SAFETY EVALUATION

OFFICE OF NUCLEAR REACTOR REGULATION

BEAVER VALLEY UNIT 1

OPERATION WITH TWO OUT OF THREE REACTOR COOLANT LOOPS*

July 20, 1984

*This Safety Evaluation Report summarizes results of the staff's review work performed in the last several years. The Unit may operate with two out of three coolant loops only when an Amendment to the Operating License has been issued by the staff authorizing such operation.

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I. Introduction

Duquesne Light Company (the licensee, DLC) has applied for approval to operate Beaver Valley Power Station Unit 1 (a three-loop Westinghouse plant) with only two active coolant loops (Reference 1). Two-loop operation is currently precluded by a license condition. Beaver Valley is one of a class of plants with loop isolation valves in both the cold leg and hot leg of each loop. The licensee proposes to operate with the isolation valves closed in the inactive loop. The purpose of the application is to allow the plant to continue operating with one loop out of service in the event of an equipment failure in that loop.

Our review has concentrated on the impact of this proposed change on the performance of safety related systems. The study included a complete review of design basis accidents, as well as an evaluation of the impact on generic issues.

II. Proposed Mode of Operation

A. Primary Coolant System and Reactor

The inactive loop will be isolated from the primary coolant system by closing loop isolation valves in both the hot and cold legs. Following closure, motive power to the valves will be removed by locking the associated circuit breakers in the open position, in conformance with technical specifications. In addition, interlocks are provided which prevent inadvertent opening of these valves (FSAR 14.1.6). The location of the isolation valves are such that the Pressurizer, charging and letdown, ECC and RHR are still open to the reactor coolant system (RCS).

A water relief valve will be installed in the isolated loop to prevent overpressurization. All high pressure interfaces with the operating loops, all drain paths and all interfaces with injection systems will be isolated.

Exceptions to these operating restrictions will be allowed only during cold shutdown or refueling, during which time maintenance and repair of the loop will be carried out. Switchover between N loop and N-1 loop operation will be performed only at cold shutdown.

Operation of the reactor would be limited by technical specifications to 65 percent of full power. The total allowable peaking factor $F_Q(Z)$ at 65% power would decrease from 3.57 to 2.77 and there would be minor changes in the normalization curve, K(Z). These changes result in an increase in the DNBR during operation from about 1.7 to about 2.3.

There will be no changes to the technical specification values of delta T, core average temperature or loop flow, although the actual loop flow would increase somewhat.

Because of the results of main steamline break calculations described below, the required shutdown margin maintained during N-1 loop operation will be 2.44% delta K/K instead of the 1.77% delta K/K maintained during N loop operation.

We consider the above status of the primary system acceptable.

B. Secondary System

The licensee indicated that the secondary side of the isolated loop would be kept filled and the steam generator would be maintained in a wet layup condition as determined by secondary side chemistry requirements. The loop would be isolated from the main and auxiliary feedwater system and the main steam system by closing the appropriate valves. Steam supply from the isolated loop to the turbine-driven auxiliary feedwater pump would also be isolated. Pressure and temperature of the isolated loop and the corresponding steam generator would be maintained within the constraints imposed by brittle fracture considerations, and steam generator tube differential pressure limits.

A small amount of steam leakage on the secondary side of the operating loops to the isolated loop is possible through a 3" check valve on the residual heat removal line. Pressure monitoring of the steam generator would remain available and excessive in-leakage would be vented by manual or automatic opening of the atmospheric dump valve. In addition, steam flow through the check valve could never be more than that through a stuck open atmospheric dump valve, which is bounded by the analysis of a stuck open safety valve performed for the N-1 case. Steam generator level will be maintained above the top of the SG tubes and within the narrow range level indication. No maintenance on the isolated loop portions would be performed during N-1 loop operation because the 3-inch check valve is not considered sufficient protection of personnel from possible steam hazards.

The isolation valves to the down loop would be reopened for three loop operation only during cold shutdown using approved start-up procedures to prevent excessive thermal and hydraulic stresses. We consider the above status of the isolated loop acceptable.

Maximum steam flow and pressure rating at 65 percent power during N-1 loop operation are slightly less than the steam flow and pressure rating at 100 percent power during 3-loop operation. Therefore, the steam safety valve capacity on the operating steam generators is adequate to remove the maximum calculated steam flow at the engineered safeguards design rating from the steam generator. There is also no change in the operating conditions for the main steam and feedwater isolation valves.

C. Instrumentation and Control System

The reactor trip system and engineered safety feature actuation system initiate protective action based on measurements of primary and secondary coolant system parameters, as well as other plant conditions. The parameters and conditions associated with an out-of-service loop and the associated protective actions are identified in Table 1. The following discussion includes those aspects of the protection systems which are unique to plant operation with a loop out of service:

1. Primary Coolant Temperature

   The overpower and overtemperature ΔT are based in part on a measurement of primary coolant hot and cold leg temperatures. Each loop provides one channel of input signals for the 2-out-of-3 logic to initiate a reactor trip. During N-1 operation the channel associated with the out-of-service loop is placed in trip and the logic operates on the basis of 1-out-of-2 with input signals from the two operating loops. In addition, the setpoint for the overtemperature ΔT trip channel function associated with the operating loops is readjusted corresponding to the value established for N-1 operation.

   In addition to the above reactor trip functions, low average reactor coolant temperature is for two other protective actions. Feedwater isolation is initiated on low T-avg and reactor trip (P-4). On low low T-avg (P-12) steam dump is terminated and a permissive is provided to reopen the cooldown condenser dump valves. These functions also operate based on 2-out-of-3 logic, with one channel of the average temperature signals being provided by hot and cold leg temperature measurements. The average temperature associated with an out of service loop will be low such that the logic for these functions will be 1-out-of-2 based on input signals from the two operating loops.

