DISTRIBUTION
Docket File
RSB R/F
RSB P/F: BV-2
ELantz R/F
AD/RS Rdg.
BSheron
LMarsh
ELantz

MAY 15 1984

MEMORANDUM FOR: Thomas M. Novak, Assistant Director for Licensing
Division of Licensing

FROM: R. Wayne Houston, Assistant Director for Reactor Safety
Division of Systems Integration

SUBJECT: BEAVER VALLEY UNIT 2 DRAFT SER

Plant Name;	Beaver Valley Unit 2
Docket No.:	50-412
Licensing Stage:	OL
Responsible Branch:	Licensing Branch #3
Project Manager:	M. Licitra
Review Branch:	Reactor Systems Branch
Review Status:	Awaiting Information

Attached is the Reactor Systems Branch's input to Chapter 15 of the draft SER for BV-2. This includes draft evaluations of the TMI related issues for which RSB has lead responsibility. RSB input to SALP is also attached.

R. Wayne Houston, Assistant Director
for Reactor Safety
Division of Systems Integration

cc: w/o enclosure
RSB Section Leaders
M. Licitra
M. Ley

CONTACT: E. Lantz, RSB
X27839

8405290144 XA

OFFICE	DSI:RSB	DSI:RSB	DSI:RSB	DSI:AD:RS
NAME	ELantz:gd	LMarsh	BSheron	RHouston
DATE	05/11/84	05/15/84	05/14/84	05/15/84

15 ACCIDENT ANALYSES

The accident analyses for Beaver Valley Unit 2 have been reviewed in accordance with Section 15 of the SRP (NUREG-0800). Conformance with the acceptance criteria, except as noted for each of the sections, formed the basis for concluding that the design of the facility for each of the areas reviewed is acceptable.

In accordance with SRP 15.1.1, Paragraph I, the applicant evaluated the ability of Beaver Valley Unit 2 to withstand anticipated operational occurrences and a broad spectrum of postulated accidents without undue hazard to the health and safety of the public. The results of these analyses are used to show conformance with GDC 10, 15, 27, and 31.

For each event analyzed, the worst operating conditions and the most limiting single failure were assumed, and credit was taken for minimum engineered safeguards response. In questions 440.73 and 440.74 the staff has asked the applicant to:

1. Supply listings of the single failures which were assumed for each event in the Chapter 15 analyses.

2. Supply the limiting single failure that results in the peak pressure or limiting performance for each event.

3. Show the effect of a loss of offsite power on all anticipated operational occurrences and postulated accidents.

When this information is received it will be incorporated into the evaluations of the individual events.

Parameters specific to individual events were conservatively selected. Two types of events were analyzed

(1) those incidents that might be expected to occur during the lifetime of the reactor

(2) those incidents not expected to occur that have the potential to result in significant radioactive material release (accidents)

The nuclear feedback coefficients were conservatively chosen to produce the most adverse core response. The reactivity insertion curve, used to represent the control rod insertion, accounts for a stuck rod; it is in accordance with GDC 26.

For transients and accidents, the applicant used a method that conservatively bounds the consequences of the event by accounting for fabrication and operating uncertainties directly in the calculations. DNBRs were calculated using the W-3 correlation with a modified spacer factor R, with a minimum DNBR of 1.3 used as the threshold for fuel failure.

The applicant accounts for variations in initial conditions by making the following assumptions as appropriate for the event being considered:

	3-Loop Operation	2-Loop Operation
Core Power (MWt)	2652 + 2%	1724 + 2%
Average Reactor Vessel Temperature (°F)	576.2 ± 4%	566.0 ± 4%
Pressure (psi) (at pressurizer)	2250 ± 30	2250 ± 30

The staff concludes the assumptions for initial conditions are acceptable because they are conservatively applied to produce the most adverse effects. These assumed values will form the basis for the technical specification limits. For transients and accidents used to verify the ESF design, the applicant used the safeguards power design value of 2780 MWt.

