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October 8, 1979

United States Nuclear Regulatory Commission
Attention: Harold R. Denton, Director
Office of Nuclear Reactor Regulation
Washington, D. C. 20555

Reference: Beaver Valley Power Station, Unit No. 1
Docket No. 50-334
License No. DPR-66
Response To NRC Letter Dated September 17, 1979

Dear Mr. Denton:

Enclosed are three (3) signed originals and thirty-seven (37) copies of our response to your September 17, 1979 letter on the subject of a "potential unreviewed safety question on interaction between non-safety grade systems and safety grade systems". This potential problem was also addressed in IE Information Notice 79-22, issued September 14, 1979.

In conjunction with Westinghouse Electric Corporation, we have reviewed the specific non-safety grade systems listed in IE Information Notice 79-22 as well as other systems for the potential interactions that could constitute a substantial safety hazard. We have not been able to identify such an interaction. While, in some cases, we have identified variations from the FSAR licensing bases, the basic conclusion of the FSAR (that these events do not constitute an undue risk to the health and safety of the public) remains unchanged.

The Nuclear Safety Analysis Committee (NSAC) has determined that the probability of severe consequences resulting from one of these high energy line breaks is v[illegible] low. Further, such breaks are more likely to be small cracks rather than [illegible]pt failures so that the resulting adverse environment builds up over a per[illegible] [illegible] time providing the potential for detection prior to component failure. Additionally, our review recognized the difference between a demonstrated deficiency (e.g. determination that a control component would operate in a fashion not within the limits presented in the safety analysis) and a potential, unreviewed question. As previously stated, we have not identified any events that would change the conclusions of the FSAR, i.e. that these events do not constitute an undue risk to the health and safety of the public.

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As you must recognize, our investigation within the limited time frame required by your September 17 letter must be considered preliminary and could not include detailed evaluations. Generic evaluations, coupled with plant-specific detailed evaluations (where required), are proceeding. The general scope of these evaluations is outlined in Attachment A. However, continued operation is warranted while these detailed evaluations proceed.

As a result of the Three Mile Island accident, there are a significant number of industry, governmental and regulatory investigations underway examining the licensing bases and the operating procedures of nuclear generating facilities. These investigations are already identifying areas where studies may result in the consideration of new or revised events as part of the bases for assuring continued safety of nuclear plants. NUREG-0578 outlines several such events and suggests remedies.

NUREG-0578 requirements for analyses of potential safety problems envisions the kinds of scenarios identified by Westinghouse and made the subject of IE Information Notice 79-22. NUREG-0578, Section 3.2, Page 17, states, in part:

> "...The NRC requirements for non-safety systems are generally limited to assuring that they do not adversely affect the operation of safety systems..."

Further, on page A-45 of NUREG-0578, it is stated that "consequential failures shall also be considered..."

We, therefore, believe that the scope of the action required by IE Information Notice 79-22 is fully consistent with the requirements of NUREG-0578 and should, therefore, be integrated with the planned response sequence for compliance with the NUREG.

Very truly yours,

C N Dunn

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 8TH day of OCTOBER, 1979, 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 COUNTY
MY COMMISSION EXPIRES JUNE 7, 1983
Member, Pennsylvania Association of Notaries

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DUQUESNE LIGHT COMPANY
Beaver Valley Power Station
Unit No. 1

ATTACHMENT A
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION
ON THE ENVIRONMENTAL INTERACTION ISSUE

Summary

This letter is in response to an NRC Staff communication dated September 17, 1979 entitled "Potential Unreviewed Safety Question on Interaction Between Non-Safety Grade Systems and Safety Grade Systems." The NRC letter required information to be provided within 20 days by all operating light water reactors to enable the Staff to determine the applicability of the potential unreviewed safety questions. The information contained in this letter justifies continued operation of the Beaver Valley Power Station Unit #1 on the basis of the improbability of the postulated scenarios as they apply to Beaver Valley Power Station, the acceptability of the consequences, and the commitments made to resolve these issues. Schedules for short and long-term commitments to resolve the environmental interaction issues are proposed to be consistent with the schedules established by the Staff in NUREG-0578.

