained as the requirement to close the discharge valves is still imposed, only the suction valve is to allowed to remain open.

ATTACHMENT 3

Consumers Power Company
Big Rock Point Plant
Docket 50-155

NRC LETTER DATED APRIL 7, 1982

6 Pages

OC0990-0074-NL04

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SYSTEMATIC EVALUATION PROGRAM

TOPIC XV-9

BIG ROCK POINT

TOPIC: XV-9, Startup of an Inactive Loop

I. INTRODUCTION

The startup of an inactive or idle recirculation loop at an incorrect temperature is examined to assure that the consequences are acceptable. The guidance for the review of this topic is provided by SRP Sections 15.4.4 and 15.4.5. This transient is evaluated because it reduces voids in the core, which causes an increase in power and reduces thermal margins. The calculated Minimum Critical Power Ratio (MCPR) is compared to the MCPR safety limit to demonstrate that fuel failures will not occur.

II. REVIEW CRITERIA

Section 50.34 of 10 CFR Part 50 requires that each applicant for a construction permit or operating license provide an analysis and evaluation of the design and performance of structures, systems, and components of the facility with the objective of assessing the risk to public health and safety resulting from operation of the facility, including determination of the margins of safety during normal operatins and transients conditions anticipated during the life of the facility.

Section 50.36 of 10 CFR Part 50 requires the Technical Specifications to include safety limits which protect the integrity of the physical barriers which guard against the uncontrolled release of radioactivity.

The General Design Criteria (Appendix A to 10 CFR Part 50) set forth the criteria for the design of water-cooled reactors.

GDC 10 "Reactor Design" requires that the core and associated cooling, control and protection system be designed with appropriate margin to assure that specified acceptable fuel design limits are not exceeded during normal operation, including the effects of anticipated operational occurrences.

GDC 15 "Reactor Coolant System Design" requires that the reactor coolant and associated protection systems be designed with sufficient margin to assure that the design conditions of the reactor coolant pressure boundary are not exceeded during normal operation, including the effects of anticipated operational occurrences.

GDC 20 "Protection System Functions" requires that the protection system be designed to initiate automatically the operation of reactivity control systems to assure that specified acceptable fuel design limits are not exceeded as a result of anticipated operational occurrences.

GDC 26 "Reactivity Control System Redundancy and Capability" requires that the reactivity control system be capable of reliably controlling reactivity changes to assure that under conditions of normal operation, including anticipated operational occurrences, and with appropriate margin for malfunctions such as stuck rods, specified acceptable fuel design limits are not exceeded.

GDC 28 "Reactivity Limits" requires that the reactivity control systems be designed with appropriate limits on the potential amount and rate of reactivity increase to ensure that the effects of postulated reactivity accidents can neither (1) result in damage to the reactor coolant pressure boundary greater than limited local yielding nor (2) sufficiently disturb the core, its support structures or other reactor pressure vessel internals to impair significantly the capability to cool the core.

III. RELATED SAFETY TOPICS

Various other SEP topics evaluate such items as the reactor protection system. The effects of single failure on safe shutdown capability are considered under Topic VII-3.

IV. REVIEW GUIDELINES

The review is conducted in accordance with SRP 15.4.4 and 15.4.5.

The evaluation includes review of the analysis for the event and identification of the features in the plant that mitigate the consequences of the event as well as the ability of these systems to function as required. The extent to which operator action is required is also evaluated. Deviations from the criteria specified in the Standard Review Plan are identified.

V. EVALUATION

The startup of an inactive loop was analyzed for the Big Rock Point plant in the reference below. The more important assumptions were:

1. Initial power is 102% of single loop operating power (188.7 MWt)
2. Water in isolated loop is at 100°F
3. Scram initiated on high flux (120% power)
4. Conservative reactivity coefficients and scram characteristics are used
5. Pump in idle loop reaches full speed in 1 second, bypass and discharge valves open sequentially at design speed

These and other assumptions described in the reference are in accordance with the SRP. The computer code used in the analysis is COBRA-IV-I modified for the Big Rock Point plant.