The applicant has also analyzed several events expected to occur one or more times in the life of the plant. A number of transients can be expected to occur with moderate frequency as a result of equipment malfunctions or operator errors in the course of refueling and power operation during the plant lifetime.

Specific events were reviewed to ensure conformance with the acceptance criteria provided in the SRP.

The acceptance criteria for transients of moderate frequency in the SRP include the following:

(1) Pressure in the reactor coolant and main steam systems should be maintained below 110% of the design values (Section III of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code).

(2) Fuel clad integrity shall be maintained by ensuring that the minimum DNBR will remain above the 95/95 DNBR limit for PWRs. (The 95/95 criterion discussed in Section 4.4 of this SER provides a 95% probability, at a 95% confidence level, that no fuel rod in the core experiences a DNB.)

(3) An incident of moderate frequency should not generate a more serious plant condition without other faults occurring independently.

(4) For transients of moderate frequency in combination with a single failure, no loss of function of any fission product barrier, other than fuel element cladding, shall occur. Core geometry is maintained in such a way that there is no loss of core cooling capability and control rod insertability is maintained.

Conformance with the SRP acceptance criteria for anticipated operational occurrences constitutes compliance with GDC 10, 15, and 26 of Appendix A to 10 CFR 50. See Section 6.8 of this SER for a discussion of auxiliary feedwater system conformance to TMI Action Plan Item II.E.1.1 and Sections 6.8 and 7.3.1.7 for a discussion of compliance with TMI Action Plan Item II.E.1.2.

The transients analyzed are protected by the following reactor trips:

(1) power range high neutron flux
(2) high pressure

(3) low pressure
(4) overpower ΔT
(5) overtemperature ΔT
(6) low coolant flow
(7) pump undervoltage/underfrequency

(8) low steam generator water level
(9) high steam generator water level

Time delays to trip, calculated for each trip signal, are included in the analyses. See Section 4.6 of this SER for a discussion of the staff review of reactivity control system functional design.

All of the events that are expected to occur with moderate frequency can be grouped according to the following plant process disturbances: changes in heat removal by the secondary system, changes in reactor coolant flow rate, changes in reactivity and power distribution, and changes in reactor coolant inventory. Design-basis accidents have been evaluated separately and are discussed at the end of this section of the SER.

15.1 Increase in Heat Removal by the Secondary System

The applicant's analysis of events that produced increased heat removal by the secondary system is addressed in the following paragraphs.

15.1.1 Decrease in Feedwater Temperature

The consequences of a decrease in feedwater temperature transient are bounded by those in Sections 15.1.2 and 15.1.4. The peak pressure is less than that in Section 15.1.2. The minimum DNBR is greater than that in Section 1.5.1.4.

15.1.2 Increase in Feedwater Flow

Increases in feedwater flow decrease the temperature of the reactor coolant water. Due to the negative moderator temperature coefficient this will insert positive reactivity and increase core power.

In Section 15.1.2.1 of the FSAR the applicant states that for these events the high neutron flux trip, overtemperature ΔT trip, and overpower ΔT trip prevent any power increase which could lead to a DNBR less than the limit value of 1.30. However, the only analytical results presented for these events are those where a steam generator hi-hi level trip closes all feedwater control.and isolation valves, trips the main feedwater pumps, trips the turbine, and initiates a reactor trip. The applicant states that continuous addition of feedwater is prevented by the steam generator hi-hi level trip.

This analysis shows that the maximum reactivity insertion rate due to an increase in feedwater flow occurs at no-load conditions and is less than the maximum value calculated for an inadvertent control rod withdrawal, which is evaluated in Section 15.4 of this SER. However, this analysis also shows that an increased feedwater flow event can cause a peak RCS pressure of 2270 psia. This is below the design pressure of 2485 psig, but it is the highest RCS pressure the applicant calculated for any of this group of events.

15.1.3 Increase in Steam Flow

The consequences of an increase in steam flow transient are bounded by those in Sections 15.1.2 and 15.1.4. The peak pressure is less than that in Section 15.1.2. The minimum DNBR is greater than that in Section 15.1.4.