2. Primary Coolant Flow

   The loss of flow reactor trip is interlocked with permissives based on reactor power such that above 10% power (P-7), a trip is initiated on loss of flow in any two loops and above 31% power (P-8), a trip is initiated on loss of flow in any loop. During N-1 operation the flow channels associated with the out-of-service loop will be in a tripped state. The licensee has proposed to increase the P-7 interlock setpoint to 71% power such that the trip on loss of flow in a single loop becomes a high reactor power trip. That is, a high reactor power will provide the P-7 permissive such that loss of flow trip will be initiated since flow channels for these out-of-service loop are in a tripped state.

   While we do not object to raising the setpoint of P-7 to defeating the single loop loss of flow trip, we find the use of the P-7 permissive to provide an overpower trip to be unacceptable. The annunciation associated with a trip based on P-7 would be an indication of loss of flow rather than high reactor power. Therefore, we require that the setpoint of the power range neutron flux channels be reduced to 71% and that P-7 be increased to a value which would not provide misleading information to the plant operator. Further, operator training should emphasize that the two-loop loss-of-flow trip is indicative of a loss of flow in either operating loop during N-1 operation.

The trip on loss of flow in two loops is initiated on sensing either low primary coolant loop flow or by contacts indicating that the reactor coolant pump breaker is tripped. During N-1 operation the channel associated with the latter will also be in a tripped state for the out-of-service loop.

3. Steam Generator Level

The logic for the low-low steam generator level reactor trip and initiation of auxiliary feedwater is interlocked to block the trip when the RCS isolation valves associated with the out-of-service loop are closed.

The steam/feedwater mismatch channels associated with the out-of-service loop will be in an untripped state. Therefore, the reactor trip which is initiated on coincident low level will not occur regardless of steam generator level in the out-of-service loop.

The licensee has proposed to bypass the hi-hi steam generator level channels associated with the out-of-service loop to preclude the potential for an inadvertent turbine trip and isolation of feedwater to the two operating loops. We find this acceptable since high-high steam generator level on the out-of-service loop does not provide a required safety action and could only result in an unnecessary challenge of plant safety systems. We will review the manner in which channels are bypassed during the technical specification review for N-1 operation.

4. RCP Bus Undervoltage and Underfrequency

The buses used to supply power to the reactor coolant pumps will remain energized during N-1 operation. A reactor trip occurs on the loss of power to any two of the three busses. Since the protection provided by these channels is primarily for the detection of degraded conditions on the electrical power grid, we find the fact that one of these busses not supplying power to an RCP during N-1 operation of no safety significance.

5. Steam Generator Pressure

The logic for the low steam generator pressure initiation of safety injection is interlocked to block this trip when the isolation valves associated with the out-of-service loop are closed.

In Summary the following actions are taken with regard to the protection systems when operating with a loop out of service.

1. Reduction of the power range neutron flux trip setpoint to 71%.

2. Readjustment of overtemperature ΔT trip setpoint for N-1 operation

3. Placing the channels of the overtemperature and overpower ΔT trips for the out-of-service loop in trip.

4. Increasing the P-8 interlock set point to preclude a reactor trip on low flow for the out of service loop.

5. Bypassing the hi-hi steam generator level channels to preclude the potential for inadvertent turbine trip and feedwater isolation.

These actions will be confirmed by appropriate notation to the limiting conditions of operation incorporated in the plant Technical Specifications.

The effect of N-1 operation with respect to the protection systems are the following:

1. The logic for the overtemperature and overpower ΔT trips is reduced from 2-out-of-3 to 1-out-of-2.

2. The logic for feedwater isolation following reactor trip is reduced from 2-out-of-3 to 1-out-of-2 coincident low T-avg.

3. The logic for blocking steam dump to the condenser is reduced from 2-out-of-3 to 1-out-of-2 on low-low T-avg.

4. The logic for reactor trip on loss of flow in a single loop when operating above 31% power (P-8) is changed to a loss of flow in a single operating loop when operating above 10% power (P-7).

Since these changes do not violate the single failure criterion, we find that requirements for redundancy to initiate safety actions is maintained and operation with a loop out of service is, therefore, acceptable.

D. Loop Isolation Valves and Loop Support

The Unit employs two motor-operated stop valves in each of the three reactor coolant loops. The sizes of those valves are 29" in hot leg and 27½" in cold leg. They are double-disc construction and each is operated together with a by-pass line and a motor-operated by-pass valve. Electric interlock circuits will permit the operation of pump and valves in the loop according to acceptable patterns. A comparison of the construction parameters for the stop valves and the reactor coolant loop piping is listed as follows:

	Stop Valves	Piping
Material	ASTM A351 Grade CF8M	ASTM A351 Grade CF8M
Design Temperature °F	650	650

	Stop Valves	Piping
Design/Working Pressure, psig	2485/2235	2485/2235
Shop/Pre-operational test pressure, psig	3350/3107	NA/3107
Code/Class	ASME III, 68/A	ANSI B31.1, 67/NA

The construction parameters for the loop piping and stop valves are similar. The valves were constructed according to the ASME B&PV Code Section III, 1968 Class A requirements while piping was constructed according to ANSI B31.1, 1967 Standard. The ASME Code requires more rigorous quality assurance than ANSI B31.1. In addition, a shop hydraulic test of 3,350 psig pressure has been performed on the valves. This was not a part of the requirements on piping.

