Duquesne Light
435 Sixth Avenue
Pittsburgh, Pennsylvania
15219

(412) 471-4300

December 14, 1978

Director of Nuclear Regulation
United States Nuclear Regulatory Commission
Attention: A. Schwencer, Chief
Branch No. 1
Division of Operating Reactors
Washington, D. C. 20555

Reference: Beaver Valley Power Station, Unit No. 1
Docket No. 50-334
Additional Information Relative to
License Amendment Request No. 25

Gentlemen:

Enclosed are three (3) signed originals and thirty-seven (37) copies of additional information relating to the referenced license amendment application. The purpose of providing this supplemental information is to clarify certain details associated with several of the requested changes.

An analysis has been performed to verify that the relaxation of Low Feedwater Flow Reactor Trip Setpoint is acceptable. This analysis is attached. Additionally, modifications will be made to the recirculation lines of the Auxiliary Feedwater System. These changes and the proposed Technical Specifications in this request relating to the Auxiliary Feedwater System will assure that the relaxation in the Low Feedwater Flow Setpoint will not result in an accident more severe than those which have been analyzed in the original plant design.

The elimination of the part length rods has been engineered by the Westinghouse Electric Corporation, the reactor supplier. The purpose of removing the part length rods (which we are restricted from utilizing) is to minimize future refueling outage time associated with unlatching and relatching these rods from the lead screw. This time saving is accompanied by a reduction in radiation exposure that would result from the performance of this work. To assure that no thermal hydraulic problems or no changes in upper head fluid temperature will result from removing these rods, a thimble plugging device is inserted in the fuel elements that occupy the core positions presently associated with part length rods. An anti-rotation device is installed on each part length rod lead screw. These leads are left in place.

P

Three (3) copies of Westinghouse Electric Corporation proprietary sketches of the stainless steel anti-rotation devices are being provided with the three (3) signed originals of this submittal.

The deletion of the augmented start-up program requirements and special test exceptions is requested to avoid possible future misapplication or misuse of this verbage. The augmented start-up program, which was conducted to gather special peaking factor data associated with operation under the constant axial offset control mode, has been completed. The Westinghouse Electric Corporation has submitted a WCAP which summarizes the results of this special test effort.

The removal of the APDMS requirement. This change requests that the requirement to immediately reduce core power levels in the event that allowable values for F_{xy} are exceeded be eliminated. Discussions with representatives of the Westinghouse Electric Corporation have disclosed that the requirement to reduce power levels is only applied in situations where the allowable value of F_Q are less than 2.32. The requirement to utilize an Axial Power Distribution Monitoring System is also applied to plants with an allowable F_Q of less than 2.32. Since we have previously successfully applied to remove from the Beaver Valley Technical Specifications all references to the APDMS, this APDMS related ACTION requirement should have been included in our request at that time.

At this time, we would like to clarify the manner in which we have submitted all proposed Technical Specifications associated with T.S. 3.2.2 and T.S. 3.2.3, since we have pending other Technical Specification change requests associated with these same specifications. We have attempted to provide in each case the proposed manner in which the existing Technical Specifications should be changed for the particular request under review. As soon as one of these requests is granted and prior to issuance of the then remaining request, we shall contact you and, if necessary, revise the outstanding change request.

We would like to take this opportunity to provide the basis on which we have previously requested a reevaluation of the need to consider burn up related restriction on F_QT, which is referenced in the staff safety evaluation for Beaver Valley License Amendment No. 9. That evaluation states that rod bow linear power penalties are not required at average burn ups less than 24,000 MWD/MTU.

Westinghouse proprietary submittal NS-TMA-1760 dated April 19, 1978 concludes that no penalty on F_QT is required with burn ups as great as 60,000 MWD/MTU. The maximum region average burn up presently anticipated at Beaver Valley is 33,000 MWD/MTU (Reference FSAR Section 3.3.1.1). This number is below the values presented in the Westinghouse Report and, therefore, further consideration of possible penalties on F_QT should not be required.