The results of the analysis show that the peak power of 127% initial value is reached 8.25 seconds after initiation of the event. The reactor pressure rises 19 psi to 1349 psia, which is below design pressure. The minimum critical power ratio was 1.77, which is above the specified acceptable fuel design limit for MCPR (1.32).

VI. CONCLUSION

The staff has reviewed the Big Rock Point submittal on SEP Topic XV-9, Startup of an Inactive Loop. The assumptions used in the analysis are in conformance with SRP Sections 15.4.4 and 15.4.5 and the results satisfy the SRP acceptance criteria and are therefore, acceptable.

REFERENCE: Letter to D. Crutchfield, NRC from R. A. Vincent, Consumers Power Company, "Big Rock Point Plant - SEP Design Basis Event Topics", dated July 15, 1981.

ATTACHMENT 4

Consumers Power Company
Big Rock Point Plant
Docket 50-155

AMENDMENT NO.48 TO FACILITY OPERATING
LICENSE NO. DPR-6 FOR BIG ROCK POINT
DATED SEPTEMBER 21, 1981

14 Pages

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

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Approval of Amend. 48 to License on Recirc pump sing. loop operation

September 21, 1981

Docket No. 50-155
LS05-81-09-059

Mr. David P. Hoffman
Nuclear Licensing Administrator
Consumers Power Company
1945 W. Parnall Road
Jackson, Michigan 49201

Dear Mr. Hoffman:

SUBJECT: OPERATION WITH ONE RECIRCULATION LOOP OUT OF SERVICE

Re: Big Rock Point Plant

The Commission has issued the enclosed Amendment No. 48 to Facility Operating License No. DPR-6 for the Big Rock Point Plant. This amendment consists of changes to the Technical Specifications in response to your application dated February 25, 1980 and supplements thereto dated June 19, 1981, July 22, 1981, and September 3, 1981.

The amendment authorizes operation of the reactor with one recirculation loop out of service.

Copies of our related Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Dennis M. Crutchfield

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

Enclosures:
1. Amendment No. 48 to License No. DPR-6
2. Safety Evaluation
3. Notice of Issuance

cc w/enclosures:
See next page

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cc
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CONSUMERS POWER COMPANY

DOCKET NO. 50-155

BIG ROCK POINT NUCLEAR PLANT

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 48
License No. DPR-6

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

   A. The application for amendment by Consumers Power Company (the licensee) dated February 25, 1980, as supplemented June 19, July 22 and September 3, 1981, complies 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 facility will operate in conformity with the application, 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 safeth 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.

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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 hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 48, 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: September 21, 1981

ATTACHMENT TO LICENSE AMENDMENT NO. 48

FACILITY OPERATING LICENSE NO. DPR-6

DOCKET NO. 50-155

Revise Appendix A Technical Specifications by removing the following page and inserting the enclosed page. This revised page includes the captioned amendment number and contains a vertical line indicating the area of change.

PAGE

5-9a

TABLE 1

	Reloads: F & Modified F	Reload G	Reload G-1U	Reload G-3/G-4
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-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
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 Ration (MCHFR), 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.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 48 TO FACILITY OPERATING LICENSE NO. DPR-6

CONSUMERS POWER COMPANY

BIG ROCK POINT PLANT

DOCKET NO. 50-155

1.0 INTRODUCTION

By letter dated February 25, 1980 and supplements dated June 19, 1981, July 22, 1981, and September 3, 1981, Consumers Power Company (the licensee) requested an amendment to Facility Operating License No. DPR-6 for the Big Rock Poing Plant. This amendment would add reactor operating limits for operation with one recirculation loop out of service. In a related action, the Commission issued Amendment No. 44 on June 9, 1981 which authorized a change in the reactor operating limits for operation with both recirculation loops in service.

2.0 BACKGROUND

The licensee's February 25, 1980 submittal requested (1) revised reactor operating limits for operation with both recirculation loops in service, and (2) operating limits for one recirculation loop out of service for the Big Rock Point Plant. Amendment No. 44 dated June 9, 1981 approved new ECCS operating limits for two loop operation.