The ASME B&PV Code Section III always considers that the piping system, not the valve body, is limiting. This is because the design and fabrication requirements of the valves result in a section modulus greater than that of the piping. We concur with the licensee's assessment.

Normal flow rate per loop at the N-1 (2-loop) operation will be 36.1 X 10^6 lbm /hr. compared to the normal flow rate per loop of 33.6 X 10^6 lbm /hr. at normal (3-loop) operation. Since this increase in flow rate is less than 8%, and since the steady state flow rate contributes only a small part of the total load on reactor coolant loop supports, these supports will provide adequate resistance to the additional loading caused by the N-1 operation flow.

E. Miscellaneous Systems

Loop isolation will indirectly affect the operation of the turbine-driven auxiliary feedwater pump and the pressurizer sprays.

The turbine-driven auxiliary feedwater (AFW) pump which can receive steam from all three loops will receive steam from either of the two operating loops during N-1 loop operation. Thus there is still redundancy in steam sources for the AFW pump turbine and, in addition, the two motor driven AFW pumps, powered from two separate class 1 power sources would be available. Therefore, the reliability of the AFW system is acceptable for N-1 loop operation. There is no change in the design flow and pressure required of the AFWS or other safety-related auxiliary cooling water systems.

Water is supplied to the pressurizer sprays from the cold legs of two of the three loops. If one of those loops is inoperable, flow from the other would still be available. Although the standard technical specifications require that the pressurizer sprays be operable, there is no requirement for redundancy. Therefore, operation of the pressurizer sprays in this manner is acceptable.

F. Initial Test Program And Procedures

We have reviewed the Unit 1 Cycle 1 Startup Report to determine if adequate testing was conducted during the initial startup to support two-loop operation. Our review indicates that reactor coolant system flow rate and flow coastdown measurements were conducted for the two-loop configuration, that acceptance criteria were met, and that no modifications were made to the reactor coolant systems that would invalidate the data. Since adequate resistance temperature detector (RTD) bypass loop flow is necessary to assure adequate RTD response time for coolant temperature input to the reactor protection system, we also verified that the RDT bypass loop flow will not be adversely affected with one loop out of service.

We conclude that, other than the surveillance tests required by Technical Specifications, no additional preoperational or initial operation tests are necessary prior to two-loop operation with the third loop isolated.

We have reviewed licensee submittals including responses to our requests for additional information regarding two-loop operations. In Reference 13 the licensee stated:

"To support two-loop operation the following steps would be taken:

1. The instrumentation, alarms, bistables and valve positions for the out-of-service loop would be made identifiable to the operator and administratively controlled to avoid confusion during an event. These items would be a part of the procedure for removing the loop from service.

2. The auxiliary feedwater flow to the out-of-service loop would be isolated as part of the procedure for removing a loop from service. Flow verification could not be made and, therefore, would require identification of this instrument as being out of service for the affected loop.

3. The surveillance tests would provide for monitoring of the following where necessary:

   - verification of the closed position of the out-of-service main steam isolation valve

   - instrument channel checks for protection and control instrumentation

   - auxiliary feedwater system alignments."

We agree with the stated need for procedures to control actions to remove a loop from service and to specify surveillance tests as indicated. The licensee should develop procedures for changing from three-loop to two-loop operation and vice versa. The needed surveillance tests and monitoring checks for two-loop operation should be specified and controlled by written procedures. All procedures related to two-loop operation and surveillance, including revised or new emergency operating procedures, should be uniquely identified and placed at locations convenient to where they will be used.

G. Control Room Indicators - Human Factors Evaluation

During a visit to the Beaver Valley control room on August 19, 1982, we reviewed some of the instrumentation that would be affected. By letter dated September 2, 1982 Duquesne Light Company supplied us with a list of affected instrumentation and an estimate of the number of affected annunciators and bistable lights.

The list provided by Duquesne Light Company includes 30 instruments, most of which will read downscale while the remaining instrument indications will depend on the nature of the work being conducted in the isolated loop. Five recorders will have one of three pens each reading downscale and one recorder will have all three pens reading low. The number of off normal indications included in instruments and recorders is large enough to be beyond the normal memory capability of an operator. A unique identifier should be provided prominently on each display to remind the operator that the indication refers to the isolated loop. This N-1 loop identifier should not be part of the normal maintenance tag-out system unless the indicator is, in fact, inoperative because of a malfunction, calibration or test.

Since certain indications can have zero as a legitimate value during this mode of operation, it is also imperative that the affected instruments fail off-scale, and not at zero.

A more subtle, but no less important human factors problem may exist with displays in the operating loops. If system operating ranges change because of the N-1 loop operation, such that normal operating zones on meters are no longer applicable, or values are different from what appear in procedures, the operator may be presented with conflicting information. The licensee should determine if this condition does exist, what its magnitude might be, and how it will be resolved (e.g. simulation, training, procedure modification, unique display identifiers).

Two different conditions exist for annunciators. In the first case, approximately 27 annunciators share inputs from all three loops. The input from the isolated loop would have to be defeated to maintain the operability of the alarm function. A rigorous administrative control procedure will be necessary to account for the defeated signals and to ensure return to normal when the N-1 loop operation is terminated. Analysis will be required on each annunciator in this group to determine if there will be any change in the operator response procedure based on one inoperative loop. If any change in procedure is required, the annunciator will require a unique identifier to remind the operator of the unorthodox mode. If no change in response procedure is required, there appears to be no need for unique identifiers on the annunciator tiles.