15.1.4 Inadvertent Opening of a Steam Generator Relief Valve or Safety Valve

The transient that is most limiting of this group of transients with respect to fuel performance is the inadvertent opening of the steam generator relief or safety valve. The suddenly increased steam demand causes a reactor power increase which results in a reactor trip due to high neutron flux, overtemperature, or overpower signals. The continued steam flow through the open valve will cause additional cooldown which will, because of the negative moderator temperature coefficient, result in positive reactivity. The safety injection system (SIS) will inject highly concentrated boric acid from the boron injection tank into the primary coolant system on either two out of three pressurizer low pressure signals, or two out of three low steamline pressure signals in any one loop. This ensures the reactor will be shut down during any subsequent cooldown. The normal steam generator feedwater would be isolated automatically upon SIS initiation, and then the plant would be gradually cooled down with only safety-grade equipment. DNB does not occur during this transient.

The applicant has provided results of its study for a transient of this group in combination with its limiting single failure. No credible single failure has been identified that could result in a more limiting peak reactor coolant system pressure or DNBR than that from the events themselves.

The applicant's analyses show that for transient events leading to an increase in heat removal by the secondary system (with or without single failure), the minimum DNBR is 1.3. Thus no fuel failure is predicted to occur, core geometry and control rod insertability are maintained with no loss of core cooling capability, and the maximum reactor coolant system pressure remains below 110% of design pressure. The staff finds the results of these analyses in conformance with the acceptance criteria of SRP 15.1.1 through 15.1.4, and, therefore, acceptable.

15.1.5 Steamline Rupture Accident

The applicant has submitted analyses of postulated steamline breaks that show no fuel failures attributed to the accident. These results are similar to those obtained for previously reviewed Westinghouse three-loop plants.

A postulated double-ended rupture at hot standby power with no decay heat was analyzed as the worst case. Since the steam generators have integral flow restrictors with a 1.4 ft^2 throat area, any rupture with a break area greater than 1.4 ft^2, regardless of location, will have the same effect on the system as a 1.4 ft^2 break; so this was assumed in the analysis. The doubled-ended rupture would cause the reactor to increase in power due to the decrease in reactor coolant temperature. The reactor would be tripped by either reactor overpower ΔT or by the actuation of the SIS. The SIS will be actuated by any of the following: two out of three low pressurizer pressure signals; two out of three HI-1 containment pressure signals; or two out of three low steamline pressure signals in any one loop. The transient is terminated using only safety-grade equipment. The injection of highly borated water ensures the reactor is returned to and then maintained in a shutdown condition.

The staff concludes that the consequences of postulated steamline breaks meet the relevant criteria in GDC 27, 18, 31, and 35 regarding control rod insertability and core coolability and TMI Action Plan Items. This conclusion is based upon the following:

(1) The applicant has met the criteria of GDC 27 and 28 by demonstrating that fuel damage, if any, is such that control rod insertability will be maintained, and there will be no loss of core cooling capability. The minimum DNBR experienced by any fuel rod was ≥1.30, resulting in none of the fuel elements being predicted to experience cladding perforation.

(2) The applicant has met the criteria of GDC 31 with respect to demonstrating the integrity of the primary system boundary to withstand the postulated accident.

(3) The applicant has met the criteria of GDC 35 with respect to demonstrating the adequacy of the emergency cooling systems to provide abundant core cooling and reactivity control (via boron injection).

(4) A mathematical model, which accounts for incomplete coolant mixing in the reactor vessel, has been reviewed and found acceptable by the staff. This model was used to analyze the effects of steamline breaks inside and outside of containment, during various modes of operation, with and without offsite power.

(5) The parameters used as input to this model were reviewed and found to be suitably conservative.

15.2 Decrease in Heat Removal by the Secondary System

The applicant's analyses of events that result in a decrease in heat removal by the secondary system are presented below.