Scope

On September 18, 1979 Westinghouse presented to the Staff a summary of the investigation that had been conducted which led to the identification of four (4) potential interaction scenarios where the effect of adverse environments, resulting from high energy line breaks, on control systems could lead to consequences more limiting than the results presented in the Safety Analysis Report. The four potential interaction scenarios are:

1. Steam generator power operated relief valve control system
2. Pressurizer power operated relief valve control system
3. Main feedwater control system
4. Automatic rod control system

Probability of Postulated Interactions

Implicit in the four (4) potential interaction scenarios identified by Westinghouse are worst case assumptions concerning the break size and location, and the type and extent of consequential failures in control systems induced by the adverse environment. These assumptions are therefore in addition to the already conservative set of assumptions ascribed to the analysis of the Design Basis Events reported in the Safety Analysis Report. It follows that these scenarios represent a significantly less probable subset of the Design Basis Events that are dependent on the occurrence of additional events, each having an associated uncertainty of occurring. While no quantitative analysis has been conducted concerning the improbability of overall scenarios, the attachments define, for each of the scenarios identified above for the Beaver Valley Power Station, the conservative assumptions already contained in the Design Basis Event analysis reported in the Safety Analysis Report and the additional conservative assumptions to be made to derive the postulated interaction scenario.

Probability of Postulated Interaction (continued)

As can be seen from the attachments, for each of the scenarios considered, the improbability of all the additional set of assumed conditions occurring simultaneously, over and above the already low probability of the Design Basis Event itself, leads to the conclusion that continued operation of the Beaver Valley Power Station can be justified until recommended solutions to these low probability event scenarios can be implemented.

With regard to the probability of any single design basis event initiating, via the adverse environment, failures in several control systems, it again can be noted from the attachments that the probability of all the additional set of conditions occurring simultaneously for more than one scenario is of an even lower order of magnitude than for each individual scenario. Furthermore, implementation of the proposed solutions for the individual scenarios will, as a consequence, address any concern for multiple interactions from a single initiating Design Basis Event.

Due to the implementation in the design of the electrical separation requirements between control and protection systems specified in IEEE-279, the only interaction mechanisms identified in the above scenarios result from conservatively assuming an adverse environment at the location of the control systems and the consequential equipment failure in the worst direction. As a consequence, it can be anticipated that any interaction scenarios yet to be identified, in as yet unreviewed control systems, will be no more probable than the particular scenarios described by Westinghouse.

Consequences of Postulated Interactions

In lieu of performing a plant specific analysis in an effort to address each of the potential postulated interactions involving a feedline break, Westinghouse has referred to bounding accident analyses that have been submitted to the NRC in WCAP-9600, Report on Small Break Accidents for Westinghouse NSSS. Section 4.2 of the report provides transient results following a total loss of main and auxiliary feedwater. Sensitivity studies as a function of time of auxiliary feedwater initiation and opening of the pressurizer power operated relief valves are presented following the initial transient. Calculations have been performed to show that the consequences following the control interaction scenarios for the steam generator PORV control system, main feedwater control system and pressurizer PORV control system are in fact bounded by the analyses in WCAP-9600. For all accident scenarios, the calculations indicated that the operator need not take corrective action to mitigate the consequences for at least 30 minutes following initiation of the event.

A typical analysis has been performed to address the rod control system interaction scenario. The results of the analysis indicate that no fuel damage occurs and the consequences are within the assumptions made in the Safety Analysis Reports.

DUQUESNE LIGHT COMPANY
Beaver Valley Power Station
Unit No. 1

ATTACHMENT I
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION
ON THE ENVIRONMENTAL INTERACTION ISSUE

Steam Generator PORV Control System

Summary Of Postulated Scenario

Following a feedline rupture outside containment in the main steam valve room, the steam generator PORVs are assumed to experience a consequential failure due to an adverse environment. Failure of the PORVs in the OPEN position results in the depressurization of multiple steam generators which are the source of steam supply for the turbine-driven auxiliary feedwater pump. Eventually, the turbine-driven auxiliary feedwater pump will not be capable of delivering auxiliary feedwater to the intact steam generators. A potential exists that no auxiliary feedwater will be injected into the intact steam generators until the operator takes corrective action to isolate the auxiliary feedwater flow spilling out the rupture.