In the second case, up to 12 annunciators that provide status monitoring on loop components would have to be identified with out-of-service tags because they are associated with the isolated loop. The specific annunciators affected would depend on the system configuration and the nature of the work being performed on the isolated loop. Unless specific response procedures are different for those not tagged out, there appears to be no requirement for unique identifiers.

In addition to annunciators, approximately 36 protection system bistable status lights could be affected, depending on the type of work to be performed on the isolated loop. To maintain consistency in uniquely identifying isolated loop indications, and to reduce the risk that, after testing, bypassed signals in the isolated loop protection system are returned to the tripped condition, unique identifiers should be installed on all isolated loop bistable status lights.

Based on our review of the instrumentation, annunciators, and protection system bistable status lights affected by the N-1 loop operation, we conclude that all instruments, recorders, annunciators and status lights providing legitimate, though abnormal, status indications in the isolated loop should be provided with unique and prominent identifiers to remind the operator that the information being presented is not that of a normal operating loop. The unique identifier should not be part of the normal tag-out system unless the system or component is actually out of service.

An analysis should be conducted to identify all other indications which, though normal for N-1 loop operations, will not remain within the full-loop normal zones or will be different from values specified in procedures for full-loop operation. This effort should also include recommendations on how these mode-dependent indications will be brought to the attention of the operators and how they will know what limiting values are actually in effect. Operating Procedures, specific to N-1 loop operations should be developed and operator training, preferably through simulation, needs to be conducted.

Finally, administrative controls need to be put in place to ensure that maintenance personnel are alerted to, and aware of, any limitations to perform routine checks and tests that could result in unit trips or unavailability of ESF systems.

H. Conclusions

Operation of the plant with one loop isolated in the manner described above provides adequate protection of the primary coolant boundary and does not significantly degrade the performance of safety related systems. We therefore conclude that N-1 loop operation as described above will not significantly increase the occurrence rate of accidents and transients.

III. Safety Issues

In the course of the review, several possible safety issues were considered. A few of them were judged to be of potential significance and were examined in detail. Results of those evaluations are presented below.

A. Core Thermal Hydraulics

Our review of the Beaver Valley Power Station (BVPS) Unit 1 thermal-hydraulic design included concerns about the effect of changing from three loop operation (N) to two loop operation (N-1) relative to: (a) thermal-hydraulic parameters, (b) inlet flow maldistribution, and (c) flow instability. The licensee's response to our questions was given in Reference 18.

Thermal-Hydraulic Parameters

The licensee's response provided Table 2, a thermal hydraulic comparison for three- and two-loop operation which is also included with this evaluation. In examining Table 2, it is noted that when operating with one loop isolated the values for the system pressure, the percent heat generated in the fuel and the affected flow area for heat transfer remain constant. However, the values for the reactor core heat output, coolant flow, coolant temperatures, average temperature rise in the core, heat flux, average linear power and core pressure drop are reduced. Also, when operating with one loop isolated, the minimum DNBR at nominal design conditions is increased, which is conservative.

The licensee responded to our question on the possibility of temperature differences of few degrees in the active cold legs due to the isolation of one loop causing the possibility of a radial power tilt and increase in the enthalpy rise factor. The licensee stated that the quadrant power tilt has a restriction of 2 percent as stated in the Beaver Valley Technical Specification (Section 3.2.4) and exceeding this value would require the necessary actions described in the Technical Specifications. Therefore there is no difference in the thermal hydraulic methodology in the evaluation of the N and N-1 loop operation, and the safety analysis for N-1 loop operation assumes that with the isolation valves closed, no temperature difference is induced.

The most limiting design transient was given as loss of reactor coolant pump flow in both three and two pump operation for which the DNBR values are 1.5 and 2.0 respectively. The margin to DNB is more convervative with N-1 loop operation, primarily because of reduced peak power.

Inlet Flow Maldistribution

In response to our question, the licensee stated that both flow model tests and analytical studies were used to examine the possibility of inlet flow maldistribution with two loop operation. The results of 1/7 scale hydraulic reactor model tests (Ref. 3 and 4) were applied in analytical studies using the THINC code (Ref. 5). The experimental data in conjunction with the THINC analyses show that it is adequate to use a 5% reduction in inlet flow to the hot assembly for operation with one loop out of service.

The licensee provided flow maps for three loop and two loop operaton showing the normalized inlet flow distributions. A comparison of the average normalized flow in the center nine assemblies for both three and two loop operation shows an agreement within approximately 2%. The studies performed in Reference 5 show that even with a 10% flow reduction into the center nine assemblies of the core, the hot channel DNBR is reduced by less than 0.5%.

The licensee stated that any asymmetries that exist due to N-1 loop operation would have little or no impact on power distribution, DNB limits and fuel integrity. From Reference 5 it was found that even with extreme inlet flow maldistributions, hot channel enthalpy rise and DNBR are only slightly affected. Generic radial power distributions and a 5% flow reduction into the hot assembly used in the safety analyses account for any inlet flow maldistributions. The restrictions placed on the fuel to ensure fuel integrity are applicable to both N and N-1 loop operation as stated in Chapter 3 of the BVPS Unit 1 FSAR.

Flow Instability

The licensee stated that Reference 6 was used for the analysis of flow instability with two loop operation. This showed that the margin to inception of thermohydrodynamic instability increases with a decrease in exit quality. Since the exit quality for N-1 loop operation is less than for N loop operation, a greater margin to flow instability exists for N-1 loop operation.

We have reviewed the thermal hydraulic information on two loop operation for BVNP-1 pertaining to thermal-hydraulic parameters, inlet flow maldistribution and flow instabilities and has found them acceptable.