15.2.1 Steam Pressure Regulator Malfunction or Failure that Results in Decreasing Steam Flow

In Section 15.2.1 of the FSAR the applicant states that any steam flow decrease caused by a malfunction or failure of any steam pressure regulator is conservatively bounded by the turbine trip event and analyzed in Section 15.2.3.

15.2.2 Loss of External Load

In Section 15.2.2 of the FSAR the applicant states that the results of the turbine trip event analysis are more severe than those expected for the loss of external load. The reason given is that a turbine trip actuates the turbine stop valve whereas a loss of external load actuates only the turbine control valves. Since the stop valve can more suddenly cut off the steam flow to the turbine this is a more severe "decreased heat removal" transient.

15.2.3 Turbine Trip

Assuming offsite power is available to run the reactor coolant pumps, the applicant analyzed the turbine trip event for a complete loss of steam load from full power without a direct reactor trip and with only the pressurizer and steam generator safety valves assumed for pressure relief. These assumptions result in the highest peak RCS pressure for any "decreased heat removal" event. The calculated peak value is 2560 psia, which is well below the ASME limit of 110% of the design pressure. For these assumptions the minimum DNBR is 1.75, which is well above the minimum limiting value of 1.30.

The applicant's analyses show that if instead of relying on just the safety valves, the pressurizer spray and PORV's are used to limit the pressure during this turbine trip event, the minimum DNBR can go down to 1.60. If a stuck open

PORV were to be assumed as the single failure during this course of action, it appears that the DNBR could go lower. The applicant has not discussed the possibility of a stuck open PORV or atmospheric steam dump valve being the worst single failure during this course of action.

The consequences of a turbine trip without offsite power available are discussed in Section 15.2.6.

15.2.4 Inadvertent Closure of Main Steam Isolation Valves

Consequences are the same as those discussed in Sections 15.2.3 and 15.2.6.

15.2.5 Loss of Condenser Vacuum

Consequences are the same as those discussed in Sections 15.2.3 and 15.2.6.

15.2.6 Loss of Nonemergency AC Power to the Plant Auxiliaries

A loss of nonemergency ac power event is more limiting than the turbine-trip-initiated decrease in secondary heat removal without loss of ac power because the reactor coolant pumps are lost and the subsequent flow coastdown further reduces the amount of heat the primary coolant can remove from the core. In this transient, the loss of offsite power is closely followed by a turbine trip and reactor trip. The reactor trip is assumed to come from low-low steam generator level which is the second safety-grade trip. The emergency feedwater system is automatically started and one electric-motor-driven pump is assumed to be feeding all three steam generators.

The applicant's LOFTRAN analysis shows that the natural circulation flow available adequately transfers the decay heat from the core to steam generators, which are being fed with emergency feedwater flow. The steam which is generated is assumed to be relieved through the steam generator safety valves. The primary system relief valves are assumed not to function.

The emergency feedwater comes from the primary plant demineralizer water storage tank (PPDWST) which, the applicant states in FSAR Section 10.4.9.1, contains sufficient water to reduce the hot leg temperatures to 350°F for this transient. At 350°F the RHRS can be started to take away the decay heat.

The DNBR remains above 1.30 throughout this transient, and the peak RCS pressure remains below 110% of the design pressure.

15.2.7 Loss of Normal Feedwater Flow

The consequences of this anticipated operational occurrence are more severe if a concurrent loss of offsite power is assumed. However, if a loss of offsite power is assumed the consequences will be the same as the loss of nonemergency ac power event discussed in Section 15.2.6.

15.2.8 Feedwater System Pipe Breaks

The applicant has provided a feedwater line break analysis for Beaver Valley Unit 2 using assumptions that minimize secondary system heat removal capability, maximize heat addition to the primary system coolant, and maximize the calculated primary system pressure. A double-ended rupture of the largest feedwater line was assumed, as well as failure of the turbine-driven auxiliary feedwater pump to start and supply emergency feedwater to the steam generator.