Very truly yours,

C. N. Dunn
Vice President, Operations

Attachment

(CORPORATE SEAL)

Attest:

H W Staas

H. W. Staas
Secretary

COMMONWEALTH OF PENNSYLVANIA)
) SS:
COUNTY OF ALLEGHENY)

On this 18th day of December, 1978, before me, DONALD W. SHANNON, a Notary Public in and for said Commonwealth and County, personally appeared C. N. Dunn, who being duly sworn, deposed, and said that (1) he is Vice President of Duquesne Light, (2) he is duly authorized to execute and file the foregoing Submittal on behalf of said Company, and (3) the statements set forth in the Submittal are true and correct to the best of his knowledge, information and belief.

Donald W. Shannon

DONALD W. SHANNON, Notary Public
Pittsburgh, Allegheny Co., Pa.
My Commission Expires
June 7, 1979

EVALUATION OF THE
LOW FEEDWATER FLOW REACTOR TRIP SETPOINT
CHANGE FOR BEAVER VALLEY UNIT NO. 1

Prepared by:

D. C. Wood
Reactor Protection Analysis I
PWR - Nuclear Safety

Low Feedwater Flow Reactor Trip Setpoint Change for Beaver Valley Unit No. 1

INTRODUCTION

The low feedwater flow reactor trip consists of a steam/feedwater flow mismatch in coincidence with low steam generator water level. A logic diagram for this trip function for a three loop plant is presented in Figure 1. This reactor trip is designed to protect the plant from a sudden loss of the reactor's heat sink, such as a postulated loss of normal feedwater or a major rupture of a main feedwater pipe. At low power levels, the possibility of an unintentional reactor trip from low feedwater flow is greatly enhanced due to manual control of main feedwater flow, minimal margin between programmed steam generator water level and the trip setpoint, and larger steam and feedwater flow measurement inaccuracy. Improvement in operability may be attained by changing any of these three factors affecting the possibility of unintentional reactor trips. The following evaluation investigates increasing the margin to trip through changes to the low feedwater flow reactor trip setpoints, while assuring that adequate protection is provided for loss of heat sink accidents.

BASIS OF EVALUATION

The "Major Rupture of a Main Feedwater Pipe" analysis presented in Section 14.2.5.2 of the BVPS No. 1 FSAR is the only safety analysis which directly assumes actuation of the Low Feedwater Flow reactor trip. For this reason, the evaluation of setpoint changes is made to verify adequate protection is available for a hypothetical rupture of a main feedwater pipe. A general discussion of the feedline break analysis follows to serve as a basis for subsequent sections discussing setpoint considerations.

The Major Feedwater Pipe Rupture is analyzed to address a break in the feedwater pipe large enough to prevent addition of sufficient feedwater to the steam generators to maintain shell-side fluid inventory in the steam generators. The following assumptions have been made in the Beaver Valley Unit No. 1 analysis:

1) The break occurs between the feedline check valve and the steam generator. Breaks upstream of the check valves behave similarly to a loss of normal feedwater, and are covered by that analysis.

2) The single failure assumed is a safeguards actuation train, resulting in the failure of a motor driven auxiliary feedwater pump. Furthermore this failure renders auxiliary feedwater isolation valves for that train inoperable from the control room.

3) All auxiliary feedwater will spill to the feedline break until the operator successfully isolates all auxiliary feedwater pumps from the break. This action is assumed to be completed ten minutes after the reactor trip occurs.

4) The worst break area is determined as the area which causes lo-lo steam generator level to occur in the broken loop simultaneously with the low feedwater flow (lo steam generator level in coincidence with steam/feedwater flow mismatch) reactor trip signal. This area minimizes heat removal capability while maximizing stored heat prior to reactor trip.

5) Loss of offsite power is assumed to occur after reactor trip and reactor coolant flow decreases to natural circulation. Natural circulation flow is assumed to continue throughout the entire course of the transient.