B. Pressurized Thermal Shock

Failure of the reactor pressure vessel can occur when highly irradiated welds are exposed to both high pressure and low temperature. This phenomenon, known as Pressurized Thermal Shock (PTS), is thought to be most severe for certain small break LOCA events, in which natural circulation is lost and repressurization occurs. In such cases, low temperature HPI water would gradually cool the cold leg and reactor vessel downcomer. The resulting combination of low temperature and high pressure would present a serious challenge to the circumferential and axial welds in the downcomer wall. The NRC staff has addressed the problem of limiting the probability of vessel failure from this and other events (Reference 7).

Operation with one loop isolated presents a potential additional PTS risk. With the cold leg isolation valve closed, the HPI water injected at the inactive loop would experience only limited mixing with hot water before entering the downcomer. This would aggravate the overcooling effect of the events calculated in Reference 7. Furthermore, the potential would exist for overcooling in the region of the isolated cold leg under all circumstances when HPI is initiated. To date there have been almost 20 HPI actuations at Beaver Valley unit 1.

The Office of Nuclear Regulatory Research (RES) contracted with S. Levy Inc. to perform a series of calculations to determine the impact of loop isolation over a range of thermal-hydraulic conditions (reference 8). The calculations were intended to answer two questions; (1) does loop isolation significantly aggravate the overcooling during accidents which have been identified as PTS events, and (2) does the isolation of a loop create new scenerios (such as spurious actuation of HPI) in which overcooling in the downcomer can occur?

To answer the first question, temperature profiles in the downcomer with no coolant flow were calculated for N-1 loop operation, in order to simulate conditions during a SBLOCA with loss of natural circulation. Temperatures at the top of the nearest axial weld below the isolated loop cold leg were found to be 25°F colder than corresponding locations under the other cold legs. Calculations of the impact of loop isolation during an excessive cooldown event were also performed and showed a 25°F temperature difference.

A temperature decrease of this magnitude is conservatively estimated to increase the risk of crack initiation and propagation by about two orders of magnitude (X100). However, this large increase in risk is reduced by two factors; (1) the increased probability applies to only one of the three cold legs and (2) N-1 loop operation will be an infrequent occurrence during the lifetime of the plant. Furthermore, the temperature difference due to N-1 loop operation is lower (< 10°F) at the location of peak neutron flux where PTS risks is greatest.

To answer the second question, temperature profiles in the downcomer below the isolated loop cold leg were calculated for both forced flow (reactor coolant pumps on) and natural circulation conditions. For all cases analyzed, the temperature of the coldest weld below the isolated loop cold leg was less than 40°F colder than the corresponding location under the active cold legs. The lowest temperature reached by any of the welds was approximately 440°F, well above the range in which pressurized thermal shock is a concern. These results indicate that no PTS problem would result from normal HPI actuation at power.

The SLI calculations were reviewed by NRC/RES staff. The validity of the computer model was checked with a limited number of benchmark calculations, and by comparison with 1/5 scale tests performed by Creare, Inc. Plant data used in the calculations were taken from the BVPS FSAR or obtained directly from the licensee.

The proposed PTS rule places requirements on all PWR's with forgings, plates or axial weld approaching an RT_{NDT} of 270°F or circumferential welds approaching 300°F. Because our calculation indicates a negative 25°F temperature difference due to loop isolation, the screening criteria for N-1 loop operation at BVPS will be 245°F and 275°F respectively. This limitation will apply only to materials in the downcomer region below the loops to be isolated.

C. Steam Generator Tube Damage

We believe that two precautions should be taken to avoid undetected damage to steam generator tubes. First, the isolated loop must be maintained in a water solid condition, or any voids in the loop must be nitrogen inerted, to prevent corrosion in the SG tubes.

Second, the addition of heat to a water solid volume can lead to differential pressure levels well beyond the design values for the steam generator tubes. Prolonged operation in such a mode could produce undetected damage in the tubes, which would represent a safety problem when the isolated loop is returned to service. To prevent overpressurization, the isolated loop must be equipped with an appropriately sized pressure-actuated relief valve and a pressure monitor. The licensee has proposed a relief valve setpoint of 200 psig.

D. Power Peaking Factors

N-1 loop operation will involve power levels at or below 65 percent power, probably over extended periods of time (likely in excess of two weeks). This could lead to the possibility of core power distributions and peaking factor increases beyond those normally considered, if there is a return to full power operation within the same cycle. The peaking factor problem associated with extended part power operation has recently been discussed by Westinghouse in the enclosure of a letter from E. P. Rahe, Westinghouse to C. H. Berlinger, NRC, November 8, 1983, "NRC Request for Reduced Power Operation - Operating Procedure." A copy of the report, "Extended Reduced Power Operation Evaluation and Recommended Operating Procedure," was also furnished to Westinghouse utility customers in August 1983. The report suggests procedures to minimize the peaking factor increase effect and to maintain a methodology which can retain power density limits on return to full power with normal (F_{xy} or F_Q) surveillance. The report is still under review by the NRC staff, but the interim position is that the procedure recommended in the report should be followed for N-1 loop operation (and all extended operation below 85 percent power) if return to full power is contemplated for that cycle.

E. Conclusions

Although several potential safety issues have been identified, the proposed solutions are considered acceptable. The staff concurs that the risk from these issues is small as long as the solutions referred to in this section are implemented.

IV. Accident Analysis

With the exception of a few transients which were analyzed for N-1 loop operation in the original FSAR, a full range of transients and accidents has been reanalyzed or reevaluated in Reference 1. These calculations have been reviewed by the staff, with comments provided by the Idaho National Engineering Laboratory (Reference 9).