The applicant used the NRC approved LOFTRAN code to do this analysis. The analysis assumed that with a single failure of the auxiliary feedwater system, emergency feedwater flow is supplied to two intact steam generators by only one electric-motor-driven auxiliary feedpump. This is sufficient feedwater flow to adequately remove the residual heat after reactor shutdown. The use of only safety-grade equipment will mitigate this accident. No fuel damage was calculated to occur, and the peak calculated pressurizer pressure was approximately 2500 psia. As required for all other events a list of the single failures that were considered and the most limiting single failure must be provided.

15.3 Decreases in Reactor Coolant Flow Rate

15.3.1/15.3.2 Loss of Forced Reactor Coolant Flow, Including Trip of Pump and Flow Controller Malfunctions

The applicant has analyzed the total loss of forced reactor coolant flow event that bounds partial loss of forced reactor coolant flow. This event was reviewed with the procedures and acceptance criteria set forth in SRP 15.3.1 -15.3.2.

The loss of offsite power and resulting loss of all forced coolant flow through the reactor core causes an increase in the average coolant temperature and a decrease in the margin to DNB. The reactor is tripped from an undervoltage trip monitoring the reactor coolant pump (RCP) power supply, and a minimum DNBR of 1.47 is reached 3.2 seconds into the transient. The maximum calculated RCS pressure is 2310 psia during the transient.

15.3.3/15.3.4 Reactor Coolant Pump Rotor Seizure and Shaft Break Accident

The applicant has analyzed the reactor coolant pump (RCP) rotor seizure and shaft break events with the LOFTRAN and FACTRAN computer codes. Since the initial rate of reduction of coolant flow is greater after an RCP rotor seizure, this is the limiting event. For the analyses the applicant assumed that the fuel cooling goes into the nucleate boiling regime (i.e., DNB) immediately at the beginning of the transient. The maximum RCS pressure will occur in the event of an RCP rotor seizure while only two of the three loops are operating. This maximum pressure is calculated to be 2647 psia with only the opening of the pressurizer and steam generator safety valves. The applicant states that 2647 psia is below the faulted condition stress limit of the RCS.

In response to a question on a loss of offsite power (LOOP) during these events, the applicant states that a LOOP will have only a negligible effect on the critical parameters of RCS pressure and clad temperature and that it would have no effect whatsoever on the conclusions. The staff finds that a quantitative analysis of the worst case, which would have only two loops in operation, with a concurrent loss of offsite power is needed for the evaluation of this issue.

The staff's evaluation and finding on fuel damage and consequent control rod insertability and core cooling considerations during this event are included in SER Section 4.2. The LOFTRAN computer code has been approved by the NRC. The remaining staff findings are

(1) The parameters used as input to the mathematical model are suitably conservative.

(2) The use of "Service Limit C" of the ASME Code is acceptable for conforming to GDC 31 and demonstrating the integrity of the RCS during this accident; the maximum pressure is below this limit.

15.4 Changes in Reactivity and Power Distribution

15.4.4/15.4.5 Startup of an Inactive Reactor Coolant Pump at an Incorrect Temperature

In FSAR Section 15.4.4, the applicant provides the results of an analysis for startup of an inactive reactor coolant pump event. This event was reviewed with the procedures and acceptance criteria set forth in SRP 15.4.4.

During the first part of the transient, the increase in core flow with cold water results in an increase in nuclear power and a decrease in core average temperature. Reactivity addition for the inactive loop startup event is the result of the decrease in core inlet water temperature. This transient was evaluated by the applicant using a mathematical model that has been reviewed and found acceptable to the staff. The maximum calculated RCS pressure is 2310 psia and the minimum DNBR is above 1.3 throughout the transient.

15.4.6 Inadvertent Boron Dilution

Various chemical and volume control system (CVCS) malfunctions which could lead to an unplanned boron dilution incident have been reviewed. The malfunctions that allow the operator the shortest time for corrective action have been analyzed starting from plant conditions of startup, power operation (automatic and manual), hot standby, and cold shutdown. The applicant used acceptably conservative assumptions in these analyses. The results show that the operator has at least 15 minutes between the time when an alarm announces an unplanned moderator dilution and the time of loss of shutdown margin, i.e., criticality.