6) Steam generator heat transfer area is assumed to decrease as the shell-side liquid inventory decreases.

7) The steam/feedwater flow mismatch would be generated immediately after the break occurs. The low feedwater reactor trip signal would be generated when the low level setpoint is reached, providing the input to the coincidence logic shown in Figure 1.

Table 1 lists the significant events which take place during the course of the hypothetical feedline rupture transient. During the initial phase of the transient prior to reactor trip the decreasing steam generator inventory causes a rise in reactor coolant temperatures and pressures (see figure 2 and figure 3). At 17.5 seconds the reactor trip circuits are actuated upon receipt of both lo-lo steam generator level in the broken loop and low feedwater flow in the intact loops. After reactor trip the core power decreases more rapidly than the steam generator heat transfer, resulting in a short period of cooldown in the Reactor Coolant System. As steam generator heat transfer drops below the residual and decay heat power the reactor coolant system heatup and pressurization begin, ultimately resulting in safety valve relief from both the pressurizer and the steam generators. Water is relieved from the pressurizer safety valves after the pressurizer fills. When auxiliary feedwater flow is established (the operator isolates auxiliary feedwater pumps from the break) the fluid inventory in the intact steam generators begins to recover. Eventually the increasing steam generator heat transfer will surpass the decreasing core decay heat source, and the temperatures and pressure in the reactor coolant system will begin to decrease. As shown in Table 1, this time is calculated to be approximately 2050 seconds after the break occurs. The consequences of the feedline rupture are considered acceptable if sufficient RCS inventory

≈ 34 min

is left to cover the core and if peak RCS pressure does not exceed the design limits. A more complete description of the transient can be found in the FSAR.

SETPOINT CONSIDERATIONS

The major considerations in obtaining operating margin from protection system setpoints are the actual margin to safety limits and the errors which must be assumed between the nominal setpoint and the setpoint at which the function may be assumed to have been actuated. Safety margin is determined through evaluation or reanalysis of the safety analysis. Errors, however must be assessed independently, considering contributions from process variable measurement uncertainty, instrument calibration and drift allowance, and additional errors resulting from an adverse environment if such an environment can result from the transient for which the protective function is necessary. These are discussed in more detail for the low feedwater flow reactor trip setpoints as they apply to the main feedline rupture analysis for Beaver Valley Unit No. 1.

- Low Steam Generator Water Level Setpoint

  The consequences of the main feedline rupture are made more severe by any change which either increases the heat generated by the reactor coolant system or decreases the ability of the steam generators to remove stored heat. Lowering the low steam generator water level setpoint to enhance operating margin would decrease the total steam generator mass at the time of trip and would degrade the ability of the intact steam generators to remove heat at an earlier time in the transient. Considering the appropriate errors in the setpoints used in the safety analysis, no increase in operating margin at the expense of safety margin can be justified by changing the low steam generator level setpoint. The nominal setpoint must remain at 25% of span as is currently required by the Beaver Valley Unit No. 1 Standard Technical Specifications.

- Steam/Feedwater Flow Mismatch Setpoint

  Assuming feedline break conditions prior to reactor trip, the only requirement placed upon the steam/feedwater flow mismatch channel is that the signal is present when the low steam generator level setpoint is reached. Because the safety margin will remain unchanged as long as this requirement is satisfied, some operating margin can be realized by raising the steam/feedwater flow mismatch setpoint. Including a sufficient error allowance for the process variable measurement, drift, calibration, and environmental effects, the nominal setpoint may be increased to 70% of steam flow at the licensed power level of 2660 MWt. For the case analyzed, the mismatch would be actuated

immediately. Mismatches less than the setpoint which would occur during a hypothetical feedline break can only be the result of a large amount of feedwater flow entering the intact steam generators resulting in a significant benefit in total heat removal capability.