By letters dated June 19, 1981 and July 22, 1981, Consumers Power Company provided additional information with regard to single loop operation. This safety evaluation addresses operation with one recirculation loop in service.

3.0 DISCUSSION AND EVALUATION

The NRC staff has completed a review of the March 31, 1977, General Electric (GE) Big Rock Point ECCS analysis (reference 1) for single loop operation. Although single loop operation (SLO) is still an outstanding issue in jet pump BWRs, Big Rock Point is a non-jet pump BWR and has sufficiently demonstrated that this mode of operation is safe and acceptable.

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The GE analysis is performed with the same codes as used in the two loop case. Certain input parameters are modified to simulate single loop operation. In the analysis, the location of the worst break was determined to be in the 20-inch recirculation discharge of the isolated loop. Further analyses of a break spectrum was made for the worst break location. A 0.5 ft^2 break was determined to result in the highest calculated peak cladding temperature of 2192°F with a local oxidation of 7.2%. A detailed description of the codes used can be found in NEDE-20566, "General Electric Company Analytical Model for Loss-of-Coolant Analysis in Accordance with 10 CFR 50 Appendix K."

Although this would appear to satisfy requirements, Big Rock Point is in a unique situation with respect to the simultaneous or exclusive use of EXXON or GE fuel during a particular cycle. The use of a GE two loop LOCA analysis to describe the behavior of an all EXXON core in a transient situation was a previous staff concern and was reviewed and found acceptable in reference 2. A further complication arises by the fact that there was no SLO ECCS performed for EXXON fuel. Rather than performing EXXON SLO ECCS, the licensee proposed to place a 20% MAPLHGR reduction (reference 3) on the EXXON fuel whereas GE fuel has only a 5% MAPLHGR reduction. The 5% MAPLHGR reduction for GE fuel is a result of faster core uncovery in SLO. The staff has concluded that the 20% reduction for EXXON fuel is an adequate margin such that in the event of a LOCA, the fuel would not be expected to violate 10 CFR 50.46 limits. In reference 4, the licensee stated that an Exxon analysis would probably predict a similar response in the most limiting break situation during single loop operation. This can be based on the similarity between EXXON and GE two loop analysis response times for the 0.5 ft^2 most limiting break size. A comparison of the analysis response can be found in Table 1 (copy attached).

Supporting analysis for safe operation in the SLO mode can be found in the Systematic Evaluation Program (SEP) review (reference 5). Although this review did not address the acceptability of the GE ECCS analysis, which prompted this review, several other areas of concern were addressed. A major point was that SLO will not affect the reactor coolant flow distribution. The coolant flows through two inlet nozzles (one per loop) which are 72 degrees apart on the vessel lower head. The flow entering through each nozzle impinges on a diffuser plate (one plant per nozzle). A flow diffuser baffle connected to the core support plate surrounds the fuel channel support tubes and causes the pressure at the inlet to the core support tubes to be relatively uniform. The fact that the vessel entrance region acts as a plenum has been supported by test ("Core Performance and Transient Flow Testing - Big Rock Point Boiling Water Reactor," GEAP-4496, July 1965, USAEC contract AT (04-3)-361). The test showed that the frictional pressure drop between the vessel nozzles and the support tube inlets to be nearly 5 times the velocity head in the support tubes. Instrumented fuel assembly measurements during forced-circulation tests (Figure 4-7 of the above reference) have shown relative assembly power to be insensitive to the number of loops in operation, further indicating that the relative flow to the assemblies is not substantially affected by the number of loops in operation. When considering the high losses due to flow resistance caused by the orifices in the assemblies, a small pressure difference in the lower plenum at the support tube entrance elevation should have a negligible effect on the core flow distribution.