The maximum reactivity insertion rate by boron dilution was found to be $1.5X10^{-5}$ $\Delta k/k$ (1.5 pcm) per second. In the event the operator does not stop the dilution, the DNBR will still remain above 1.49, and the RCS and main steam pressures will remain below 110% of design.

In response to a question on protection from inadvertent boron dilution during refueling, the applicant stated that during refueling the RCS is isolated from the potential source of unborated water. This isolation is accomplished by having the operators place danger tags on the primary grade water header isolation valves, or by locking these valves closed whenever the RCS water is below the normal level. The operator performing these tasks is required to sign off on each step of a procedural checklist. This long term use of administrative controls to prevent an inadvertent boron dilution during refueling has not been accepted by the staff on other plants, and will be evaluated. The staff is not at this point, convinced that a design basis event can be eliminated from detailed evaluation based on administrative means alone. We will report the resolution of this issue in a subsequent safety evaluation.

With the exception of the refueling mode the staff concludes that the analysis for the decrease in reactor coolant boron concentration event is acceptable and conforms to General Design Criterion 10, 15, and 26. This conclusion is based on the following:

1. The applicant has met the criteria of GDC 10 with respect to demonstrating that the specified acceptable fuel design limits are not exceeded for this event. This criterion has been met since the results of the analysis showed that the thermal margin limits are satisfied as indicated by SER Section 4.4.

2. The applicant has met the criteria of GDC 15 with respect to demonstrating that the reactor coolant pressure boundary limits have not been exceeded for this event. This criterion has been met since the analysis showed that the maximum pressure in the reactor coolant and main steam systems did not exceed 110% of the design pressure.

3. The applicant has met the criteria of GDC 26 with respect to demonstrating that the control rod system has the capability of overcoming the effects of boron dilution events during reactor operation. The applicant has demonstrated conformance with these criteria by showing that under the postulated accident conditions, and with appropriate margins for stuck rods, the specified acceptable fuel design limits are not exceeded.

15.5 Increases in Reactor Coolant System Inventory

15.5.1 Inadvertent Operation of the Emergency Core Cooling System During Power Operation

ECCS operation could be initiated by a spurious signal or an operator error. Two cases were examined, one in which reactor trip occurs simultaneously as a result of the safety injection signal, and the other in which the reactor trips later in the transient because of low reactor coolant system (RCS) pressure. The reactor pressure decreases during the initial phase of the transient and then increases to a peak pressure of 2350 psia at 200 seconds into the transient. The DNBR never drops below its initial value for either case. All of these transients are terminated by use of only safety-grade systems. If the operator fails to turn off the HHSI/charging pumps the safety valves will open. Continued operation of these pumps would overfill the Pressure Relief Tank. However, as stated in Table 6.3-1 of the FSAR the cutoff head of the HHSI/charging pumps is 6000 ft (2600 psig); so they cannot create 110% of the reactor vessel design pressure (2733 psig) and thus cannot fail the vessel.

15.5.2 CVCS Malfunction That Increases Reactor Coolant Inventory

Evaluation of consequences is included in Section 15.4.6.

15.6 Decrease in Reactor Coolant Inventory

15.6.1 Inadvertent Opening of a Pressurizer Safety or Relief Valve

In FSAR Section 15.6.1, the applicant provides the results of an analysis for inadvertent opening of a pressurizer safety valve. During this event, nuclear power is maintained at the initial value until reactor trip occurs on low pressurizer pressure. The DNBR decreases initially, but increases rapidly following the trip. The minimum DNBR of 1.50 occurred at 31 seconds into the transient. The RCS pressure decreases throughout the transient.

15.6.3 Steam Generator Tube Rupture

In response to the staff's concern that 30 minutes is not sufficient to diagnose and isolate a stream generator tube rupture, the applicant has provided additional data regarding the systems response and radiological consequences after a steam generator tube rupture accident. This information, however, did not support the isolation time of the affected steam generator at 30 minutes.