CONCLUSIONS

The steam/feedwater flow mismatch reactor trip setpoint may be increased to 70% of steam flow at the licensed power level at 2660 MWt. No reduction in the low steam generator water level reactor trip setpoint can be justified at the expense of safety margin for the hypothetical major rupture of a main feedwater line. Attachment 1 to this report shows changes to Table 2.2-1 of the Beaver Valley Unit No. 1 Standard Technical Specification to be consistent with the recommended setpoints changes. Attachment 2 shows similar changes to the Precautions Limitations and Setpoints document.

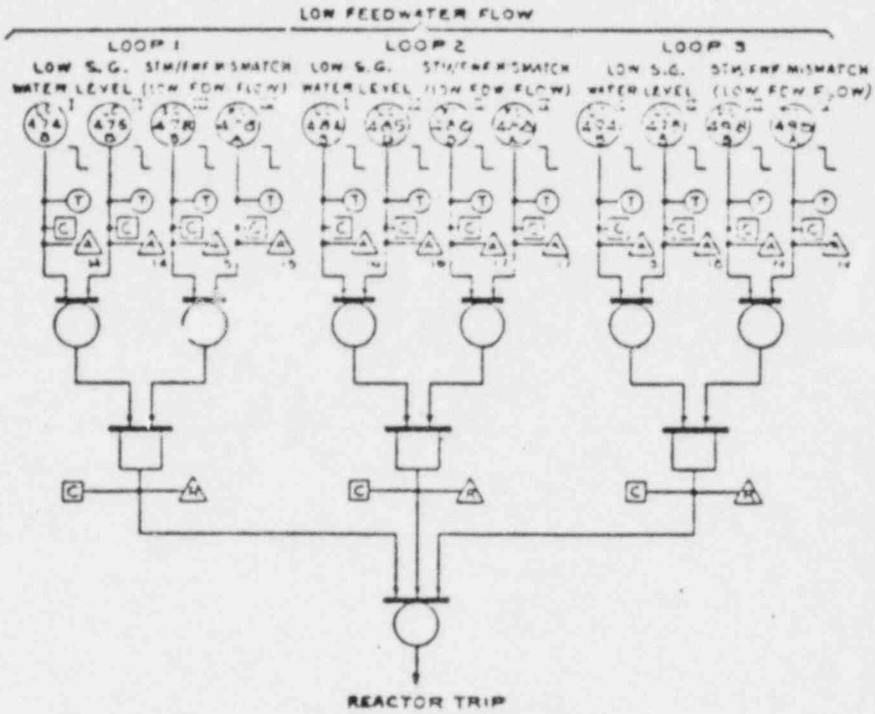

FIGURE 1 Low Feedwater Flow Reactor Trip Logic
Beaver Valley Power Station Unit No. 1