The SEP review also concluded that the acceptability of the SLO is contingent upon the licensee's agreement to (1) include in a procedure for N-1 loop operation a statement that the bypass and isolation valves in the inactive loop be closed during N-1 operation, (2) physically lock-out power to the inactive pump, and (3) incorporate the MAPLHGR limits for N-1 loop operation in the Technical Specifications. By letter dated September 3, 1981, the licensee stated that items (1) and (2), above, have been included in the plant procedures. This amendment would incorporate MAPLHGR limits for SLO into the Technical Specifications.

Based on the references presented, the NRC staff agrees with the licensee that the GE SLO ECCS analysis will adequately represent the behavior of EXXON fuel elements in a LOCA situation. Based on the supportive analyses, the NRC staff concludes that SLO is a safe means of reactor operation.

4.0 ENVIRONMENTAL CONSIDERATION

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

5.0 CONCLUSIONS

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

6.0 REFERENCES

(1) "Big Rock Point Single-Loop Operation Loss-of-Coolant Accident Analysis for General Electric Fuel in Conformance with 10 CFR 50 Appendix K (Non-Jet Pump Boiling Water Reactor)," March 31, 1977.

(2) Safety Evaluation Report from D. Zieman (NRC) to Consumers Power, June 4, 1976.

(3) Letter from G. C. Winthrow (Consumers Power) to D. Crutchfield, July 22, 1981.

(4) Letter from G. C. Winthrow (Consumers Power) to D. Crutchfield, June 19, 1981.

(5) Letter from D. Ziemann (NRC) to D. Bixel (Consumers Power), August 9, 1979.

Dated: September 21, 1981

Attached: Table 1

TABLE 1

COMPARISON OF KEY SYSTEM RESPONSE PARAMETERS AS PREDICTED BY EXXON AND GE APPENDIX K LOCA MODELS

Cold Leg Break Size (Ft^2)	Time (Sec) Low Reactor Pressure (200 Psig) Exxon (2 Loops)	GE (2 Loops)	GE (1 Loop)	Rated Core Spray Flow Exxon (2 Loops)	GE (2 Loops)	GE (1 Loop)	Core Midplane Uncovery Exxon (2 Loops)	GE (2 Loops)	GE (1 Loop)
3.926	-	5.	-	-	20.4	-	-	7.1	-
3.53	11.9	-	-	26.9	-	-	*	-	-
1.6	-	-	12.	-	-	27.7	-	-	4.0
1.0	20.4	21.5	20.	35.4	36.9	35.4	*	17.2	6.1
.50	36.3	40.	40.	51.3	55.1	64.	*	20.6	10.8
.375	46.5	-	54.	61.5	-	71.	~23.(1)	-	13.6
.25	67.9	75.	75.	85.6	98.2	103.	*	29.9	19.4
.10	161.(2)	140.	140.	189.	156.	158.	*	62.2	48.9

\- implies case not analyzed.

* implies information not readily available.

(1)Time at which quality at hot node (core midplane) goes to 1.0.

(2)Break smaller than about 0.1 ft^2 result in ADS actuation. ADS blowdown is prolonged in the Exxon analyses versus the GE analyses because Exxon assumed only three operable blowdown paths whereas GE assumed four.

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-155

CONSUMERS POWER COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO FACILITY OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 48 to Facility Operating Licensing No. DPR-6, issued to the Consumers Power Company (the licensee), which revised the Technical Specifications for operation of the Big Rock Point Plant (the facility) located in Charlevoix County, Michigan. The amendment is effective as of its date of issuance.

The amendment authorizes operation of the reactor with one recirculation loop out of service.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR 51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with issuance of this amendment.

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For further details with respect to this action, see (1) the application for amendment dated February 25, 1980 and its supplements dated June 19, 1981, July 22, 1981, and September 3, 1981, (2) Amendment No. 48 to License No. DPR-6, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. and at the Charlevoix Public Library, 107 Clinton Street, Charlevoix, Michigan 49720.

A copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Licensing.

Dated at Bethesda, Maryland, this 21st day of September, 1981.

FOR THE NUCLEAR REGULATORY COMMISSION

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