Upon receipt of additional information, the staff will complete the review of the consequences of this accident and provide our evaluation.

15.6.5 LOCAs

In FSAR Section 15.6.5, the applicant has analyzed the double-ended cold leg guillotine (DECLG) as the most limiting large-break LOCA. The analysis was done for three different flow coefficients. The results of these show that the DECLG with a Moody break discharge coefficient of 0.4 is the worst case. In this analysis, the peak clad temperature reached is 2179°F. For the small-break LOCA the applicant has determined that a cold leg rupture of less than 10-in. diameter is the most limiting. The analysis was performed for 3-in., 4-in. and 6-in.-diameter breaks. The results show that the 3-in.-diameter break is the worst case, and it results in a peak clad temperature of 1985°F. Both of these accidents are terminated by SIS and ECCS operations. Only safety-grade equipment is used to mitigate the accident.

The applicant has performed analyses of the performance of the ECCS in accordance with the Commission's regulations (10 CFR 50.46 and Appendix K to 10 CFR 50).
The analyses considered a spectrum of postulated break sizes and locations. As shown in NUREG-0390, these analyses were performed with an evaluation model that had been previously reviewed and approved by the staff. The results show that the ECCS satisfy the following criteria:

(1) The calculated maximum fuel rod cladding temperature does not exceed 2200°F.

(2) The calculated maximum local oxidation of the cladding does not exceed 17% of the total cladding thickness before oxidation.

(3) The calculated total amount of hydrogen generated from the chemical reaction of the cladding with water or steam does not exceed 1% of the hypothetical amount that would be generated if all of the metal in the cladding cylinders surrounding the fuel, excluding the cladding surrounding the plenum volume, were to react.

(4) Calculated changes in core geometry are such that the core remains amenable to cooling.

(5) After any calculated successful initial operation of the ECCS, the calculated core temperature is maintained at an acceptably low value and decay heat is removed for the extended period of time required by the long-lived radioactivity.

The staff concludes that the calculated performance of the ECCS following postulated LOCA accidents conform to the Commission's regulations and to applicable regulatory guides and staff technical positions, and the ECCS performance is considered acceptable for the postulated accidents.

15.9 TMI Action Plan Requirements

15.9.1/15.9.2

II.K.1.5 Review ESF Valve Positions, Controls, and Related Test and Maintenance Procedures To Assure Proper ESF Functioning

II.K.1.10 Review and Modify Procedures for Removing ESF From Service To Assure Operability Status Is Known

The applicant states that the intent of these two items will be met when the Operating and Maintenance Procedures are written. They are scheduled to be completed in June, 1985. The acceptability of the measures taken to satisfy these items will be evaluated when these procedures are submitted.

15.9.3 II.K.2.13 Thermal Mechanical Report: Effect of High-Pressure Injection on Vessel Integrity for Small-Break LOCA With No Auxiliary Feedwater

Staff review of this item will be covered in NRC unresolved safety issue A-49, "Pressurized Thermal Shock."

15.9.4 II.K.2.17 Potential for Voiding in the Reactor Coolant System During Transients

Westinghouse has performed a study that addresses the potential for void formation in Westinghouse-designed NSSS during natural circulation cooldown/depressurization transients. This study has been submitted to the NRC by the Westinghouse Owners Group. As stated in R. Wayne Houston's December 6, 1983 memorandum to Gus C. Lainas entitled, "Multiplant Action Item F-33, Voiding in the Reactor Coolant System During Anticipated Transients," the results of this study have been accepted.

15.9.5 II.K.2.19 Sequential Auxiliary Flow Analysis

Sequential auxiliary feedwater flow criteria are only of concern to once-through steam generator designs. Since Westinghouse has inverted U-tube steam generator designs, the analysis requested by Item II.K.2.19 is not needed for Beaver Valley Unit 2.