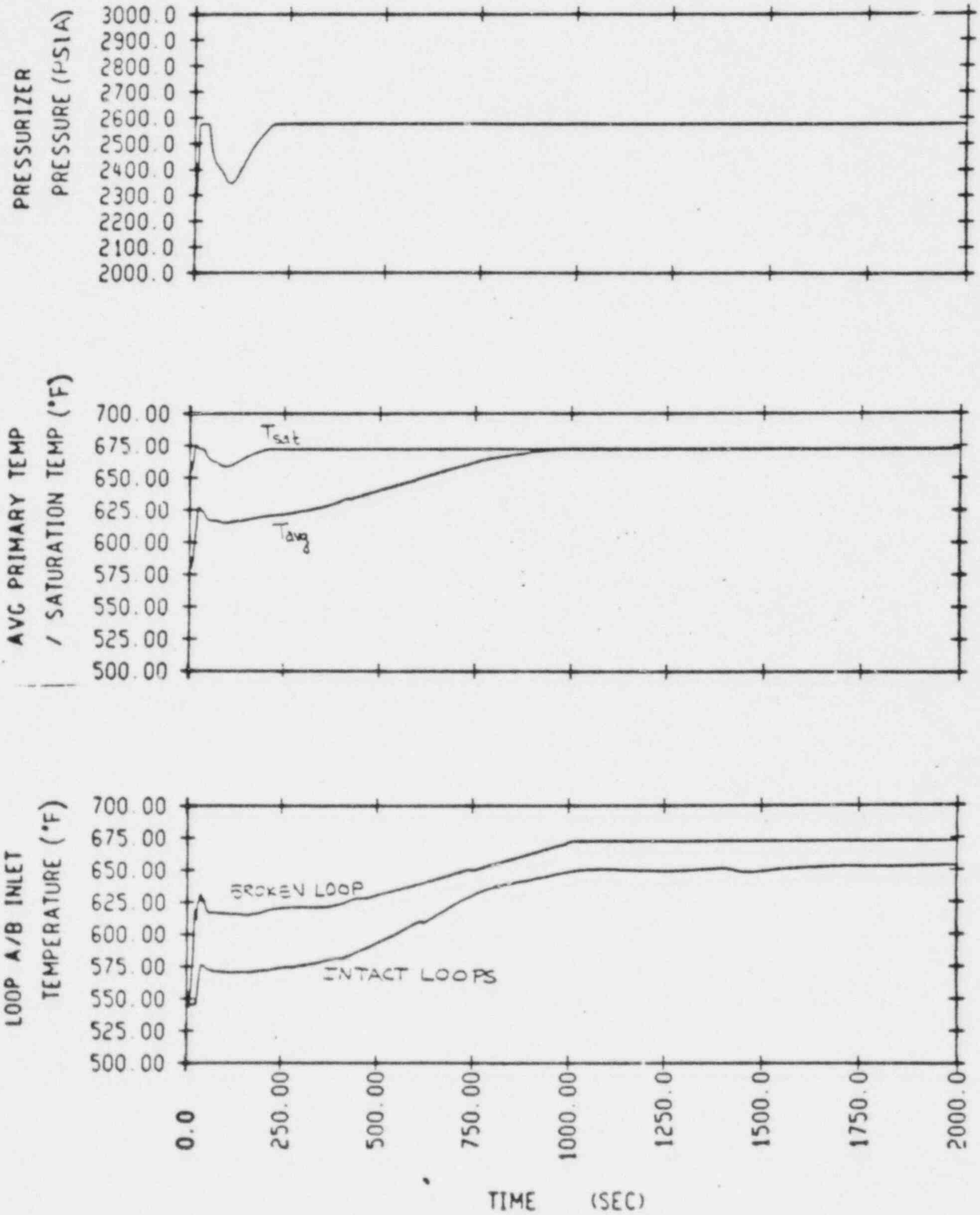

FIGURE 2 Pressurizer Pressure , Average Primary Temperature, and Inlet Temperature versus Time