Probability - Assumptions Affecting Event Probability And Consequences

a. Standard Safety Analysis Report Assumptions Concerning Feedline Break:

- Conservative initial assumptions

  * Appendix K decay heat model

  * Engineered safeguards power plus calorimetric error

  * Programmed RCS temperature plus control deadband and instrument errors

  * Initial conservative S/G inventory

  * Conservative core physics

- Conservative accident assumptions

  * Break (all sizes) in Safety Class 2 feedline piping

  * Maximum adverse environmental errors for protective instrumentation

  * Worst single active failure (loss of one motor-driven auxiliary feed pump)

  * Operator action time

b. Additional Assumptions Required For This Scenario:

- Break must occur outside containment between the penetration and feedline check valve. However, at Beaver Valley, this distance is only about thirty (30) inches in each of the three (3) loops.

  The feedwater piping in the three (3) loops at Beaver Valley is 16 inch, schedule 80, seamless A106 Gr B carbon steel with a wall thickness of .843 inches. As part of the steam generator nozzle crack problem identified in IE Bulletin No. 79-13, the station was required to radiographically examine a portion of the feedwater lines outside the containment wall but downstream of the main feedwater check valves. This is the thirty (30) inch area mentioned above. One (1) weld and one (1) entire sixteen (16) inch band of feedwater piping were radiographically inspected using an iridium source and 2T sensitivity. This inspection was performed in accordance with ASME Section III, Subsection NC, Article NC-5000. All radiographs were judged to be acceptable prior to resuming power operation.

- Adverse environment resulting from the rupture can impact the steam generator PORV control systems associated with the ruptured loop and the intact loops. The PORVs are air-to-open valves with the air supply being controlled by solenoid-operated valves. The PORVs are opened on high steam pressure in the main steam header. Both the PORVs and the SOVs are located in the main steam valve room. The remainder of the control circuit is external to the main steam valve room.

- The single active failure is a motor-driven auxiliary feed pump. The loss of the turbine-driven auxiliary feed pump as the single active failure or no active failure would invalidate the postulated scenario.

  The auxiliary feedwater system at Beaver Valley uses one steam-driven 700 gpm centrifugal pump and two 350 gpm electric-driven centrifugal pumps, as an emergency source, to supply feedwater to the steam generators from the Primary Plant Demineralized Water Storage Tank or river water system. This system automatically starts on a loss of normal station power or on a safety injection signal. Auxiliary feedwater is supplied to each steam generator through two redundant supply headers, each containing a motor-operated throttle valve. The supply headers join downstream of the throttle valves and flow through a flow measuring device, a check valve, and a containment isolation valve, before connecting with the main feed line downstream of the main feed line containment isolation.

- Due to the adverse environment, the steam generator PORV control system initiates a spurious signal to open the PORVs. Should the control system continue to operate within specification or initiate a spurious signal to close the PORVs, the scenario is invalidated. The PORVs are designed to fail closed on loss of air and power at the Beaver Valley Power Station.

- PORV on steam generators supplying steam to turbine-driven auxiliary feed pump is assumed to open as a result of spurious signal. If this PORV is not affected or fails closed, the scenario is invalidated.

Accident Consequences

Section 4.2 of WCAP-9600, Report on Small Break Accidents for Westinghouse NSS Systems, describes transient analyses for postulated loss of all main and auxiliary feedwater (no pipe rupture). The results indicate that the operator has at least 4,000 seconds following the loss of all feedwater to reinitiate auxiliary feedwater flow to the steam generators before the core begins uncovering.