15.9.6 II.K.3.2 Report on Overall Safety Effect of Power-Operated Relief Valve Isolation System

As a response to Item II.K.3.2, the applicant referenced a generic Westinghouse Owners Group submittal. Should staff generic review of this material conclude otherwise, NRC will request further consideration of modification of Beaver Valley Unit 2.

15.9.7 II.K.3.3 Reporting SV and PORV Challenges and Failures

The applicant states in FSAR Table 1.10-1 that it will be responsible for ensuring that any failure of PORVs or safety valves to close will be reported promptly to the NRC and that all challenges to PORVs and safety valves will be documented in the annual report. The staff concludes that the Beaver Valley Unit 2 procedures meet the criteria of this item and are acceptable.

15.9.8 II.K.3.5 Automatic Trip of RCPs During LOCA

In response to this criterion, the applicant stated that Westinghouse performed an analysis of delayed RCP trip during LOCA. This analysis is documented and is the basis for the Westinghouse position on RCP trip (i.e., automatic RCP trip is not necessary because sufficient time is available for manual tripping of the RCPs).

Westinghouse has submitted a generic report which is under review. The applicant should state whether or not it intends to endorse this report and comply with the criteria proposed in it assuming the NRC finds it acceptable.

discussed in FSAR Section 15.6.5. However, this does not constitute a review that shows Beaver Valley Unit 2 is in full compliance with 10 CFR 50.46. After the staff's review of this evaluation model is completed a specific submittal on this issue will be required.

SALP FOR BEAVER VALLEY UNIT 2 SER

Evaluation Criteria*	Category
1. Management involvement in assuring quality	2
2. Approach to resolution of technical	2
3. Responsiveness to NRC initiatives	2
4. Staffing (including management)	N/A
5. Reporting and analysis of reportable events	N/A
6. Training effectiveness and qualification	N/A

*Ref. NRC Appendix 0516 - Systematic Assessment of Licensee Performance.

ENCLOSURE 1

JUN 18 1984

MEMORANDUM FOR: Thomas M. Novak, Assistant Director for Licensing
Division of Licensing

FROM: R. Wayne Houston, Assistant Director for Reactor Safety
Division of Systems Integration

SUBJECT: BASIS FOR BEAVER VALLEY 2 FEEDWATER ISOLATION ON HIGH STEAM GENERATOR LEVEL DESIGN REQUIREMENTS

Plant Name: Beaver Valley 2
Docket No.: 50-412
Licensing Status: OL
Responsible Branch: LB#3
Project Manager: L. Lazo
Review Branch: ICSB
Review Status: Incomplete

In Section 7.3.3.12 of the Beaver Valley 2 draft SER, ICSB expressed a concern that the design of feedwater isolation on a high steam generator level did not meet the requirements of paragraph 4.7 of IEEE-STD-279. The applicant's response to that concern, dated March 28, 1984, stated that IEEE-STD-279 is not applicable to the issue.

In response, ICSB stated, in our Licensing Position #1 for Beaver Valley 2 dated April 30, 1984, that either the design of the feedwater isolation on a high steam generator level be modified to meet the requirements of IEEE-STD-279 or an analysis be provided to show that the consequences of feedwater addition not being terminated by the high steam generator level signal are not safety significant. In a May 30, 1984, response, the applicant claims that ICSB's position is a new requirement and should be processed in accordance with NRR procedures for plant specific backfitting.

ICSB has reviewed the applicant's claim and believes that the characterization of this issue as a backfit is inappropriate. As stated in ICSB Licensing Position #1, the applicant, in Chapter 15 of the Beaver Valley 2 FSAR, takes credit for feedwater isolation on a high steam generator level signal and identifies this isolation action as an engineered safety feature actuation function. Consistent with this, the applicant, in Section 7.3 of the FSAR, has identified IEEE-STD-279 as an acceptance criterion for the feedwater isolation function and has further claimed to meet those requirements (specifically including paragraph 4.7 of IEEE-STD-279).

CONTACT: F. Burrows, ICSB
x29455

OFFICE	8406230186 XA					
SURNAME						
DATE						

C18