A. Loss of Coolant Accidents And Asymmetric Blowdown Loads

The Large Break Loss of Coolant Accident (LBLOCA) was analyzed in Reference 1 using the 1975 version of the Westinghouse evaluation model. Compliance with the acceptance criteria of 10 CFR 50.46 was demonstrated for double-ended cold leg breaks in an active loop and in the unisolated segment of the inactive loop. As in the N-loop case, the highest peak clad temperature (2155°F) resulted for a discharge coefficient of 0.4. At the time Reference 1 was submitted, the 1975 Westinghouse model had already been superseded by the 1979 model, which has since been superseded by the 1981 model. In order to confirm the adequacy of the ECCS evaluation, the licensee submitted a reanalysis of the limiting LBLOCA (C_d = 0.4) performed with the 1981 model (Reference 17). The analysis showed compliance with 10 CFR 50-46, with a peak clad temperature of 1882°F.

The Small Break Loss of Coolant Accident (SBLOCA) was reevaluated but not reanalyzed. This is primarily because the SBLOCA was shown to be nonlimiting in the original FSAR for N-loop operation, and because loss of one coolant loop is not a significant factor in SBLOCA, particularly since ECCS from that loop is still available. In cases where heat removal is through natural circulation, one steam generator is capable of providing more than enough cooling. Moreover, two significant benefits are derived from the reduced core average power (65%); (1) reduced system pressure allows higher ECC flow and earlier actuation of the accumulators, and (2) the lower steam production rate delays core uncovery. Furthermore, the reduced peak power leads to slower heatup of the hot pin following uncovery.

The licensee has demonstrated satisfactory ECCS performance for both LBLOCA and SBLOCA during N-1 operation.

In the event of a double-ended guillotine break, a decompression shock wave would propagate in the cold leg and impinge on the core barrel. The amplitude of the wave is proportional to the difference between the system pressure and the saturation pressure ($P - P_{sat}$). With the isolation valve in the cold leg closed, the segment of pipe between the isolation valve and the vessel will experience little or no flow. Consequently, the cold-leg temperature and saturation pressure could be less than those in the operating loops. Therefore, the amplitude of a blowdown shock wave in the inoperable leg could be significantly larger than for cases previously analyzed, and could exceed the structural capacity of the reactor internals to withstand it. The licensee should perform an analysis to evaluate the possible magnitude of such a shock wave. The licensee may, alternatively, submit a fracture mechanics analysis to demonstrate that a double-ended guillotine break is not a credible event (see Generic Letter 84-04, dated February 1, 1984).

B. Non-LOCA Transients and Accidents

Main Steamline Break (MSLB)

The MSLB is more limiting with one loop isolated. The reduced coolant inventory leads to more rapid cooling and a greater reactivity insertion. To compensate, the technical specification shutdown margin will be increased from 1.7% $\Delta k/k$ to 2.4% $\Delta k/k$. The new value is similar to that required in Westinghouse two-loop plants.

The steam line break analysis for 2-loop operation was performed with approved codes and reasonable assumptions. The results show that reactor system pressure remains below operating pressure and that the minimum DNBR is greater than 1.30. Consequently the criteria of the Standard Review Plan are met.

Beaver Valley UNit 1 has received approval for diluting the boron concentration in the boron injection tank (BIT) from 20,000 ppm to 2,000 ppm. The analysis presented in support of that amendment did not include the N-1 case. Consequently, the BIT must either be maintained at 20,000 ppm during N-1 loop operation, or an analysis be submitted to justify the 2,000 ppm concentration for N-1 loop operation.

Feedwater Line Break

The feedwater line break accident for N-1 loop operation was analyzed with significantly different assumptions from the FSAR N-loop analysis. The most notable difference is that the reactor tripped on low-low level in the faulted steam generator, a fact which resulted in significantly less stored energy in the primary system at the time of reactor trip. A second difference is that safety injection was assumed to operate. Although significantly different from the FSAR analysis, these assumptions are acceptable and are the same as those used in more recent FSAR's. The analysis shows slower pressurization of the primary system, and lower primary temperatures.

The main negative impact of loop isolation on this accident is the availability of one less SG for heat removal. This is offset to some extent by the reduced core power. The fact that the new analysis shows less severe response to the feedline break is due primarily to the new assumptions discussed above.

This accident was analyzed with acceptable codes and methods, and produced results which generally conform to the staff's understanding of how the accident should be affected by isolating one loop. All of the applicable acceptance criteria were met.

Other Class IV Events

The RCP locked rotor event with one loop isolated led to somewhat higher peak pressure (2730 psia vs. 2690 psia), but still did not exceed the 110% of design pressure. The calculated peak clad temperature is lower in the N-1 loop case because of the reduced core power.

Steam Generator Tube Rupture (SGTR)

The original FSAR for BVPS-1 showed compliance with the radiation dose limits of 10 CFR 100 for a SGTR during N loop operation. In the submittal under review, the licensee has asserted that the consequences of a SGTR during N-1 loop operation are bounded by the N loop case. Most of the system parameters which affect SGTR are not altered by the isolation of one loop. A significant advantage of N-1 loop operation is reduced fission product inventory due to operation at 65% of full power. Offsite releases would be significantly reduced.

Although one less steam generator would be available for decay heat removal, the remaining SG would have more than sufficient capacity to remove decay heat and cool the RCS at a rate of 75°F per hour.

Several questions have been raised covering the technical basis for the SGTR analysis presented in the FSAR for BVPS-1 and other plants. The main issues relate to the use of non safety-grade PORV's, and to the need for the licensee to provide justification for the assumption that the operator can isolate the affected SG within 30 minutes. Although resolution of these issues could potentially affect the results of the N-1 loop analysis, they are not strictly associated with the question of N-1 loop operation.