The interaction scenario postulated above is similar to that presented in Section 4.2 of WCAP-9600. The only additional assumption made is that a feedline rupture occurs outside containment between the containment penetration and the feedline check valve. Conservatively assuming that all liquid inventory in the steam generator associated with the ruptured feedline is lost via the rupture without removing any heat (i.e., liquid blowdown), calculations have shown that the heat removal capability of the liquid inventory blowdown requires operator action 1200 seconds earlier than reported in WCAP-9600. Thus, if a feedline rupture is assumed coincident with the analyses performed in WCAP-9600 the operator still has at least 2800 seconds to take corrective action to inject auxiliary feedwater into the intact steam generators. No Safety Analysis Reports assume greater than 30 minute operator action following a feedline rupture.

Recommended Solution

The operator will be alerted to the possibility of the steam generator PORVs failing in the open position following a secondary high energy line rupture outside containment in the main steam valve room. The operator will be cautioned that the turbine-driven auxiliary feedwater pump could potentially be lost due to loss of steam supply and he can only rely upon the motor-driven auxiliary feedwater pumps to supply the minimum feedwater requirements following a secondary line rupture.

DUQUESNE LIGHT COMPANY
Beaver Valley Power Station
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ATTACHMENT II
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION
ON THE ENVIRONMENTAL INTERACTION ISSUE

Main Feedwater Control System

Summary Of Postulated Scenario

Following a small feedline rupture the main feedwater control system malfunctions in such a manner that the liquid mass in the intact steam generators is less than for the worst case presented in Safety Analysis Reports. The reduced secondary liquid mass at time of automatic reactor trip results in a more severe reactor coolant system heatup following reactor trip.

Probability - Assumptions Affecting Event Probability And Consequences

a. Standard Safety Analysis Report Assumptions Concerning Feedline Break:

- Conservative initial assumptions
  * Appendix K decay heat model
  * Engineered safeguards power plus calorimetric error
  * Programmed RCS temperature plus control deadband and instrument error
  * Initial conservative S/G inventory
  * Conservative core physics
- Conservative Accident Assumptions
  * Break (all sizes) in Safety Class 2 feedline piping
  * Maximum adverse environmental errors for protective instrumentation
  * Worst single active failure (loss of any one auxiliary feed pump)
  * Operator action time

b. Additional Assumptions Required For This Scenario:

- Break must occur between S/G nozzle and feedline check valve.

  Included in Attachment I of this document are the feedwater piping physical characteristics as well as the radiographic parameters used for weld/piping inspection outside containment. The same piping physical characteristics as well as radiographic parameters apply to the feedwater piping inside containment. During the radiographic inspection performed in response to IE Bulletin No. 79-13, several weld-related defects were detected.

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These defects were removed, welds repairs performed and additional radiographic inspection completed. All feedwater line weldments (with the exception of the nozzle to steam generator weld) in containment have undergone radiographic analysis in accordance with IE Bulletin 79-13 with all welds judged to be acceptable.

- Small breaks less than 0.2 sq ft. Larger breaks invalidate the scenario.

- Adverse environment resulting from the break can impact both the main feedwater control systems associated with the broken loop and the intact loops. The control circuit for the feedwater regulating valves utilizes a combination of steam and feedwater flow, steam generator level and turbine impulse pressure for controlling the position of the valve. During the postulated scenario the only instrument that could be affected would be the steam flow transmitters which are located inside of containment. The remainder of the control system is external to containment.

- Due to the adverse environment the main feedwater control system initiates a spurious signal to close the feedwater control valves (FCV) in the intact loops. Should the control system continue to operate within specification the scenario is invalidated.

Accident Consequences

Section 4.2 of WCAP-9600, Report on Small Break Accidents for Westinghouse NSSS System, describes transient analyses for a postulated loss of all main and auxiliary feedwater (no pipe rupture). Following a loss of all main and auxiliary feedwater, the operator is not required to take action for at least 4,000 seconds following the loss of all feedwater to prevent the core from uncovering. With a feedline rupture assumed coincident with the assumptions made in WCAP-9600, the operator continues to have at least 2800 seconds before corrective action must be taken to inject auxiliary feedwater into the intact steam generators to prevent core uncovering. No Safety Safety Analysis Reports assume greater than 30 minute operator action following a feedline rupture.