Beaver Valley No. 1 Main Feedline Rupture

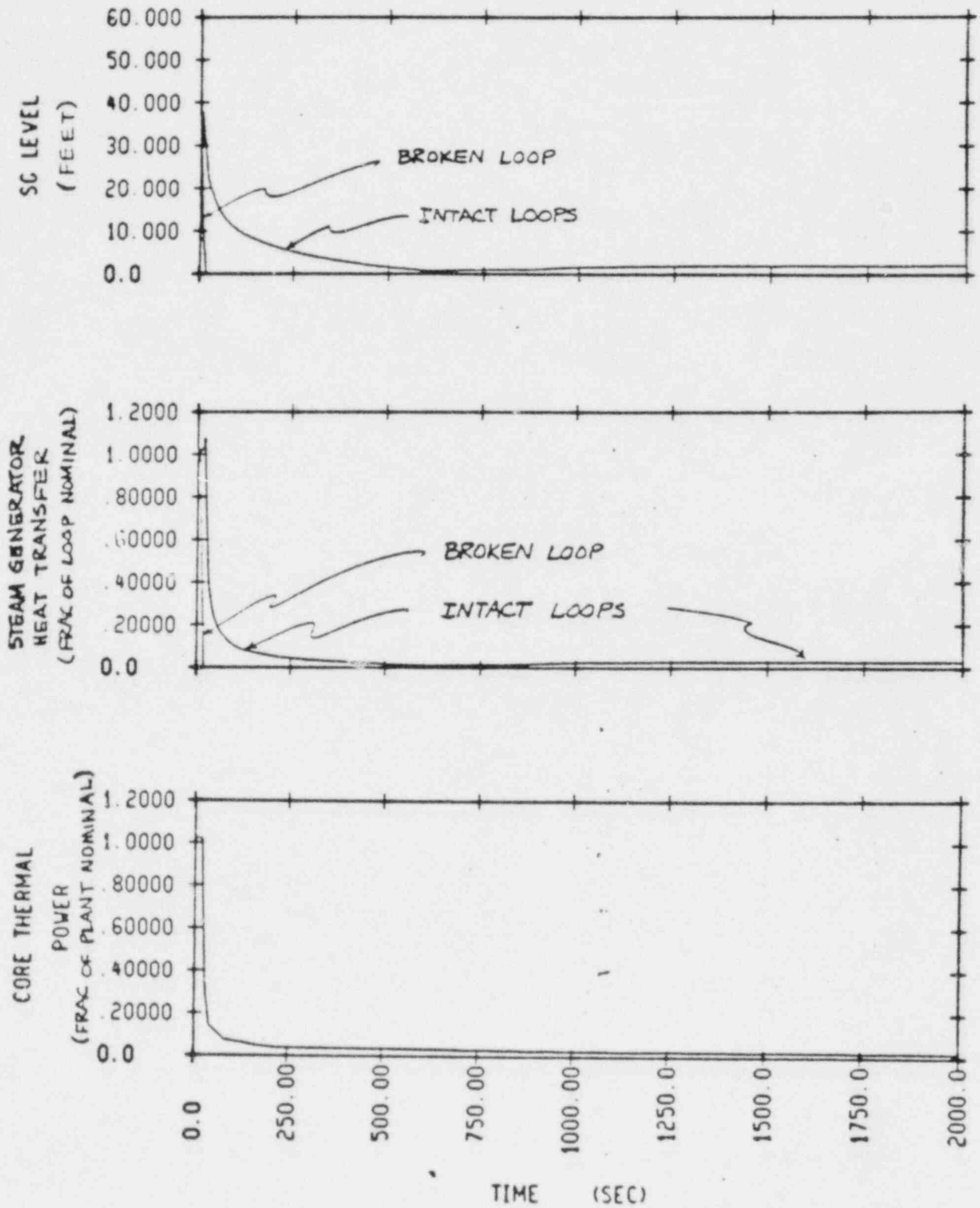

FIGURE 3 Steam Generator Level, Steam Generator Heat Transfer, and Core Thermal Power versus Time

Beaver Valley No. 1 Main Feedline Rupture

TABLE 1

SEQUENCE OF EVENTS FOR MAJOR RUPTURE OF A MAIN FEEDWATER PIPE

BEAVER VALLEY UNIT NO. 1

EVENT	TIME(SEC)
Feedline Break Occurs	0.0
High pressure setpoint reached (no credit is taken for this function in the analysis)	9.5
Reactor trip setpoint reached (lo-lo steam generator level and low feedwater flow)	17.5
Rods begin to fall	19.5
Peak steam relief from pressurizer safety valves	20.0
Steam generator safety valves actuated	30.0
Pressurizer safety valves actuated	210.0
Pressurizer filled with water	414.0
Alignment of auxiliary feedwater system completed by operator	630.0
Auxiliary feedwater flow commences to intact steam generators	630.0
Boiling occurs in the reactor coolant system	744.0
Hot water purged from main feedwater lines	869.0
Core decay heat decreases to auxiliary feedwater heat removal capacity (at this time the transient is essentially considered to have terminated as temperatures and pressures begin to decrease)	2055.