The licensee has demonstrated that the consequences of a SGTR during N-1 loop operation would be less severe than for N loop operation.

DNBR-Limited Transients

For a broad range of transients involving loss of reactor coolant flow, depressurization of the primary system or loss of secondary heat removal, the principal acceptance criterion of the Standard Review Plan is that DNBR must remain above 1.30. The margin to DNB during normal operation with N-1 loops is significantly higher than for N loop operation, primarily because of the reduced peak power. Consequently the severity of this class of transients for N-1 loop operation has been found to be bounded by the N-loop results. In all cases, the peak pressures were within the limit of 110% of design pressure. Typical calculated DNBR values for these transients are shown in Table 3. The loss of normal feedwater event, not included in the table, produces only small increases in primary coolant temperature, and is significantly more benign for the N-1 loop case. Loss-of-offsite-power calculations produce results similar to the loss-of-reactor-coolant-flow event.

Boron Dilution

Isolation of one RCS loop would have two major impacts on the boron dilution event. The reduced RCS volume would allow more rapid dilution, but this effect would be offset by the higher shutdown margin (2.4 vs. 1.7%). Hand calculations by the staff indicate that the net effect would be a small (approximately 25%) increase in the time allowed for operator action.

Excessive Heat Removal

Excess heat removal due to malfunctions of the steam or feedwater systems are bounded by the main steamline break, which meets the pressure and DNBR criteria for this type of event.

Reactivity Transients

The reactivity transients include control bank withdrawal at startup and at power, control rod ejection and the control rod misoperation events including rod drop, single rod withdrawal and rod misalignment. The licensee has submitted reanalyses of the rod bank withdrawal and rod ejection events with conditions applicable to N-1 loop operation and using, for the most part, methods and criteria of the FSAR N loop analyses.

For the control rod bank withdrawal at startup, DLC reanalyzed using the same methodology and criteria except that a newer spatial kinetics code (TWINKLE) was used instead of the point kinetics code used on the FSAR. This is (part of) the current methodology as used, for example, in the Beaver Valley 2 FSAR. The transient results should change very little since this event is not very sensitive to the primary parameter change, reduced flow. But this improved methodology provides lower transient power and fuel temperature than with point kinetics and gives temperatures sufficiently low that there is a large margin to DNB even with the reduced flow. This result is to be expected from more recent analyses of the event, e.g., Beaver Valley 2, where analysis with only two pumps and no isolation results in large DNB margin as calculated with this methodology and with THINC analysis of DNB.

The control rod withdrawal at power events were reanalysed with N-1 loop parameters including those for the overtemperature delta T trip setpoint. The resulting margins to DNB are larger than for N loop operation, largely because of the improved initial DNBR state.

The control rod ejection events at N-1 loop full power (65 percent) conditions were calculated using standard Westinghouse methodology. The resulting fuel temperatures (and enthalpies) were well within normal criteria. This would be expected since the results are not very sensitive to the parameter changes. The standard conservative generic results for system overpressure and amount of fuel failures assumed for DNB are fully applicable to the range of conditions in N-1 loop operation and thus no new calculations are needed in these areas.

These reactivity transients have been suitably analyzed at N-1 loop conditions with acceptable methods. All results meet required criteria and the changes from N loop analysis results are in accordance with staff expectations.

The events falling under the control rod misoperation category have not been specifically analyzed at N-1 loop conditions. However, the events proceed along paths parallel to the N loop condition analyses but with the core further removed from limiting DNB conditions because of the improved initial DNBR conditions. The rod drop, rod misalignment and the single rod withdrawal at power would each have the same extreme rod configuration to analyze at N-1 loop conditions as normally used for N loop conditions, but the improved initial state would result in improved extreme DNBR states.

It is thus concluded that under the allowed conditions for N-1 loop operation the reactivity transients normally analyzed are no more severe than for N loop conditions and that all applicable criteria for these events would be met.

Events Not Reanalyzed

Several accidents, for which N-1 loop operation is not judged to be a significant factor, were not reanalyzed in Reference 1. These include non-thermal-hydraulic events such as fuel handling accidents and accidental releases of stored waste. Events which are judged to be precluded by administrative controls or automatic interlocks were also not reevaluated. These include the inadvertent startup of an inactive loop and inadvertent mis-loading of a fuel assembly.

C. Conclusion

The impact of RCS loop isolation varies from accident to accident. In some instances, there is a measurable loss of safety margin, while in others, it is increased. In all cases the calculated responses to accidents and transients meet the acceptance criteria of the standard review plan.

V. Restrictions to N-1 Loop Operation

Proposed technical specifications have been submitted by the licensee (Reference 1). However, due to the concerns discussed in detail above, these proposed technical specification changes should be revised accordingly; commitments have been made by the licensee in his responses to staff inquiry (References 10 - 20). They are summarized as follows:

1. To avoid undetected damage to the SG tubes, the licensee has suggested a maximum pressure of 200 psig for both the primary and secondary side of the isolated loop. This limitation must be proposed as a technical specification (Section III.C).

2. The isolated loop should be monitored in a water solid or nitrogen-inerted condition to prevent SG tube corrosion (Section III.C).

3. Because the steamline break analysis was performed assuming that the boron injection tank contains water with 20,000 ppm boron, this concentration must be a technical specification for N-1 loop operation, until a revised analysis based on a lower concentration is submitted and accepted (IV.B).

4. Isolation of a loop or returning one to service is permitted only at cold shutdown (II.A, II.B and II.F).

5. The setpoint for the power range neutron flux channels will be reduced to 71% and P-7 will be increased to a value which would avoid misleading information to the plant operator (Section II.C).