Recommended Solution

To ensure that the operator is aware of this possible control system environmental interaction, the system transient characteristics following a small feedline rupture with and without feedwater control system operation will be reviewed by the operator.

The general system characteristics following a small feedline rupture would be the following: a slowly decreasing indicated water level in at least one steam generator, a resultant opening of the associated feedwater control valve, and a corresponding increase in main feedwater flow. One or more of the above trends would be indicative to the operator that a small feedline rupture has occurred.

If, in addition, a main feedwater control valve was assumed to close in a loop with a decreasing steam generator water level due to a control system environmental interaction, the abnormal operating characteristic of the feedwater control system would be immediately apparent to the operator. After observing the abnormal operating characteristics, the operator would immediately initiate corrective action to restore main feedwater flow and, if not successful, manually trip the reactor. Provided that the operator manually trips the reactor before the secondary liquid inventory is less than that assumed in the analysis, the Safety Analysis Report licensing basis is met.

DUQUESNE LIGHT COMPANY
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ATTACHMENT III
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION
ON THE ENVIRONMENTAL INTERACTION ISSUE

PRESSURIZER PORV CONTROL SYSTEM

Summary of Postulated Scenario

Following a feedline rupture inside containment, the pressurizer PORV control system malfunctions in such a manner that the power operated relief valves fail in the open position. Thus in addition to a feedline rupture between the steam generator nozzle and the containment penetration, a breach of the reactor coolant system boundary has occurred in the pressurizer vapor space.

Probability

Assumptions Affecting Event Probability and Consequences

a. Standard Safety Analysis Report Assumptions Concerning Feedline Break

- conservative initial assumptions

  * Appendix K decay heat model

  * Engineered safeguards power plus calorimetric error

  * Programmed RCS temperature plus control deadband and instrument errors

  * initial conservative S/G inventory

  * conservative core physics

- conservative accident assumptions

  * break (all sizes) in Safety Class 2 feedline piping

  * maximum adverse environmental errors for protective instrumentation

  * worst single active failure (loss of any one auxiliary feed pump)

  * operator action time

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b. Additional Assumptions Required for this Scenario

- break must occur inside the containment between the steam generator nozzle and the containment penetration. A break at other locations invalidates this scenario. See Attachments I and II for a description of the feedwater system piping inside containment.

- double ended break leads to limiting consequences.

  Smaller breaks permit longer operator action times.

- adverse environment resulting from the break can impact the pressurizer power operated relief valve control system. The PORVs are air to open valves with the air supply being controlled by solenoid operated valves. The PORVs and SOVs are in containment. The PORVs are designed to fail close on loss of air and power. In the plant system, prior to each of the three (3) PORV's are three (3) motor operated valves. The MOVs and associated control systems are qualified for post accident environmental conditions.

- due to the adverse environment the pressurizer PORV control system initiates a spurious signal to open the PORVs. Should the control system continue to operate within specification or initiate a spurious signal to close the PORV's the scenario is invalidated.

- should the PORV's fail to the safe position (i.e. closed) the scenario is invalidated.

Accident Consequences

Section 4.2 of WCAP-9600, Report on Small Break Accidents for Westinghouse NSSS Systems, describes transient analyses for a postulated loss of all main and auxiliary feedwater (no pipe rupture). The results indicate that, in the event the operator cannot restore auxiliary feedwater to the steam generator, the operator is required to open the pressurizer PORV's within 2,500 seconds to maintain adequate core coolant inventory.

The interaction scenario postulated above is similar to that presented in Section 4.2 of WCAP-9600. The additional assumptions made are the following:

a. a feedline rupture is assumed to occur between the steam generator nozzle and the containment penetration.

b. auxiliary feedwater is injected into the intact steam generator following the feedline rupture.