TABLE 2.2-1 (Continued)

REACTOR TRIP SYSTEM INSTRUMENTATION TRIP SETPOINTS

FUNCTIONAL UNIT	TRIP SETPOINT	ALLOWABLE VALUES
13. Steam Generator Water Level--Low-Low	≥ 10% of narrow range instrument span-each steam generator	≥ 9% of narrow range instrument span-each steam generator
14. Steam/Feedwater Flow Mismatch and Low Steam Generator Water Level	< 70% ~~40%~~ of full steam flow at RATED THERMAL POWER coincident with steam generator water level ≥ 25% of narrow range instrument span--each steam generator	< 72.5 ~~42.5%~~ of full steam flow at RATED THERMAL POWER coincident with steam generator water level ≥ 24% of narrow range instrument span--each steam generator
15. Undervoltage-Reactor Coolant Pumps	≥ 2750 volts-each bus	≥ 2725 volts-each bus
16. Underfrequency-Reactor Coolant Pumps	≥ 57.5 Hz - each bus	≥ 57.4 Hz - each bus
17. Turbine Trip		
A. Auto stop oil Pressure	45 psig	± 5 psig
B. Turbine Stop Valve	≥ 1% open	≥ 1% open
18. Safety Injection Input from ESF	Not Applicable	Not Applicable
19. Reactor Coolant Pump Breaker Position Trip	Not Applicable	Not Applicable

C. High Pressurizer Pressure

(PC-455A, PC-456A, PC-457A) 2385 psig

D. High Pressurizer Water Level

(LC-459A, LC-460A, LC-461A) 92% of span

E. Low Pressurizer Pressure

(PC-455C, PC-456C, PC-457C)
(PM-455A, PM-456A, PM-457A)

trip setpoint	1945 psig
lead time constant	10 sec.
lag time constant	1 sec.

F. Loss of Primary Coolant Flow

(FC-414, FC-415, FC-416)
(FC-424, FC-425, FC-426)
(FC-434, FC-435, FC-436)

low flow (flow channel is calibrated at full load flow and temperature such that 100% = normal flow)	90%
low frequency	57.5 Hz
low voltage	2750 volts
undervoltage time delay	0.5 second

G. Loss of Feedwater

1. Low-low steam generator water level

(LC-474A, LC-475A, LC-476A)
(LC-484A, LC-485A, LC-486A) 10% of span
(LC-494A, LC-495A, LC-496A)

2. Coincident low level and steam/feedwater flow mismatch

low level 25% of span

(LC-474B, LC-475B)
(LC-484B, LC-485B)
(LC-494B, LC-495B)

mismatch 2.709 ~~1.549~~ $\times 10^6$ lbs/hr.*

(FC-478A, FC-478B)
(FC-488A, FC-488B)
(FC-498A, FC-498B)

*assumes steam flow at 2660 MWt is 3.87×10^6 lbm/hr per loop.