6. As noted in Section II.C above, several changes in instrumentation setpoints will have to be made in the plant Technical Specifications.

In addition, the following staff concerns should be addressed by analysis, new procedures, or both:

7. Procedures to prevent excessive power peaking factors following return to N loop operation (Section III.D)

8. Human factors concerns (Section II.G)

9. Asymmetrical LOCA blowdown load (Section IV.A)

By letter dated April 10, 1984 the licensee informed us that due to the long time this review effort has spanned, and the changeover of personnel at the NSSS vendor (Westinghouse), work has been initiated to confirm the documentation being used to support the pending amendment. Thus the licensee should wait until such confirmatory design review is completed before the revised amendment request is submitted.

VI. Summary and Conclusions

We conclude that the proposed method of loop isolation provide adequate protection for the integrity of the primary pressure boundary. We also conclude that the likelihood of accidents and transients would not be significantly increased. The isolation of one loop will not seriously degrade the performance of safety related systems, that of the instrumentation and control systems, or that of the closed loop isolation valves.

With respect to the safety issues examined, we conclude that no major safety problems would result from N-1 loop operation (these include core thermal hydraulics, pressurized thermal shock, and SG tube damage).

The licensee has reanalyzed the full spectrum of transients and accidents using acceptable codes and assumptions. The results demonstrate compliance with the acceptance criteria of the regulations and the standard review plan.

As a result of the analyses submitted by the licensee and the review conducted by the staff, some changes to the licensee's proposed technical specifications will be necessary to ensure safe operation with one isolated loop. These changes are summarized in Section V. We will evaluate the adequacy of the revised technical specification changes to be proposed by the licensee against these concerns. Issuance of an amendment authorizing N-1 loop operation will be contingent upon satisfactory resolution of these concerns.

On the basis of the considerations discussed above, we conclude that N-1 loop operation at Beaver Valley Unit 1 does not constitute a threat to public health and safety.

VII. References

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TABLE 1

Protection Systems Parameters and Conditions
Associated with an Out-of-Service Loop

Parameter	Safety Functions
1. Primary Coolant	
a. Hot leg temperature b. Cold leg temperature	Input signals for overpower and overtemperature ΔT reactor trip. Feedwater isolation and steam dump interlocks
2a. Primary coolant flow	Input signal for low flow reactor trip.
2b. RCP Breaker tripped	Input signal for low flow reactor trip (two loops).
3. Steam Generator Level	Input signals for low level coincident with steam/feedwater flow mismatch and low-low level reactor trip and low-low level initiation of auxiliary feedwater. Input for hi-hi level (P-14) turbine trip and feedwater isolation.
4. RCP Bus Undervoltage and Underfrequency	Input signals for reactor trip
5. Steam Generator Pressure	Input signal for safety injection on low pressure and high negative pressure rate.

TABLE 2

BEAVER VALLEY UNIT 1

THERMAL AND HYDRAULIC COMPARISON

Design Parameters	3 Loop Operation	1 Loop Isolated
Reactor core heat output (MWt)	2,652	1724
Reactor core heat output (10^6 Btu/hr)	9,051	5884
Heat generated in fuel (%)	97.4	97.4
System pressure, nominal (psia)	2,250	2,250
System pressure, minimum steady state (psia)	2,220	2,220
Minimum DNBR at nominal design conditions		
Typical flow channel	2.26	2.97
Thimble (cold wall) flow channel	1.83	2.43
Minimum DNBR for design transients	>1.30	>1.30
DNB Correlation	"R" (W-3 with modified spacer factor)	"R" (W-3 with modified spacer factor)
Coolant Flow		
Total thermal flow rate (10^6 lbm/hr)	100.8	72.1
Effective flow rate for heat transfer (10^6 lbm/hr)	96.3	68.8
Effective flow area for heat transfer (ft^2)	41.6	41.6
Average velocity along fuel rods (ft/sec)	14.4	10.1
Average mass velocity (10^6 lbm/hr-ft^2)	2.32	1.66
Coolant Temperatures		
Nominal inlet (°F)	542.5	534.4
Average rise in vessel (°F)	67.5	63.2
Average rise in core (°F)	70.3	65.9
Average in core (°F)	579.4	568.7
Average in vessel (°F)	576.2	566.0
Heat Transfer		
Active heat transfer, surface area (ft^2)	48,600	48,600
Average heat flux (Btu/hr-ft^2)	181,400	118,000
Maximum heat flux for normal operation (Btu/hr-ft^2)	420,900*	326,700**

Table 2 (Continued)

Heat Transfer	Operation	Isolated
Average linear power (kW/ft)	5.20	3.38
Peak linear power for normal operation (kW/ft)	12.1*	9.4**
Peak linear power resulting from overpower transients/operator errors, assuming a maximum overpower of 118% (kW/ft)	18.0	18.0
Peak linear power which would result in centerline melt (kW/ft)	>18.0	>18.0
Fuel Central Temperature		
Peak at linear power for prevention of centerline melt (°F)	4,700	4,700
Pressure drop*** Across core (psi)	21.3 ± 2.1	11.2 ± 1.1

NOTES:

*This limit is associated with the value of F_Q = 2.32.
**This limit is associated with the value of F_Q = 2.77
*** Based on best estimate reactor coolant flow rate.

Table 3

Calculated Minimum DNBR

	N loop	N-1 loop
Loss of RCP flow	1.5	2.0
Loss of Load (worst case)	1.6	2.1
Depressurization of The RCS	1.45	Not calculated