Conservatively assuming that all liquid inventory in the steam generator associated with the ruptured feedline is lost via the rupture without removing any heat (i.e., liquid blowdown), the loss of heat sink due to the liquid inventory blowdown of the ruptured steam generator is more than counterbalanced by the auxiliary feedwater being injected into the intact steam generators following reactor trip. Therefore, the results of the analyses present in WCAP-9600, Section 4.2, which illustrates that the operator is not required to take corrective action for at least 2,500 seconds following the loss of feedwater also applies to this scenario. No Safety Analysis Reports assume greater than 30 minute operator action following a feedline rupture.

Recommended Solution

The operator will be alerted to the possibility of the pressurizer PORV's failing in the open position following a high energy line rupture inside containment. After identifying a high energy line rupture inside containment, the operator will be instructed to check for an open PORV and if the PORV is not required to be open, close the MOV block valves.

Operating Instructions already instruct the operator to close the pressurizer PORV's after a primary high energy line rupture is diagnosed.

After the operator closes the PORV relief line block valves, the actions recommended in the Westinghouse Reference Operating Instructions continue to be applicable. No additional actions are required to mitigate the consequences of this scenario.

DUQUESNE LIGHT COMPANY
Beaver Valley Power Station
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ATTACHMENT IV
RESPONSE TO NRC REQUEST FOR ADDITIONAL INFORMATION
ON THE ENVIRONMENTAL INTERACTION ISSUE

ROD CONTROL SYSTEM

Summary of Postulated Scenario

Following an intermediate steamline rupture inside containment, the automatic rod control system exhibits a consequential failure due to an adverse environment which causes the control rods to begin stepping out prior to receipt of a reactor trip signal on overpower delta-T. This scenario results in a lower DNB ratio than presently presented in Safety Analysis reports.

Probability

Assumptions Affecting Event Probability and Consequences

a. Standard Safety Analysis Report Assumptions Concerning Steamline Break

- conservative initial assumptions

  * nominal rated power plus calorimetric error

  * Programmed RCS temperature plus control deadband and instrument errors

  * conservative end of life core physics

- conservative accident assumptions

  * break (all sizes) in Safety Class 2 steamline piping

  * maximum adverse environmental errors for protective instrumentation

  * worst single active failure (loss of any one Safety Injection pump)

  * operator action time

b. Additional Assumptions Required for this Scenario

- break must occur inside the containment between the steam generator nozzle and the containment penetration. A break at other locations invalidates this scenario. The main steam lines in containment are 32" electric fusion welded pipe, A155 CL.1 Gr CMS 75. Wall thickness for this pipe is .970 inches minimum. Three (3) individual steam lines, one (1) from each generator, feed a common steam header outside containment.

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Probability (continued)

- intermediate steamline breaks (0.1 to 0.25 sq. ft. per loop) at power levels from 70 to 100 percent. Other break sizes and power levels invalidate the scenario.

- adverse environment from the break can impact the nuclear instrumentation system (NIS) equipment (i.e. excore neutron detectors, cabling connectors, etc.) prior to reactor trip (i.e. within 2 minutes). Should the NIS equipment not be affected until after reactor trip (i.e. later than 2 minutes) the scenario is invalidated.

- due to the adverse environment the NIS system initiates a spurious low power signal without causing a reactor trip on negative flux rate. Should the NIS continue to operate within specification, initiate a spurious high power signal or cause a reactor trip on negative rate the scenario is invalidated.

Accident Consequences

A typical bounding analysis of the intermediate steamline rupture was performed to calculate the extent of fuel damage due to rod control system withdrawal prior to reactor trip. Based upon the reduction in radial peaking factor with burn-up and conservative end-of-life physics parameters, no fuel damage was calculated to occur following the intermediate steamline rupture with a consequential rod control system failure.

Recommended Solutions

As discussed above, a generic intermediate steamline rupture inside containment which results in control rod withdrawal due to a control system environmental interaction prior to reactor trip was analyzed. The results of the analysis indicated that no fuel damage occurred, which is consistent the assumptions made in the applicable Safety Analysis Reports.

Presently, the Beaver Valley Power Station is operating the rod control system with the power mismatch circuit defeated. The Reactor Rod Control System is in automatic on the average temperature signal. RTD's which generate the signal are qualified for post accident environment Currently, this scenario is not valid.