12401-1

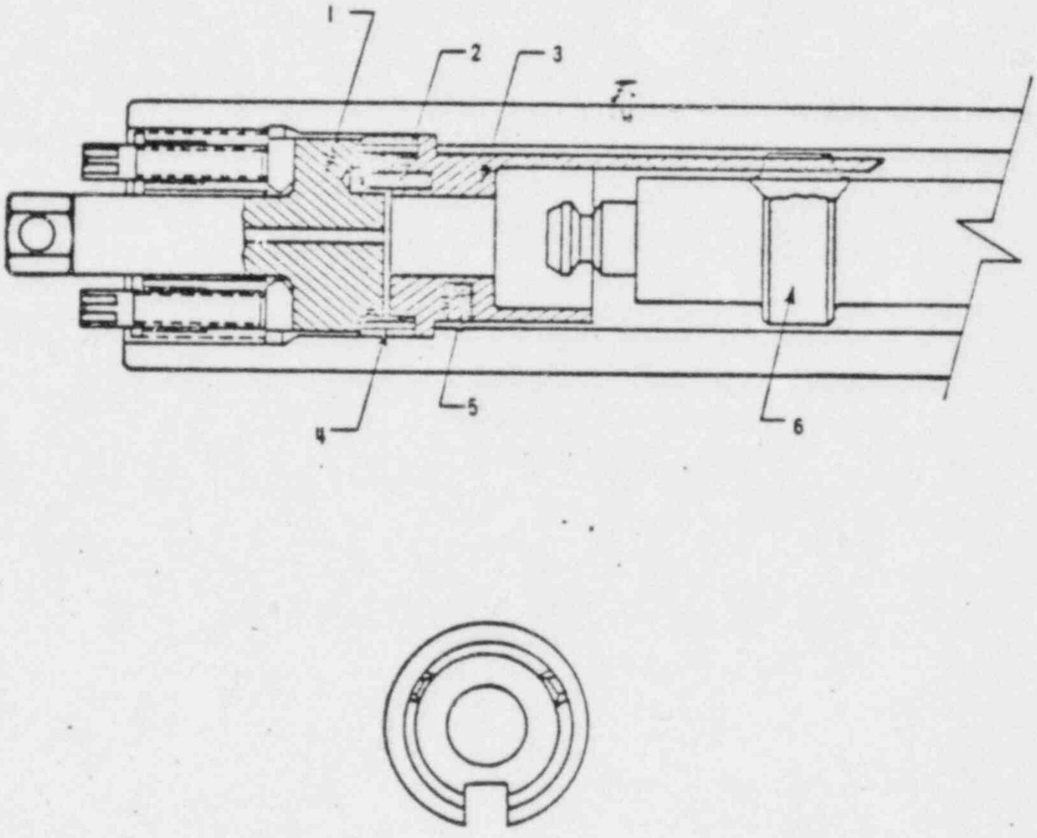

Figure 1
Part Length CRDM - Rollernut Antirotation Housing Details and Assembly

12401-2

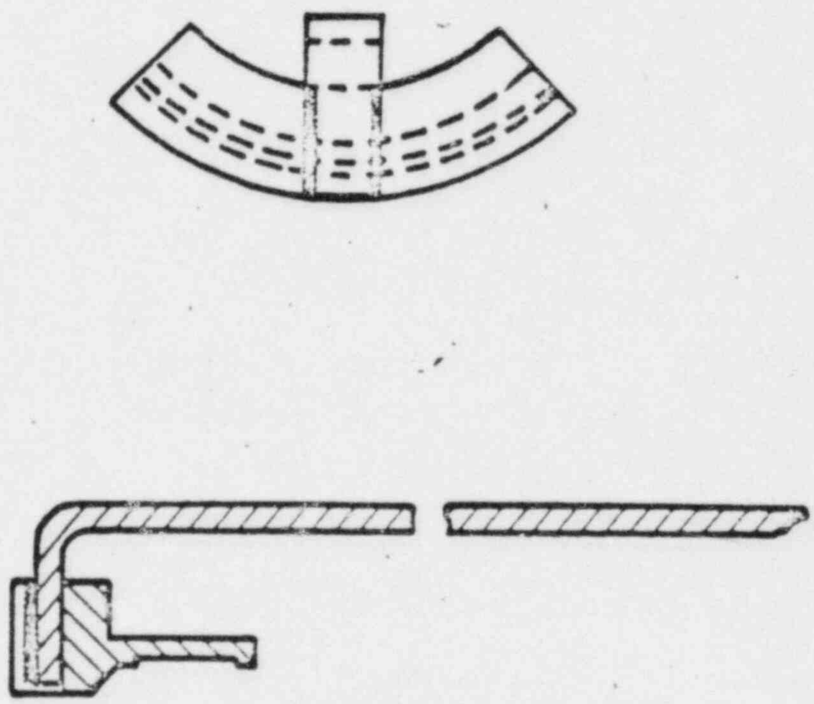

Figure 2
Part Length CRDM - Rollernut Up Position Lead Screw Clamp